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

February 24, 2012

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

SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING RELOCATION OF CYCLE SPECIFIC PARAMETERS TO THE CORE OPERATING LIMITS REPORT (TAC NOS. ME5721 AND ME5722)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 247 to Renewed Facility Operating License No. DPR-31 and Amendment No. 243 to Renewed Facility Operating License No. DPR-41 for the Turkey Point Plant, Units Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 21, 2011, supplemented by letter dated July 21, 2011.

The amendments would relocate selected figures and values from the TSs to the Core Operating Limits Report (COLR) including TS Figure 2.1-1 cited in TS 2.1.1, selected portions of Note 1 on Overtemperature ΔT and Note 3 on Overpower ΔT cited in TS Table 2.2-1, TS Figure 3.1-1 cited in TS 3/4.1.1.1, Shutdown Margin value cited in TS 3/4.1.1.2, Moderator Temperature Coefficient values cited in TS 3/4.1.1.3, and Departure from Nucleate Boiling values cited in TS 3.2.5. The description of the COLR in TS 6.9.1.7 is also revised to refelct these proposed changes. The affected TS figures and technical limits cited above are only being relocated to the COLR and are not being changed under these license amendments.

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

Sincerely,

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

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 247 to DPR-31
2. Amendment No. 243 to DPR-41
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT, UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 247
Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated February 21, 2011, supplemented by letter dated July 21, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the 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. 247 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: February 23, 2012

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 243
Renewed License No. DPR-41

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

    A. The application for amendment by Florida Power and Light Company (the licensee) dated February 21, 2011, supplemented by letter dated July 21, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 243 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: February 23, 2012

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 247 RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 243 ENEWED FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace Page 3 of Renewed Operating License DPR-31 with the attached Page 3.

Replace Page 3 of Renewed Operating License DPR-41 with the attached Page 3.

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

Remove pages	Insert pages
iii	iii
iv	iv
xvi	xvi
2-1	2-1
2-2	2-2
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
3/4 1-1	3/4 1-1
3/4 1-3	3/4 1-3
3/4 1-4	3/4 1-4
3/4 1-5	3/4 1-5
3/4 1-6	3/4 1-6
3/4 2-16	3/4 2-16
6-21	6-21
6-21a	6-21a
6-22	6-22
6-22a	6-22a

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

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

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

A. Maximum Power Level

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

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 247 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 2300 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 243 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.

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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ADMINISTRATIVE CONTROLS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure specified in the COLR, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits, for 3 loop operation; and the following Safety Limits shall not be exceeded:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for the WRB-1 DNB correlation.

b. The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT (Those values denoted with [*] are specified in the COLR.)

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \frac{1}{(1+\tau_6 S)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT	=	Measured ΔT by RTD Instrumentation
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead/Lag compensator on measured ΔT; τ_1 = [*]s, τ_2 = [*]s
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT; τ_3 = [*]s
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
K_1	=	[*];
K_2	=	[*]/°F;
$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4 , τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = [*]s, τ_5 = [*]s;
T	=	Average temperature, °F;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg}; τ_6 = [*]s
T'	≤	[*] °F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	[*]/psig;
P	=	Pressurizer pressure, psig;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P' $\geq$ [*] psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

And $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between [*]% and + [*]%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds [*]%, that ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER.

NOTE 2: The channels maximum trip setpoint shall not exceed its computed setpoint by more than 0.84% of instrument span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT (Those values denoted with [*] are specified in the COLR.)

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 S}{1+\tau_7 S} \left(\frac{1}{1+\tau_6 S}\right) T - K_6 \left[T \frac{1}{1+\tau_6 S} - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	As defined in Note 1,
$\frac{1}{1+\tau_3 S}$	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	≤	[*],
K_5	≥	[*]/°F for increasing average temperature and [*] for decreasing average temperature,
$\frac{\tau_7 s}{1+\tau_7 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_7 \geq$ [*]s,
$\frac{1}{1+\tau_6 S}$	=	As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 =		[*]/°F for T > T"
	=	[*] for T ≤ T",
T	=	As defined in Note 1,
T"	≤	[*]°F (Nominal T_{avg} at RATED THERMAL POWER)
S	=	As defined in Note 1, and
f_2 (ΔI) =		[*]

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than 0.96% of instrument span.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

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

ACTION:

With the SHUTDOWN MARGIN not within limits, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the limits specified in the COLR:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

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REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limit specified in the COLR.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN not within the limit, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be within the limit specified in the COLR:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

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 x 10^{-5} $\Delta k/k/°F$ 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 $\Delta k/k/°F$ 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; and

3. 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 MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

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

** See Special Test Exceptions Specification 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg} is less than or equal to the limit specified in the COLR

b. Pressurizer Pressure is greater than or equal to the limit specified in the COLR*, and

c. Reactor Coolant System Flow ≥ 264,000 gpm

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less then 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 RCS flow rate shall be monitored for degradation at least once per 12 hours.

4.2.5.3 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.4 After each fuel loading, and at least once per 18 months, the RCS flow rate shall be determined by precision heat balance after exceeding 90% RATED THERMAL POWER. The measurement instrumentation shall be calibrated within 90 days prior to the performance of the calorimetric flow measurement. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

PEAKING FACTOR LIMIT REPORT

6.9.1.6 The W(Z) function(s) for Base-Load Operation corresponding to a ± 2% band about the target flux difference and/or a ± 3% band about the target flux difference, the Load-Follow function $F_z(Z)$ and the augmented surveillance turnon power fraction P_T shall be provided to the U.S. Nuclear Regulatory Commission, whenever P_T is <1.0. In the event, the option of Baseload Operation (as defined in Section 4.2.2.3) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the Peaking Factor Limit Report shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation, unless otherwise approved by the Commission.

The analytical methods used to generate the Peaking Factor limits shall be those previously reviewed and approved by the NRC. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

CORE OPERATING LIMITS REPORT

6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1. Reactor Core Safety Limits for Specification 2.1.1.
2. Overtemperature ΔT, Note 1 of Table 2.2-1 for Specification 2.2.1, determination of values K_1, K_2, K_3, T', P', τ_1, τ_2, τ_3, τ_4, τ_5, τ_6, and the breakpoint and slope values for the f_1 (ΔI).
3. Overpower ΔT, Note 3 of Table 2.2-1 for Specification 2.2.1, determination of values for K_4, K_5, K_6, T'', τ_7 and f_2 (ΔI).
4. Shutdown Margin - T_{avg} >200°F for Specification 3/4.1.1.1.
5. Shutdown Margin - T_{avg} ≤200°F for Specification 3/4.1.1.2.
6. Moderator Temperature Coefficient for Specification 3/4.1.1.3.
7. Axial Flux Difference for Specification 3.2.1.
8. Control Rod Insertion Limits for Specification 3.1.3.6.
9. Heat Flux Hot Channel Factor - $F_Q(Z)$ for Specification 3/4.2.2.
10. All Rods Out position for Specification 3.1.3.2.
11. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.
12. DNB Parameters for Specification 3.2.5, determination of values for Reactor Coolant System T_{avg} and Pressurizer Pressure.

The analytical methods used to determine the AFD limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983.

2. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974.

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine F_Q (Z), $F_{\Delta}H$ and the K(Z) curve shall be those previously reviewed and approved by the NRC in:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.

2. WCAP-10054-P-A, (proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

3. WCAP-10054-P, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection in the Broken Loop and Improved Condensation Model," October 1995.*

4. WCAP-12945-P, "Westinghouse Code Qualification Document For Best Estimate LOCA Analysis," Volumes I-V, June 1996.**

5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,' " June 28, 1996.**

6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."**

7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

The analytical methods used to determine Overtemperature ΔT and Overpower ΔT shall be those previously reviewed and approved by the NRC in:

1. WCAP-8745-P-A, " Design Basis for the Thermal Overtemperature ΔT and Overpower ΔT Trip Functions, " September 1986

2. WCAP-9272-P-A, " Westinghouse Reload Safety Evaluation Methodology," July 1985

The analytical methods used to determine Safety Limits, Shutdown Margin - $T_{avg} > 200°F$, Shutdown Margin - $T_{avg} \leq 200°F$, Moderator Temperature Coefficient, DNB Parameters, Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants."

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

The AFD, $F_Q(Z)$, $F_{\Delta}H$, K(Z), Safety Limits, Overtemperature ΔT, Overpower ΔT, Shutdown Margin - $T_{avg} > 200°F$, Shutdown Margin -$T_{avg} \leq 200°F$, Moderator Temperature Coefficient, DNB Parameters, and Rod Bank Insertion Limits shall be determined such that all applicable limits of the safety analyses are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector, unless otherwise approved by the Commission.

* This reference is only to be used subsequent to NRC approval.

**As evaluated in NRC Safety Evaluation dated December 20, 1997.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

Note: Report items i, j, and k are applicable following completion of inspections performed through Refueling Outage 25 at Unit 3 (and any inspection performed in the next operating cycle) and Refueling Outage 25 at Unit 4 (and any inspections performed in the subsequent operating cycles until the next scheduled inspection).

i. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

j. The calculated accident induced leakage rate from the portion of the tubes below 17.28 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

k. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 247 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-31 AND

AMENDMENT NO. 243 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT PLANT, UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated February 21, 2011 (ML110550160), as supplemented by letter dated July 21, 2011 (ML11203A501), Florida Power and Light Co. (the licensee) proposed an amendment to the Technical Specifications (TSs) for Turkey Point Plant, Unit Nos. 3 and 4. The requested changes would relocate selected figures and values from the TSs to the Core Operating Limits Report (COLR) including TS Figure 2.1-1 cited in TS 2.1.1, selected portions of Note 1 on Overtemperature ΔT and Note 3 on Overpower ΔT cited in TS Table 2.2-1, TS Figure 3.1-1 cited in TS 3/4.1.1.1, Shutdown Margin value cited in TS 3/4.1.1.2, Moderator Temperature Coefficient values cited in TS 3/4.1.1.3, and Departure from Nucleate Boiling values cited in TS 3.2.5. The description of the COLR in TS 6.9.1.7 is also revised to reflect these proposed changes. The affected TS figures and technical limits cited above are only being relocated to the COLR and are not being changed under these license amendments.

The supplement dated July 21, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 31, 2011 (76 FR 31374).

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36. The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls. The requirements for system operability during movement of irradiated fuel are included in the TSs in accordance with 10 CFR 50.36(c)(2), "Limiting Conditions for Operation."

The four criteria defined in 10 CFR 50.36 to be used in determining whether a particular limiting condition for operation (LCO) and related surveillance is required to be included in the TSs are as follows:

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

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

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

4. a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

If a particular LCO does not meet or satisfy one of the above four criteria then the TS may be relocated to other licensee controlled documents.

Guidance on the relocation of cycle-specific TS parameters to the COLR was developed by the Nuclear Regulatory Commission (NRC) staff. This guidance was provided to all power reactor licensees and applicants by Generic Letter (GL) 88-16, dated October 3, 1988. In addition, the proposed TS changes follow the guidelines presented in WCAP-1 4483-A, "Generic Methodology for Expanding Core Operating Limits Report," which was accepted for referencing by the staff on January 19, 1999.

GL 88-16 specified the actions required to move the values of cycle-specific parameter limits to the COLR from the individual TSs. The three actions are:

1. The addition of the definition of a named formal report that includes the values of cycle-specific parameter limits that have been established using an NRC-approved methodology and consistent with all applicable limits of the safety analysis,

2. The addition of an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information, and

3. The modification of individual TSs to note that cycle-specific parameters shall be maintained within the limits provided in the defined formal report.

3.0 <u>TECHNICAL EVALUATION</u>

The license amendment requested the following changes:

1. TS 2.1.1, "Safety Limits," relocation of Figure 2.1-1, "Reactor Core Safety Limit - Three Loops in Operation," to the COLR is enabled by the insertion of fuel

departure-from-nucleate-boiling (DNB) correlation design basis limit and peak fuel centerline temperature design basis limit in the TS;

2. TS 2.2.1, "Limiting Safety System Settings," relocation of the overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) T' and T" nominal T_{avg} at Rated Thermal Power values, P' nominal Reactor Coolant System pressure value, K constant values, dynamic compensation tau (τ) values, and the breakpoint and slope values for the f(ΔI) penalty function(s) in TS Table 2.2-1 to the COLR;

3. TS 3/4.1.1.1, "Boration Control Shutdown Margin – T_{avg} Greater Than 200°F," relocation of Figure 3.1-1, "Required Shutdown Margin vs. Reactor Coolant Boron Concentration," to the COLR;

4. TS 3/4.1.1.2, "Boration Control Shutdown Margin – T_{avg} Less Than or Equal To 2000°F," relocation of shutdown margin limit to the COLR;

5. TS 3/4.1.1.3, "Moderator Temperature Coefficient," relocation of the Moderator Temperature Coefficient (MTC) limits to the COLR;

6. TS 3.2.5, "DNB Parameters," relocation of Reactor Coolant System T_{avg} and Pressurizer Pressure limits to the COLR; and

7. TS 6.9.1.7, "Core Operating Limits Report," is revised to reflect the above changes.

3.1 TS 2.1.1, "Safety Limits," Figure 2.1-1

The relocation of the reactor core safety limit curves to the COLR and their replacement by the DNB ratio limits and peak fuel centerline temperature limits in the safety limit TS was approved by the staff as described in WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report." The licensee correctly applied the methodology. The COLR reactor core safety limit curves will be referenced by TS 2.1.1. Therefore, these proposed changes are acceptable.

3.2 TS 2.2.1, "Limiting Safety System Settings"

The licensee proposed to relocate the overtemperature and overpower ΔT setpoint parameters and function modifiers to the COLR. The licensee's basis for relocating these parameters to the COLR is consistent with WCAP-14483-A and is, therefore, acceptable.

3.3 TS 3/4.1.1.1, "Boration Control Shutdown Margin – T_{avg} Greater Than 200°F," TS 3/4.1.1.2, "Boration Control Shutdown Margin – T_{avg} Less Than or Equal To 2000 °F," and TS 3/4.1.1.3, "Moderator Temperature Coefficient"

Relocating these parameters to the COLR is consistent with WCAP-14483-A and was previously approved for Westinghouse plants in NUREG-1431, "Westinghouse Plants, Revision 3, Standard Technical Specifications." In addition, these parameters, as well as the NRC-approved analytical methods used to determine them, will be referenced in the COLR section of the Turkey Point TSs (TS 6.9.1.7). Therefore, the relocation of these parameters to the Turkey Point COLR is acceptable.

3.4 TS 3.2.5, "DNB Parameters"

The licensee proposed to relocate the cycle-specific DNB parameters related to reactor coolant system (RCS) temperature, pressure, and flow from TS 3.2.5 to the COLR. As stated in WCAP-14483-A, the relocation of these parameters to the COLR has been approved by the staff with the provision that the minimum staff-approved flow limits be retained in the TSs. Therefore, the licensee has not proposed to relocate the following RCS design minimum flow limits in TS 3.2.5:

- RCS flow $\geq$ 264,000 gpm

These proposed changes conform to WCAP-14483-A and are, therefore, acceptable.

3.5 TS 6.9.1.7, "Core Operating Limits Report"

The proposed change incorporates all of the above changes into the COLR including the NRC-approved methodologies that are the basis for acceptance of the above changes. This change will satisfy the TS 6.9.1.7 requirement to reference applicable methodologies; therefore, this change is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (76 FR 31374). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The NRC review established that the contents of the TSs are not diminished by the proposed changes. The proposed changes have been previously reviewed on a generic basis and found acceptable by the staff. In addition, the requirements of administrative TS 6.9.1.7 for referencing the applicable methodologies are also satisfied. The staff finds the proposed changes acceptable because they satisfy applicable regulatory requirements.

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

Principal Contributor: J. Gall, NRR

Date: February 23, 2012

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

Sincerely,

/RA/

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

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 247 to DPR-31
2. Amendment No. 243 to DPR-41
3. Safety Evaluation

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