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

October 31, 2011

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

SUBJECT: TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING FUEL CRITICALITY ANALYSIS (TAC NOS. ME4470 AND ME4471)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 246 to Renewed Facility Operating License No. DPR-31 and Amendment No. 242 to Renewed Facility Operating License No. DPR-41 for the Turkey Point Nuclear Plant, Units Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 5, 2010, supplemented by letters dated February 22, May 20, September 14, and September 22, 2011.

The amendments revise TS 5.5.1 Fuel Storage – Criticality, to include new spent fuel storage patterns that account for both the increase in fuel maximum enrichment from 4.5 weight (wt) percent (%) U-235 to 5.0 wt% U-235 and the impact on the fuel of higher power operation proposed under the Extended Power Uprate license amendment request. Although the fuel storage has been analyzed at the higher fuel enrichment in the new criticality analysis, the fuel enrichment limit of 4.5 wt% U-235 specified in TS 5.5.1 will not be changed with the issuance of 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,

for 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. 246 to DPR-31
2. Amendment No. 242 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 NUCLEAR PLANT, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 246
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 August 5, 2010, supplemented by letters dated February 22, and May 20, 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. 246 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 by the completion of the Cycle 26 refueling outage for Unit 3.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: October 31, 2011

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR PLANT, UNIT 4

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 242
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 August 5, 2010, supplemented by letters dated February 22, and May 20, September 14, and September 22, 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. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 242 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 by the completion of the Cycle 27 refueling outage for Unit 4.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A. Broaddus

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: October 31, 2011

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 246 RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 242 RENEWED FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

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

Replace Page 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
xiv	xiv
5-5	5-5
5-6	5-6
5-7	5-7
5-8	5-8
5-9	5-9
5-10	5-10
5-11	5-11
5-12	5-12
5-13	5-13
5-14	5-14
5-15	5-15
5-16	5-16
5-17	5-17

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. 246 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. 242 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

DESIGN FEATURES

5.5 FUEL STORAGE

5.5.1 CRITICALITY

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} less than 1.0 when flooded with unborated water, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

b. A k_{eff} less than or equal to 0.95 when flooded with water borated to 500 ppm, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

c. A nominal 10.6 inch center-to-center distance for Region I and 9.0 inch center-to-center distance for Region II for the two region spent fuel pool storage racks. A nominal 10.1 inch center-to-center distance in the east-west direction and a nominal 10.7 inch center-to-center distance in the north-south direction for the cask area storage rack.

d. A maximum enrichment loading for fuel assemblies of 4.5 weight percent of U-235.

e. No restriction on storage of fresh or irradiated fuel assemblies in the cask area storage rack.

f. Fresh or irradiated fuel assemblies not stored in the cask area storage rack shall be stored in accordance with Specification 5.5.1.3.

g. The Metamic neutron absorber inserts shall have a minimum certified ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.

5.5.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.5 weight percent of U-235.

5.5.1.3 Credit for burnup and cooling time is taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks. Unless otherwise specified in accordance with Specification 5.5.1.1.f, fresh or irradiated fuel assemblies shall be stored in compliance with the following:

a. Any 2x2 array of Region I storage cells containing fuel shall comply with the storage patterns in Figure 5.5-1 and the requirements of Tables 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less reactive than that shown for the 2x2 array.

b. Any 2x2 array of Region II storage cells containing fuel shall:

   i. Comply with the storage patterns in Figure 5.5-2 and the requirements of Tables 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less reactive than that shown for the 2x2 array,

   ii. Have the same directional orientation for Metamic inserts in a contiguous group of 2x2 arrays where Metamic inserts are required, and

   iii. Comply with the requirements of 5.5.1.3.c for cells adjacent to Region I racks.

c. Any 2x2 array of Region II storage cells that interface with Region I storage cells shall comply with the rules of Figure 5.5-3.

d. Any fuel assembly may be replaced with a fuel rod storage basket or non-fuel hardware.

e. Storage of Metamic inserts or RCCAs is acceptable in locations designated as empty (water-filled) cells.

DRAINAGE

5.5.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.5.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1404 fuel assemblies in two region storage racks, and the cask area storage rack is designed and shall be maintained with a storage capacity limited to no more than 131 fuel assemblies. The total spent fuel pool storage capacity is limited to no more than 1535 fuel assemblies.

Table 5.5-1

Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See notes 1-6 for use of Table 5.5-1

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	5.66439153	-14.7363682	-7.74060457	-7.63345029	24.4656526	8.5452608	26.2860949
A2	-7.22610116	11.0284547	5.13978237	10.7798957	-20.3141124	-4.47257395	-18.0738662
A3	2.98646188	-1.80672781	-0.360186309	-2.81231555	6.53101471	2.09078914	5.8330891
A4	-0.287945644	0.119516492	0.0021681285	0.29284474	-0.581826027	-0.188280562	-0.517434342
A5	-0.558098618	0.0620559676	-0.0304713673	0.0795058096	-0.16567492	0.157548739	-0.0614152031
A6	0.476169245	0.0236575787	0.098844889	-0.0676341983	0.243843226	-0.0593584027	0.134626308
A7	-0.117591963	-0.0088144551	-0.0277584786	0.0335130877	-0.0712130368	0.0154678626	-0.0383060399
A8	0.0095165354	0.0008957348	0.0024057185	-0.0040803875	0.0063998706	-0.0014068318	0.0033419846
A9	-47.1782783	-20.2890089	-21.424984	14.6716317	-41.1150	-0.881964768	-12.1780
A10	33.4270029	14.7485847	16.255208	-10.0312224	43.9149156	9.69128392	23.6179517
A11	-6.11257501	-1.22889103	-1.77941882	5.62580894	-9.6599923	-0.18740168	-4.10815592
A12	0.490064351	0.0807808548	0.127321203	-0.539361868	0.836931842	0.0123398618	0.363908736

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$Bu = (A_1 + A_2*En + A_3*En^2 + A_4*En^3)*\exp[-(A_5 + A_6*En + A_7*En^2 + A_8*En^3)*Ct] + A_9 + A_{10}*En + A_{11}*En^2 + A_{12}*En^3$$

2. Initial enrichment, En, is the nominal central zone U-235 enrichment. Axial blanket material is not considered when determining enrichment. Any enrichment between 2.0 and 5.0 may be used.
3. Cooling time, Ct, is in years. Any cooling time between 0 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.
4. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.
5. Category I-2 is fresh unburned fuel that obeys the IFBA requirements in Table 5.5-4 or contains an equivalent amount of another burnable absorber.
6. This Table applies for any blanketed fuel assembly.

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Table 5.5-2

Non-Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See notes 1-4 for use of Table 5.5-2

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	2.04088171	-27.6637884	-11.2686777	20.7284208	29.8862876	-83.5409405	35.5058622
A2	-4.83684164	26.1997193	2.0659501	11.9673275	-37.0771132	94.7973724	-30.1986997
A3	2.59801889	-7.2982252	2.66204924	-14.4072388	16.3986049	-31.9583373	11.0102438
A4	-0.300597247	0.723731768	-0.513334362	2.83623963	-2.1571669	3.55898487	-1.27269125
A5	-0.610041808	0.401332891	-0.0987986108	-1.49118695	1.02330848	0.299948492	1.34723758
A6	0.640497159	-0.418616707	-0.0724198633	1.75361041	-1.21889631	-0.312341996	-1.19871392
A7	-0.219000712	0.144304039	0.106248806	-0.659046438	0.467440882	0.107463895	0.352920811
A8	0.0252870451	-0.0154239536	-0.0197359109	0.080884618	-0.0560129443	-0.0108814287	-0.0325155213
A9	-4.48207836	-5.54507376	-1.34620551	-245.825283	12.1549	39.4975573	-5.2576
A10	-2.12118634	-5.76555416	-10.1728821	243.59979	-22.7755385	-50.5818253	10.1733379
A11	2.91619317	6.29118025	8.71968815	-75.7805818	14.3755458	23.3093829	0.369083041
A12	-0.196645176	-0.732079719	-1.14461356	8.10936356	-1.80803352	-2.69466612	0.0443577624

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

   $$Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\exp[-(A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3$$

2. Initial enrichment, En, is the nominal U-235 enrichment. Any enrichment between 1.8 and 4.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 15 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. This Table applies only for pre-EPU non-blanketed fuel assemblies. If a non-blanketed assembly is depleted at EPU conditions, none of the burnup accrued at EPU conditions can be credited (i.e., only burnup accrued at pre-EPU conditions may be used as burnup credit).

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Table 5.5-3

Fuel Categories Ranked by Reactivity
See notes 1-5 for use of Table 5.5-3

Region I	I-1	High Reactivity
	I-2	
	I-3	
	I-4	Low Reactivity
Region II	II-1	High Reactivity
	II-2	
	II-3	
	II-4	
	II-5	Low Reactivity

Notes:

1. Fuel Category is ranked by decreasing order of reactivity without regard for any reactivity-reducing mechanisms, e.g., Category I-2 is less reactive than Category I-1, etc. The more reactive fuel categories require compensatory measures to be placed in Regions I and II of the SFP, e.g., use of water filled cells, Metamic inserts, or full length RCCAs.

2. Any higher numbered fuel category can be used in place of a lower numbered fuel category from the same Region.

3. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.

4. Category I-2 is fresh unburned fuel that obeys the IFBA requirements of Table 5.5-4 or contains an equivalent amount of another burnable absorber.

5. All Categories except I-1 and I-2 are determined from Tables 5.5-1 and 5.5-2.

Table 5.5-4

IFBA Requirements for Fuel Category I-2

Nominal Enrichment (wt% U-235)	Minimum Required Number of IFBA Pins
Enr. ≤ 4.3	0
4.3 < Enr. ≤ 4.4	32
4.4 < Enr. ≤ 4.7	64
4.7 < Enr. ≤ 5.0	80

FIGURE 5.5-1

ALLOWABLE REGION I STORAGE ARRAYS

See notes 1-8 for use of Figure 5.5-1

DEFINITION | **ILLUSTRATION**

Array I-A

Checkerboard pattern of Category I-1 assemblies and empty (water-filled) cells.

I-1	X
X	I-1

Array I-B

Category I-4 assembly in every cell.

I-4	I-4
I-4	I-4

Array I-C

Combination of Category I-2 and I-4 assemblies. Each Category I-2 assembly shall contain a full length RCCA.

I-2	I-4
I-4	I-4

I-2	I-2
I-4	I-4

I-2	I-2
I-2	I-4

I-2	I-2
I-2	I-2

Array I-D

Category I-3 assembly in every cell. One of every four assemblies contains a full length RCCA.

I-3	I-3
I-3	I-3

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.
3. Category I-2 is fresh unburned fuel that obeys the IFBA requirements in Table 5.5-4 or contains an equivalent amount of another burnable absorber.
4. Categories I-3 and I-4 are determined from Tables 5.5-1 and 5.5-2.
5. Shaded cells indicate that the fuel assembly contains a full length RCCA.
6. **X** indicates an empty (water-filled) cell.
7. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

FIGURE 5.5-2

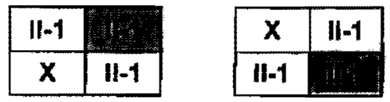
ALLOWABLE REGION II STORAGE ARRAYS

See notes 1-6 for use of Figure 5.5-2

DEFINITION — **ILLUSTRATION**

Array II-A

Category II-1 assembly in three of every four cells; one of every four cells is empty (water-filled); the cell diagonal from the empty cell contains a Metamic insert or full length RCCA.

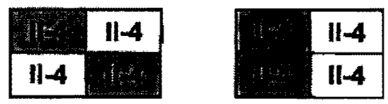


Array II-B

Checkerboard pattern of Category II-3 and II-5 assemblies with two of every four cells containing a Metamic insert or full length RCCA.

Array II-C

Category II-4 assembly in every cell with two of every four cells containing a Metamic insert or full length RCCA.



Array II-D

Category II-2 assembly in every cell with three of every four cells containing a Metamic insert or full length RCCA.

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 5.5-1 and 5.5-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
6. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

FIGURE 5.5-3

INTERFACE RESTRICTIONS BETWEEN REGION I AND REGION II ARRAYS

See notes 1-8 for use of Figure 5.5-3

DEFINITION	ILLUSTRATION
Array II-A, as defined in Figure 5.5-2, when placed on the interface with Region I shall have the empty cell in the row adjacent to the Region I rack.	see Array II-A below
Arrays II-B, II-C and II-D, as defined in Figure 5.5-2, when placed on the interface with Region I shall have an insert in every cell in the row adjacent to the Region I rack.	see Arrays II-B, II-C and II-D below

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-1	X	II-1	X
II-1	II-1	II-1	II-1

Array II-A

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-3	II-5	II-3	II-5
II-5	II-3	II-5	II-3

Array II-B

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
[illegible]	[illegible]	[illegible]	[illegible]
II-4	II-4	II-4	II-4

Array II-C

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	II-2	[illegible]	II-2

Array II-D

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 5.5-1 and 5.5-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only. Region I Array I-B is depicted as the example; however, any Region I array is allowed provided that
    a. For Array I-D, the RCCA shall be in the row adjacent to the Region II rack, and
    b. Array I-A shall not interface with Array II-D.
6. If no fuel is stored adjacent to Region II in Region I, then the interface restrictions are not applicable.
7. Figure 5.5-3 is applicable only to the Region I - Region II interface. There are no restrictions for the interfaces with the cask area rack.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 246 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-31 AND

AMENDMENT NO. 242 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR PLANT, UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated August 5, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102220031), as supplemented by letters dated February 22 (ML110560338), May 20 (ML111450582), September 14 (ML11259A007), and September 22, 2011 (ML11269A213), the Florida Power and Light Company (FPL, the licensee) proposed an amendment to the Technical Specifications (TSs) for Turkey Point, Units 3 and 4. The requested changes would revise TS 5.5.1 Fuel Storage – Criticality, to include new spent fuel storage patterns that account for both the increase in fuel maximum enrichment from 4.5 weight (wt) percent (%) U-235 to 5.0 wt% U-235 and the impact on the fuel of higher power operation proposed under the Extended Power Uprate (EPU) license amendment request (LAR). Although the fuel storage has been analyzed at the higher fuel enrichment in the new criticality analysis, the fuel enrichment limit of 4.5 wt% U-235 specified in TS 5.5.1 will not be changed with the issuance of these license amendments.

The proposed LAR revises the current licensing basis analysis for both new fuel and spent fuel storage. The evaluation credits neutron absorber inserts placed into the Region II racks to partially offset an assumed full loss of Boraflex. Credit is taken for the presence of soluble boron in the spent fuel pool (SFP) and for the presence of full-length rod cluster control assemblies (RCCAs) placed in selected fuel assemblies, as well as for the presence of Integrated Fuel Burnable Absorber (IFBA) rods in fresh fuel evaluations. FPL submitted Westinghouse Report, WCAP-17094-P, Revision 3, documenting Turkey Point's criticality analysis.

The supplements dated February 22, May 20, September 14, and September 22, 2011 provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 5, 2010 (75 FR 61527).

2.0 BACKGROUND

On January 27, 2006 (ML060900250), FPL submitted an LAR, Boraflex Remedy, to eliminate the need to credit Boraflex neutron absorbing material for reactivity control in the spent fuel pools through the use of analyzed new spent fuel storage patterns and Metamic rack inserts. Metamic neutron absorber material is a metal matrix composite consisting of a matrix of 6061 aluminum alloy reinforced with Type 1 ASTM C-750 boron carbide. On July 17, 2007 (ML071800198), the NRC staff approved the Boraflex Remedy LAR in which reactivity control is performed by a combination of RCCAs, Metamic rack inserts, open water holes, and administrative controls that require mixing higher reactivity fuel with lower reactivity fuel. FPL completed full implementation of the Boraflex Remedy LAR in 2010 for Turkey Point, Units 3 and 4.

On August 5, 2010, the licensee submitted a Fuel Storage Criticality Analysis LAR to revise TS 5.5.1, Fuel Storage-Criticality, to include new spent fuel storage patterns that account for both the increase in fuel maximum enrichment from 4.5 wt% U-235 to 5.0 wt% U-235 and the impact on the fuel of higher power operation proposed under the EPU, currently under review by the NRC staff. The proposed TS changes were based on the results of a new criticality analysis provided as Attachment 4 to the August 5, 2010, letter, "Turkey Point Units 3 and 4 New Fuel Storage Rack and Spent Fuel Pool Criticality Analysis," WCAP-17094-P, Revision 2, dated July 2010. Although the fuel storage was analyzed at the higher fuel enrichment in the new criticality analysis, the fuel enrichment limit of 4.5 wt% U-235 specified in TS 5.5.1 will not change with the issuance of this license amendment.

On September 27, 2010, the NRC staff issued the draft Interim Staff Guidance (ISG) DSS-ISG-2010-1 (ML102220567) for public comment. The purpose of the ISG is to provide updated guidance to the NRC staff reviewer to address the increased complexity of recent SFP license application analyses and operations. On February 22, 2011, FPL supplemented its August 5, 2010, letter with a revised WCAP-17094-P to address the interim staff guidance.

3.0 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." As required by 10 CFR 50.36(c)(4), design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of this section.

The applicable regulatory requirements for criticality safety analysis for SFPs are contained in 10 CFR Part 50, Appendix A, General Design Criteria for Nuclear Power Plants Criterion 62, Prevention of Criticality in Fuel Storage and Handling and 10 CFR 50.68, Criticality Accident Requirements.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, Criterion 62 requires that:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

10 CFR Part 50.68(b) (4) requires, in part, that:

> If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

As guidance for reviewing criticality analyses, the NRC staff issued an internal memorandum on August 19, 1998 (ML003728001), for performing the review of SFP criticality analysis. This memorandum is known as the 'Kopp Letter,' after the author. The Kopp Letter provides guidance on salient aspects of a criticality analysis. The guidance is germane to boiling-water reactors and pressurized-water reactors, and to borated and unborated conditions. The NRC staff used the Kopp Letter as guidance for the review of the current Turkey Point analysis, as approved in the NRC safety evaluation dated July 17, 2007.

On September 27, 2010, the NRC staff issued the draft ISG DSS-ISG-2010-1 for public comment (ML102220567). The purpose of the ISG is to provide updated guidance to the NRC staff reviewer to address the increased complexity of recent SFP license application analyses and operations. Also, the guidance is intended to reiterate existing guidance, clarify ambiguity in existing guidance, and identify lessons learned based on recent submittals. The guidance in this draft DSS-ISG-2010-01 is used by NRC staff to review: (i) future applications; and (ii) future licensee applications for license amendments and requests for exemptions from compliance with applicable requirements. The NRC staff primarily used this ISG to review Turkey Point's criticality analysis contained in WCAP-17094-P, Revision 3, dated February 2011 to ensure that Units 3 and 4 SFPs satisfy the regulations.

Additional guidance is available in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," particularly Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3, issued March 2007. Section 9.1.1 provides the existing recommendations for performing the review of the nuclear criticality safety analysis of SFPs.

4.0 <u>TECHNICAL EVALUATION</u>

Summary

The licensee must submit a plant-specific SFP criticality analysis that includes technically supported margins. The NRC staff reviewed the analysis to ensure that the assumptions made are technically substantiated. The NRC staff reviewed the application and supplemental information to determine whether the submittal provides reasonable assurance that the regulatory requirements will be met. As discussed below, the NRC staff finds that the licensee has provided the technical information needed for the NRC staff to complete its review of the LAR.

Selection of Design Basis Assembly

The criticality analysis uses Standard Fuel Assembly (STD) fuel, during depletion and in the SFP environment, as the bounding assembly design. Section 4.1 and Tables 4.1-4.6 of WCAP-17094-P, Revision 3 provide information that supports the selection of the design basis assembly.

The licensee uses two assembly designs, the 15x15 STD and 15x15 Optimized Fuel Assembly (OFA) fuel designs. There are different variations of the OFA fuel that have been used, which are differentiated by the grids and the composition of structural materials. Models were created for each category of fuel assembly. The models were then compared and the most limiting fuel design was selected. The method used to determine the limiting assembly follows the guidance set forth in DSS-ISG-2010-01.

Depletion Analysis

Section 4.2 of WCAP-170940P, Revision 3 provides information on depletion calculations. The methodology for depleting fuel assemblies in reactor to support burnup credit in SFP criticality safety calculations includes the depletion of two dimensional unit assemblies as an infinite array in reactor core geometry with the use of the PARAGON code at the bounding core conditions. These reactor parameters include moderator temperature, fuel temperature, soluble boron concentration, and specific power and operating history.

The licensee used two criteria to define the depletion conditions for each set of isotopic concentrations. The two criteria are the region of the pool that will be used to store the assembly and whether or not the fuel contains reduced enrichment axial blankets. Three sets of isotopic concentrations were used in this analysis, a single set for nonblanketed fuel, and two sets for blanketed fuel. Of the two sets for blanketed fuel, one is used for analysis for Region I of the SFP and the other is used for analysis for Region II.

The criticality analysis used different methods for burnable absorbers depending on the fuel. For pre-EPU nonblanketed fuel, two different burnable poison insert designs were used (Pyrex and WABA) in the criticality analysis. To determine the bounding burnable absorber insert for nonblanketed fuel, three sets of PARAGON fuel lattice calculations are performed. One set was performed without any burnable absorber insert, one was performed modeling a fuel assembly containing a 20 finger WABA insert and one set was performed with a 12 finger Pyrex insert. For blanketed fuel, the maximum burnable absorber loading of all of the assemblies used in the EPU fuel management studies is a 20 finger WABA in addition to 100 IFBA rods.

Hafnium Vessel Flux Depression (HVFD) absorbers are used in a few highly burned fuel assemblies on the core periphery during the third cycle of operation. These absorber inserts are present only near the mid-plane of the fuel assembly's axial length to reduce the fluence at critical weld locations along the core vessel. The associated burnup profiles of those assemblies that contain HVFD are limiting, since the HVFDs push the neutron flux out of the middle of the core and toward the ends of the fuel during fuel depletion. The selection of limiting axial burnup profiles for pre-EPU fuel include shapes from assemblies with HVFDs. These absorbers have not been in use since Cycle 24, and they are not used in EPU fuel.

The methodology used to determine the limiting assembly follows the guidance set forth in DSS-ISG-2010-01 and was found acceptable.

Criticality Analysis

Section 4.3 of WCAP-170940-P, Revision 3 provides information on criticality calculations. The axial burnup profiles are grouped into burnup ranges for both blanketed and nonblanketed fuel. All burnup profiles are modeled with 26 axial nodes. The most limiting axial burnup profile in each range is determined by considering the burnup in the top few feet of the fuel assemblies. To determine the length of fuel from the top of the assembly that dominates the reactivity, the axial burnup profiles are compared by integrating the nodal relative burnups over several selected lengths from the top of the fuel. For each selected distance from the top of the active fuel, the burnup profiles are ordered from lowest to highest integrated burnup, with the lowest integrated burnup being the most reactive shape.

The interface requirements for WCAP-170940-P, Revision 3 are that each 2X2 array in Region I or II must match all of the analyzed arrays, and that each array must have the required number of inserts, full length RCCAs or empty cells, and that the assemblies must have at least the required burnup for the appropriate category. All allowable Region I to Region II interface configurations must meet a k_{eff} value of less than 0.99, including all biases and uncertainities. The Cask Area Rack has sufficient absorber panels that the maximum k_{eff} is much less than the limiting k_{eff} in Region I or Region II. Therefore, there were no interface loading constraints on the Cask Area Rack/Region I or II interfaces.

New fuel storage racks must meet the appropriate acceptance criteria. If the rack is fully flooded by water, k_{eff} must be less than 0.95, and if the rack is flooded by optimum reduced density water, k_{eff} must be less than 0.98. Each criterion must be met including all biases and uncertainties. Analysis has shown that 5.0 wt% U235 fuel would not meet the 95 percent acceptance criterion in the fresh fuel rack without additional absorbers. The absorber selected is 16 IFBA rods in the assembly, which is the minimum number of rods in any Westinghouse IFBA design. These designs contain IFBAs that cover more than the central 7 feet of the fuel. Four cases were modeled to show that the new fuel storage racks would be within regulations.

Section 5.7 of WCAP-170940-P, Revision 3 provides information on normal and accident conditions. The licensee has listed 15 normal conditions and 6 accident conditions that were taken into consideration in the analysis. Accident cases considered were: a misloaded fresh fuel assembly, inadvertent removal of an absorber insert, SFP temperature greater than normal operating range, loss of water gap between Region I and Region II due to a seismic event, dropped fresh fuel assembly, and misplaced fuel assembly. All of the accident calculations were performed with 1683 parts per million (ppm) of soluble boron, which bounds the TS limit of 1950 ppm of soluble boron.

For the inadvertent removal of an absorber insert accident, the licensee stated that the removal of an absorber insert from an already analyzed array is bounded by the misload because the incorrectly placed assembly will be more heavily burned than the analyzed misload case, therefore this accident is covered. For the dropped fuel assembly accident, the licensee stated that the misloaded fresh fuel assembly is far more limiting due to the assumption that the dropped assembly could land horizontally on top of the other fuel assemblies in the rack. In this case, there is significant separation between the dropped fuel assembly and the rest of the fuel assemblies due to the top nozzle, fuel rod plenum, fuel rod end plug, and the separation between the fuel rod and the top nozzle. The licensee continued by stating that it is also possible that a fuel assembly could be dropped in its location with such force that the resultant fuel assembly deforms the support structure such that more of the fuel assembly is below the absorbers. The removal of an absorber insert represents 100% of the assembly below the absorber and, as stated above, this was found by the licensee to be nonlimiting. For the misplaced fuel assembly, the licensee analyzed two cases: 1) the Cask Area Rack has a corner where there is no storage cell box such that there is only one panel of Boral separating this misplaced fuel assembly from the fuel assemblies in the Cask Area Rack; and 2) one side of the Cask Area Rack does not contain any Boral absorber because it is designed to face the pool wall. In both cases, the licensee concluded that these conditions are bounded by the misload event because any fuel assembly placed outside of the racks is surrounded by water on at least two sides as opposed to the misloaded fresh assembly surrounded by fuel on all four sides. Below is a table with a summary of the accident results.

Results of the Accident Calculations	
Accident Description	**Max keff (Including Biases and Uncertainties**
Misload into II-A	0.93981
Misload into I-A	0.82654
Loss of Region I - Region II gap	0.81340
Temperature Above Normal Operations (density = 0.96 gm/cm 3)	0.82441
Temperature Above Normal Operations (density = 0.75 gm/cm 3)	0.82785

In all cases, normal and accident conditions provided a k_{eff} that is within regulations. Based on the licensee's analysis, the NRC staff has found the normal and accident conditions acceptable.

Criticality Code Validation

The analysis methodology uses the two-dimensional transport lattice code PARAGON Version 1.2.0, as well as SCALE Version 5.1. PARAGON is used for simulation of in-reactor fuel assembly depletion and SCALE is used for reactivity determinations of fuel assemblies in the SFPs. PARAGON is generically approved for depletion calculations. SCALE 5.1 is validated by using Haut Taux de Combustion critical experiment data.

The validation of SCALE Version 5.1 to perform criticality safety calculations was performed in accordance with NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," and follows the guidance set forth in DSS-ISG-2010-01.

Additional TS Changes

During teleconferences between the NRC and FPL on September 1, and September 13, 2011, the NRC staff requested two additional TS changes to (1) clarify the allowable fuel storage configurations in TS 5.5.1.1f and (2) include the minimum areal density criteria for the Metamic™ inserts. For item 1, the current Turkey Point TS 5.5.1.1f includes a statement ". . . or configurations that have been shown to comply with Specifications 5.5.1.1a and 5.5.1.1b using the NRC-approved methodology in UFSAR [Updated Final Safety Analysis Report] Chapter 9," which is not consistent with the intent of NUREG-1431, "Westinghouse Plants, Revision 3, Standard Technical Specifications and allows the licensee to generate new SFP configurations other than those approved by the NRC as specified in Turkey Point TS 5.5.1.3a and TS 5.5.1.3b. For item 2, there is currently no TS areal density limit for the credited Metamic™ inserts to monitor degradation. Section 50.36(c)(4) of 10 CFR, Design Features states "Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of this section." By letter dated July 17, 2007, the NRC approved the Boraflex Remedy LAR, which included a licensee Metamic™ surveillance program. The program ensures that the physical and chemical properties of Metamic™ behave in a similar manner as that found in the Metamic™ qualification data. The surveillance program is incorporated in Section 16.2 of the UFSAR. Incorporating the areal density limit in the TSs will ensure that the SFP is in compliance with 10 CFR 50.36(c)(4).

By letters dated September 14, and September 22, 2011, the licensee provided revised TS changes to address items 1 and 2, respectively. The NRC staff finds the proposed TS changes acceptable because (1) the change is consistent with NUREG-1431 and (2) the areal density limit ensures compliance with 10 CFR 50.36(c)(4).

Conclusion

The licensee has demonstrated through its submittal that its methodologies used in its criticality analysis follows the guidelines set forth in DSS-ISG-2010-001 and the appropriate NUREG. After reviewing the licensee's original submittal and subsequent supplemental information, the NRC staff could determine with reasonable assurance that the proposed TS changes would comply with the regulatory requirements.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (75 FR 61527). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

Principal Contributor: Davida K. Cunanan

Date: October 31, 2011

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. 246 to DPR-31
2. Amendment No. 242 to DPR-41
3. Safety Evaluation

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