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

September 19, 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: ST. LUCIE PLANT, UNIT 2 - ISSUANCE OF AMENDMENT REGARDING NEW FUEL VAULT AND SPENT FUEL POOL NUCLEAR CRITICALITY ANALYSIS (TAC NO. ME8782)

Dear Mr. Nazar:

The Commission has issued the enclosed Amendment No. 162 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 25, 2011, as supplemented by letters dated November 4 and December 8, 2011, and April 30 and May 4 and 7, 2012.

This amendment raises the maximum fuel enrichment for fresh fuel storage from a maximum of 4.5 weight-percent uranium-235 to a maximum lattice averaged value of 4.6 weight-percent uranium-235. The TS changes associated with fuel stored in the spent fuel pool include increasing the maximum initial enrichment from 4.5 weight-percent uranium-235 to a maximum planar average initial enrichment of 4.6 weight-percent uranium-235, credit for empty storage locations, credit for use of METAMIC™ inserts, credit for installation of full-length full-strength, five-fingered control element assemblies, and definition of three special configurations referred to in the nuclear criticality safety analysis as inspection and maintenance configurations.

The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations,* Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Accordingly, the NRC staff has also prepared a redacted, publicly-available, non-proprietary version of the SE. Copies of theproprietary and non-proprietary versions of the SE are enclosed.

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,

Tracy J. Orf

Tracy J. Orf, Project Manager
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 162 to NPF-16
2. Non-Proprietary Safety Evaluation
3. Proprietary Safety Evaluation

cc w/ Enclosures 1 and 2: Distribution via Listserv

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

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 162
Renewed License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company (the licensee), dated February 25, 2011, as supplemented by letters dated November 4 and December 8, 2011, and April 30 and May 4 and 7, 2012; 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, Renewed Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 3.B to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 162, are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Jessie F. Quichocho, Acting Chief
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: September 18, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 162

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace Page 3 of Renewed Operating License NPF-16 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 vertical lines indicating the area of change.

| Remove Pages | Insert Pages |
|---|---|
| XXII | XXII |
| XXV | XXV |
| 3/4 9-12 | 3/4 9-12 |
| 5-4 | 5-4 |
| 5-4a | 5-4a |
| 5-4b | 5-4b |
| 5-4c | - |
| 5-4d | - |
| 5-4e | - |
| 5-4f | - |
| 5-4g | - |
| - | 5-4h |
| - | 5-4i |
| - | 5-4j |
| - | 5-4k |
| - | 5-4l |
| - | 5-4m |
| - | 5-4n |
| - | 5-4o |

neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required.

D. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission's regulations: 10 CFR Part 20, Section 30.34 of 10 FR Part 30, Section 40.41 of 10 CFR Part 40, Section 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

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal).

Commencing with the startup for Cycle 16 and until the Combustion Engineering Model 3410 Steam Generators are replaced, the maximum reactor core power shall not exceed 89 percent of 2700 megawatts (thermal) if:

a. The Reactor Coolant System Flow Rate is less than 335,000 gpm but greater than or equal to 300,000 gpm, or

b. The Reactor Coolant System Flow Rate is greater than or equal to 300,000 gpm AND the percentage of steam generator tubes plugged is greater than 30 percent (2520 tubes/SG) but less than or equal to 42 percent (3532 tubes/SG).

This restriction in maximum reactor core power is based on analyses provided by FPL in submittals dated October 21, 2005 and February 28, 2006, and approved by the NRC in Amendment No. 145, which limits the percent of steam generator tubes plugged to a maximum of 42 percent (3532 tubes) in either steam generator and limits the plugging asymmetry between steam generators to a maximum of 600 tubes.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 162 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

**LIST OF FIGURES (continued)**

**LIST OF TABLES (Continued)**

TABLE PAGE

## REFUELING OPERATIONS

## 3/4.9.11 SPENT FUEL STORAGE POOL

### LIMITING CONDITION FOR OPERATION

3.9.11 The Spent Fuel Pool shall be maintained with:

a. The fuel storage pool water level greater than or equal to 23 ft over the top of irradiated fuel assemblies seated in the storage racks, and

b. The fuel storage pool boron concentration greater than or equal to 1900 ppm.

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

**ACTION:**

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

b. With the boron concentration requirement not satisfied, immediately suspend all movement of fuel assemblies in the fuel storage pool and initiate action to restore fuel storage pool boron concentration to within the required limit.

c. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

4.9.11.1 Verify the fuel storage pool boron concentration is within limit at least once per 7 days.

## 5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

## 5.6 FUEL STORAGE

### CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1. A $k_{eff}$ equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

2. A $k_{eff}$ equivalent to less than or equal to 0.95 when flooded with water containing 500 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

3. A nominal 8.965 inch center-to-center distance between fuel assemblies placed in the spent fuel pool storage racks and a nominal 8.80 inch center-to-center distance between fuel assemblies placed in the cask pit storage rack.

4. For storage of enriched fuel assemblies, requirements of Specification 5.6.1.a.1 and 5.6.1.a.2 shall be met by positioning fuel in the spent fuel pool storage racks consistent with the requirements of Specification 5.6.1.c.

5. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Specifications 5.6.1.a.1 and 5.6.1.a.2.

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

b. The cask pit storage rack shall contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool.

c. Loading of spent fuel pool storage racks shall be controlled as described below.

1. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel pool storage rack shall be less than or equal to 4.6 weight percent.

2. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure 5.6-1 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

3. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions or allowed special arrangement definitions of Figure 5.6-2 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

**CRITICALITY** (continued)

5.6.1 c. (continued)

4. The 2x2 array of fuel assemblies that span the interface between Region 1 and Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure 5.6-3 and the minimum burnup requirements as defined in Table 5.6-1. The allowed special arrangements in Region 2 as shown in Figure 5.6-2 shall not be placed adjacent to Region 1. (See Specification 5.6.1.c.7 for exceptions)

5. Fuel placed in the cask pit storage rack shall comply with the storage pattern definitions of Figure 5.6-4 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

6. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.

7. Fresh or spent fuel in any allowed configuration may be replaced with non-fuel hardware, and fresh fuel in any allowed configuration may be replaced with a fuel rod storage basket containing fuel rod(s). Also, storage of Metamic inserts or control rods, without any fissile material, is acceptable in locations designated as completely water-filled cells.

d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a maximum planar average U-235 enrichment less than or equal to 4.6 weight percent, while maintaining a $k_{eff}$ of less than or equal to 0.98 under the most reactive condition.

**DRAINAGE**

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

**CAPACITY**

5.6.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1491 fuel assemblies, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 225 fuel assemblies. The total Unit 2 spent fuel pool and cask pit storage capacity is limited to no more than 1716 fuel assemblies.

## 5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

Pages 5-4C through 5-4F (Amendment 101) and page 5-4G (Amendment 135) have been deleted from the Technical Specifications.
The next page is 5-4h.

Allowable Storage Patterns
(See Notes 1 and 2)

Pattern "A"
See Definition 1

| | |
|---|---|
| F | WH |
| WH | F |

Pattern "B"
See Definition 2

| | |
|---|---|
| 1 | 1 |
| CEA 1 | 1 |

Pattern "C"
See Definition 3

| | |
|---|---|
| 2 | 2 |
| WH | 2 |

DEFINITIONS:

1. Allowable pattern is fresh or burned fuel checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fuel and WH represents a completely water-filled cell.

2. Allowable pattern is placement of fuel assemblies that meet the requirements of type 1 in each 2x2 array location with at least one full-length full-strength CEA placed in any cell. Minimum burnup for fuel assembly type 1 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

3. Allowable pattern is placement of fuel assemblies that meet the requirements of type 2 in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 2 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

NOTES:

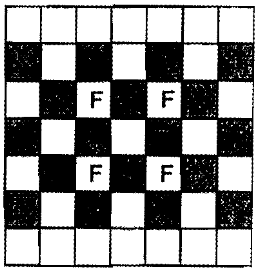
1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

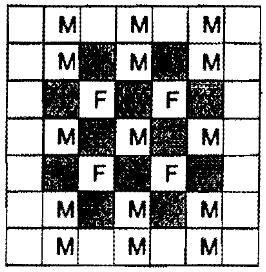
**FIGURE 5.6-1**
**Allowable Region 1 Storage Patterns and Fuel Arrangements**

ALLOWED SPECIAL ARRANGEMENTS (See Notes 1 and 2)
Fresh Fuel Assemblies in Region 2 Racks

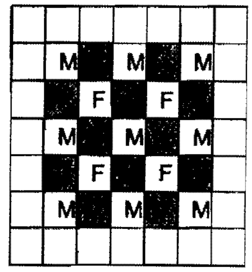
See Definition SR1




See Definition SR2




See Definition SR3




F = Fresh Fuel Assembly

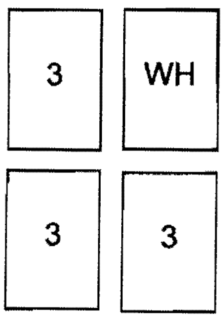
M = Fuel Assembly with Metamic Insert

= Empty Cell

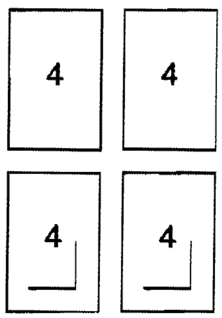
= Fuel Assembly per Definition

ALLOWABLE STORAGE PATTERNS (See Notes 1 and 2)

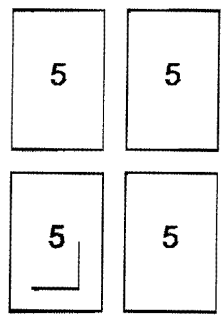
Pattern "D"
See Definition 1



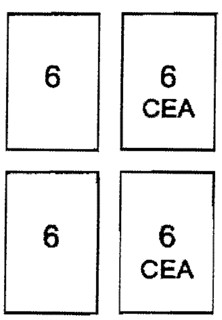

Pattern "E"
See Definition 2




Pattern "F"
See Definition 3




Pattern "G"
See Definition 4




Pattern "H"
See Definition 5

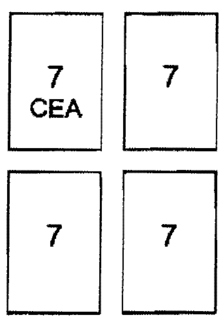



**FIGURE 5.6-2 (Sheet 1 of 3)**
**Allowable Region 2 Storage Patterns and Fuel Arrangements**

**DEFINITIONS for Figure 5.6-2**

1. Allowable pattern is fuel assemblies that meet the requirements of type 3 in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 3 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

2. Allowable pattern is fuel assemblies that meet the requirements of type 4 in each of the 2x2 array locations with at least two Metamic inserts placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 4 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

3. Allowable pattern is fuel assemblies that meet the requirements of type 5 in each of the 2x2 array locations with at least one Metamic insert placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 5 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

4. Allowable pattern is fuel assemblies that meet the requirements of type 6 in each of the 2x2 array locations with at least two full-length, full strength 5 finger CEAs placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 6 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

5. Allowable pattern is fuel assemblies that meet the requirements of type 7 in each of the 2x2 array locations with at least one full-length, full strength 5 finger CEA placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 7 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

SR1. Allowable pattern is up to four fresh or bumed fuel assemblies placed in a 3x3 array in combination with Pattern "D" placed outside the 3x3 array. Fresh or burned fuel shall be placed in the comers of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 3 shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 3 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

SR2. Allowable pattern is up to four fresh or bumed fuel assemblies placed in a 3x3 array in combination with Pattern "E" placed outside the 3x3 array. Fresh or burned fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 4 with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 4 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

**FIGURE 5.6-2 (Sheet 2 of 3)**
**Allowable Region 2 Storage Patterns and Fuel Arrangements**

SR3. Allowable pattern is up to four fresh or burned fuel assemblies placed in a 3x3 array in combination with Pattern "F" placed outside the 3x3 array. Fresh or burned fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 5 with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 5 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

**NOTES**

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

**FIGURE 5.6-2 (Sheet 3 of 3)**
**Allowable Region 2 Storage Patterns and Fuel Arrangements**

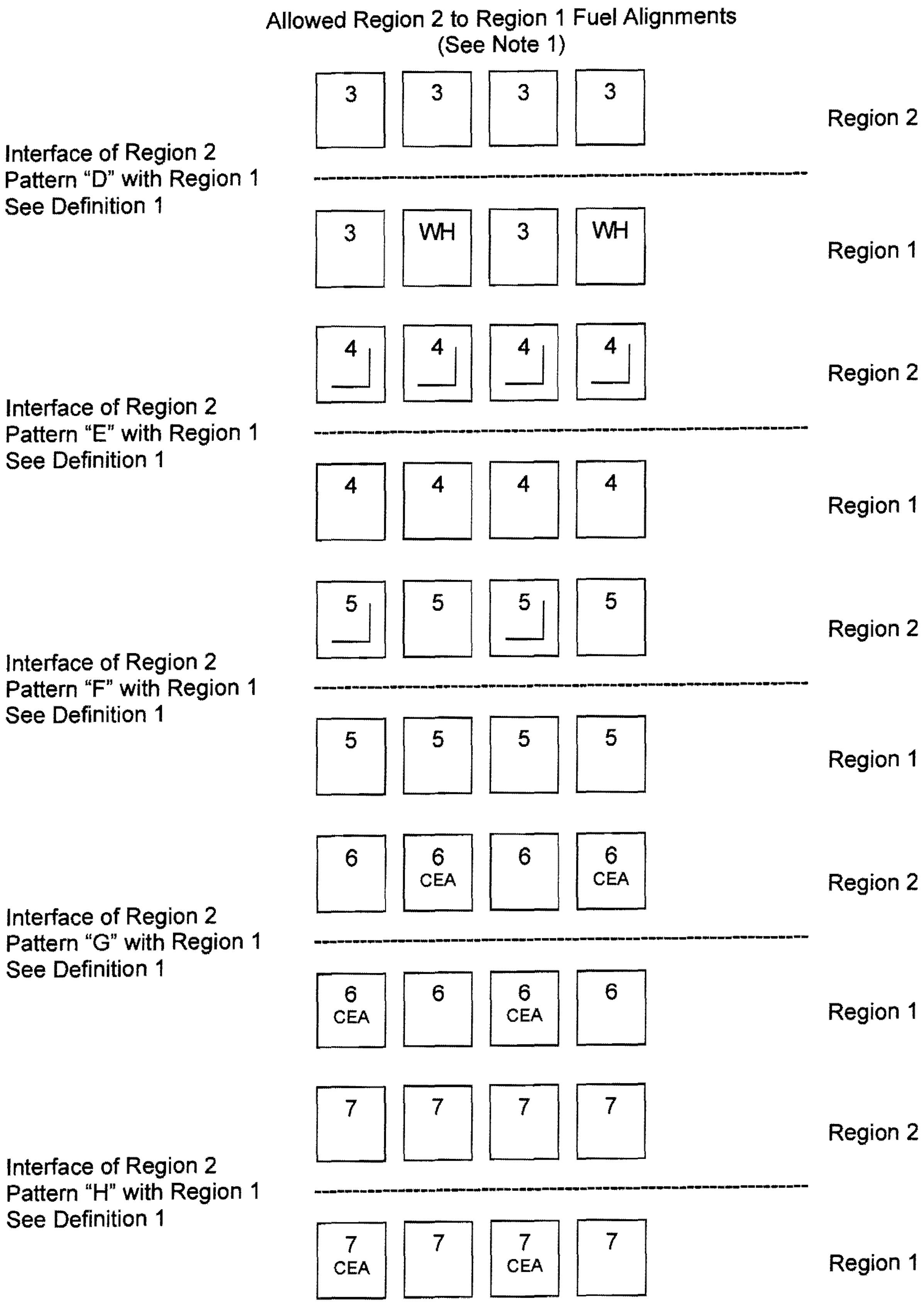



**FIGURE 5.6-3 (Sheet 1 of 2)**
**Interface Requirements between Region 1 and Region 2**

DEFINITION:

1. Each 2x2 array that spans Region 1 and Region 2 shall match one of the Region 2 allowable storage patterns as defined by Specification 5.6.1.c.3. Any required Metamic inserts must be placed into the fuel assemblies in Region 2. Locations of completely water-filled cells or CEAs may be in either Region 1 or Region 2. For interface assemblies, the requirements of Specifications 5.6.1.c.2 and Specification 5.6.1.c.3 shall be followed within Region 1 and Region 2, respectively. The Diagrams are for illustration only.

NOTES:

1. Completely water-filled cells within any pattern are acceptable.

**FIGURE 5.6-3 (Sheet 2 of 2)**
**Interface Requirements between Region 1 and Region 2**

Allowable Storage Patterns
(See Notes 1 and 2)

Pattern "A"
See Definition 1

| | |
|---|---|
| F | WH |
| WH | F |

Pattern "B"
See Definition 2

| | |
|---|---|
| 8 | 8 |
| WH | 8 |

DEFINITIONS:

1. Allowable pattern is fresh or burned fuel checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fuel and WH represents a completely water-filled cell.

2. Allowable pattern is placement of fuel assemblies that meet the requirements of type 8 in three of the 2x2 array locations in combination with one completely water-filled cell in any location. Minimum burnup for fuel assembly type 8 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

NOTES:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

**FIGURE 5.6-4**
**Allowable Cask Pit Storage Rack Patterns**

**TABLE 5.6-1**
**Minimum Burnup Coefficients**

| Fuel Type | Cooling Time (Years) | Coefficients | | |
|---|---|---|---|---|
| | | A | B | C |
| **1** | 0 | -33.4237 | 25.6742 | -1.6478 |
| **2** | 0 | -25.3198 | 14.3200 | -0.4042 |
| **3** | 0 | -23.4150 | 16.2050 | -0.5500 |
| **4** | 0 | -33.2205 | 24.8136 | -1.5199 |
| | 2.5 | -31.4959 | 23.4776 | -1.4358 |
| | 5 | -30.4454 | 22.7456 | -1.4147 |
| | 10 | -28.4361 | 21.2259 | -1.2946 |
| | 15 | -27.2971 | 20.3746 | -1.2333 |
| | 20 | -26.1673 | 19.4753 | -1.1403 |
| **5** | 0 | -24.8402 | 23.5991 | -1.2082 |
| | 2.5 | -22.9981 | 21.6295 | -1.0249 |
| | 5 | -21.8161 | 20.5067 | -0.9440 |
| | 10 | -20.0864 | 19.0127 | -0.8545 |
| | 15 | -19.4795 | 18.3741 | -0.8318 |
| | 20 | -18.8225 | 17.7194 | -0.7985 |
| **6** | 0 | -32.4963 | 25.3143 | -1.5534 |
| | 2.5 | -30.6688 | 23.6229 | -1.4025 |
| | 5 | -29.2169 | 22.5424 | -1.3274 |
| | 10 | -27.2539 | 21.0241 | -1.2054 |
| | 15 | -25.7327 | 19.8655 | -1.1091 |
| | 20 | -25.2717 | 19.5222 | -1.1163 |
| **7** | 0 | -24.6989 | 24.1660 | -1.2578 |
| | 2.5 | -23.0399 | 22.3047 | -1.0965 |
| | 5 | -21.2473 | 20.6553 | -0.9403 |
| | 10 | -20.1775 | 19.5506 | -0.9015 |
| | 15 | -19.4037 | 18.6626 | -0.8490 |
| | 20 | -18.3326 | 17.7040 | -0.7526 |
| **8** | 0 | -43.4750 | 11.6250 | 0.0000 |

NOTES:

1. To qualify in a "fuel type", the burnup of a fuel assembly must exceed the minimum burnup "BU" calculated by inserting the "coefficients" for the associated "fuel type" and "cooling time" into the following polynomial function:

   $BU = A + B*E + C*E^2$, where:

   BU = Minimum Burnup (GWD/MTU)

   E = Maximum Initial Planar Average Enrichment (weight percent U-235)

   A, B, C = Coefficients for each fuel type

2. Interpolation between values of cooling time is not permitted.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 162

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NOS. 50-389

## 1.0 INTRODUCTION

By letter dated February 25, 2011 (Reference 1), Florida Power and Light Company (the licensee) requested to amend Renewed Facility Operating License No. NPF-16 and revise the St. Lucie, Unit No. 2 (St. Lucie 2), Technical Specifications (TSs). The proposed amendment requested revisions to the Renewed Facility Operating License and TSs to support an extended power uprate (EPU) for St. Lucie, Unit 2, at a licensed core thermal power level increased from 2700 megawatts thermal (MWt) to 3020 MWt. For scheduling purposes, the review of the nuclear criticality safety (NCS) analysis for the St. Lucie 2 spent fuel pool (SFP) and new fuel vault (NFV) has been processed as a separate amendment request. The following evaluation presents the results of the U.S. Nuclear Regulatory Commission's (NRC) review of the NCS analysis.

The License Amendment Request (LAR) was amended by a letter dated November 4, 2011 (Reference 2). Attachment 3 to that letter included Revision 2 to Holtec International Report HI-2104753. In subsequent submittals, the licensee provided revised proposed technical specifications (Reference 3); responses to NRC requests for additional information (RAIs) (References 4, 5, and 6); Revision 4 to Holtec International Report HI-2104753 (HI-2104753) (Attachment 3 to Reference 6); and Holtec International Report HI-2094416, Revision 1, the NCS analysis for the St. Lucie 2 new fuel vault (Attachment 4 to Reference 6).

As part of Attachment 5 to Reference 1 and in support of the EPU, the licensee submitted Holtec International Report HI-2104753, Revision 1, "St. Lucie Unit 2 Criticality Analysis for EPU and Non-EPU Fuel," which describes the NCS analysis for the St. Lucie 2 SFP storage racks. It describes the methodology and analytical models used in the NCS analysis to show that the storage racks maximum k-effective ($k_{eff}$) will be less than 1.0 when flooded with unborated water for normal conditions, and less than or equal to 0.95 when flooded with borated water for normal and credible accident conditions at a 95-percent probability, 95-percent confidence level.

The St. Lucie 2 SFP has three fuel storage rack types, with one or more racks for each type: one Cask Pit Rack (CPR), 6 Region 1 Spent Fuel Racks (SFRs) that contain full-length nonstructural L-shaped stainless steel inserts that serve to enhance local neutron absorption, and 13 Region 2 SFRs. HI-2104753, Revision 4 proposes 10 storage configurations, 2 in the CPR, 3 in the Region I SFRs, and 5 in the Region 2 SFRs. The CPR contains and credits empty cells, fuel burnup, and the neutron absorbing material BORAL™. The Region 1 SFRs credit fuel burnup, empty cells, control rods, and steel inserts. The Region 2 SFRs credit higher burnups, empty cells, control rods, cooling time, and inserts made from the neutron absorbing material METAMIC™. The presence of soluble boron is credited in the St. Lucie 2 SFP.

New fuel assemblies may be stored in what are normally dry conditions in the St. Lucie 2 new fuel racks in the NFV. It is necessary to update the NFV NCS analysis for the EPU because the maximum planar average enrichment has been increased to 4.6 weight-percent uranium-235. No credit is taken in fresh or spent fuel storage racks for integral burnable absorbers.

## 2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, Criterion 62 requires, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

Paragraph 50.68(b)(1) of 10 CFR requires, "Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water."

Paragraph 50.68(b)(2) of 10 CFR requires, "The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95-percent probability, 95-percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used."

Paragraph 50.68(b)(3) of 10 CFR requires, "If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95-percent probability, 95-percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used."

Paragraph 50.68(b)(4) of 10 CFR requires, "If no credit for soluble boron is taken, 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 unborated water. 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."

Paragraph 50.36(c)(4) of 10 CFR requires, "Design features. 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 St. Lucie 2 SFP NCS analysis does take credit for soluble boron for normal operating conditions. Therefore, the regulatory requirement is for the St. Lucie 2 $k_{eff}$ 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 $k_{eff}$ must remain below 1.0 (subcritical), at a 95-percent probability, 95-percent confidence level, if flooded with unborated water. The St. Lucie 2 NCS uses the double contingency principle to take credit for soluble boron for abnormal/accident operating conditions.

The St. Lucie 2 NFV NCS analysis demonstrated that, consistent with the requirements of 10 CFR 50.68, the maximum $k_{eff}$ at full water density was less than 0.95, at a 95-percent probability/95-percent confidence level and that the maximum $k_{eff}$ at optimum water density, around 8 percent of full density, was less than 0.98, at a 95-percent probability, 95-percent confidence level.

## 3.0 TECHNICAL EVALUATION

### 3.1 Proposed Change

There are several proposed TS changes that either impact NCS analyses or implement changes in fuel storage requirements. The reactor operating condition changes related to the EPU affect fuel depletion, which is credited in the spent fuel pool NCS analysis. EPU-related changes impacting NCS analysis may include higher power density, fuel and moderator temperature changes, and soluble boron concentration changes.

One of the TS changes raises the maximum fuel enrichment for fresh fuel storage from a maximum of 4.5 weight-percent uranium-235 to a maximum lattice averaged value of 4.6 weight-percent uranium-235. The TS changes associated with fuel stored in the SFP include increasing the maximum initial enrichment from 4.5 weight-percent uranium-235 to a maximum planar average initial enrichment of 4.6 weight-percent uranium-235, credit for empty storage locations, credit for use of METAMIC™ inserts, credit for installation of full-length full-strength, five-fingered control element assemblies and definition of three special configurations referred to in the NCS analysis as inspection and maintenance configurations.

### 3.2 Method of Review

This safety evaluation involves a review of the NCS analyses for the SFP provided as Attachment 3 to Reference 6 and for the NFV provided as Attachment 4 to Reference 6 and the related proposed TS changes that were provided as attachments to Reference 3. The SFP NCS analysis includes the effects of the changes in reactor operating conditions associated with the EPU and supports expansion of fuel storage requirements to additional storage configurations. The review was performed consistent with Section 9.1.1 of NUREG-0800.

The staff issued an internal memorandum on August 19, 1998, containing guidance for performing the review of SFP NCS analysis (Reference 2). This memorandum is known colloquially as the 'Kopp Memo' (Reference 9), after the author. While the Kopp memorandum does not specify a methodology, it does provide some guidance on the more salient aspects of an NCS analysis, including computer code validation. The guidance is germane to boiling-water reactors and pressurized-water reactors (PWRs), borated and unborated. The Kopp memorandum has been used for virtually every light-water reactor SFP NCS analysis since, including this St. Lucie 2 analysis.

## 3.3 <u>SFP NCS Analysis Review</u>

### 3.3.1 SFP NCS Analysis Method

There is no generic or standard methodology for performing NCS analyses for fuel storage and handling. The methods used for the NCS analysis for fuel in the St. Lucie 2 SFP are described in HI-2104753, Revision 4. Additional information describing the methods used is provided in the RAI responses attached to References 4, 5, and 6. Some SFP analysis deficiencies were identified during the review, but as will be discussed below, sufficient margin is built into the analysis methodology to offset the deficiencies. Consequently, the methodology is specific to this analysis and, without further revision, is not appropriate for other applications.

#### 3.3.1.1 Computational Methods

The LAR seeks to credit fuel assembly burnup in 8 of the 10 defined configurations. The CASMO-4 computer code and its 70-group cross-section library were used to calculate burned fuel compositions and to generate lumped-fission-product cross sections for use with the MCNP-5 computer code. MCNP-5 was used, with its ENDF/B-V and ENDF/B-VI continuous energy cross sections and the CASMO-4-generated lumped fission product cross sections, to calculate $k_{eff}$ values. These computer codes and the nuclear data sets with them have been used in many NCS analyses, are industry standards, and are considered acceptable.

Although not common, the computational method used "lumped fission product" (LFP) number densities and cross sections generated by CASMO for use in MCNP. The preparation, testing, and use of LFP are described in Holtec International report HI-2033031 (Reference 7). The use of LFPs was verified by performing equivalent calculations with CASMO-4 and with MCNP-5 using the LFP number densities and cross sections. These calculations were performed to ensure that CASMO-4 cross sections were appropriately converted to the format required by MCNP-5. There is some unquantified and unvalidated uncertainty associated with using the LFP method. The spent fuel analysis includes a "5 percent of the reactivity decrement" uncertainty (Reference 9) to cover lack of validation of spent fuel compositions, including fission products, and a "**[** **]** of the minor actinide and fission product worth" uncertainty to cover the lack of validation of minor actinides and fission products for calculation of $k_{eff}$. Recent work published in NUREG/CR-7109 (Reference 8) indicates that an uncertainty of about 3 percent of the minor actinide and fission product worth should be sufficient to conservatively bound biases that may be associated with calculating $k_{eff}$ for systems with minor actinides and fission products. The uncertainties adopted for fuel composition and $k_{eff}$ calculations are large enough to cover bias and uncertainty associated with use of LFP.

During the review, a question, RAI SRXB-121, was asked concerning how source convergence was checked in the MCNP calculations. According to the RAI response, in all MCNP calculations, results for 50 skipped cycles were used. A review of the Shannon entropy convergence criteria for these calculations showed that in most cases more than 50 skipped cycles would be required to achieve convergence. The RAI response provided an assessment of the impact of revised converged calculations and noted that there was a reduction in the margin to the $k_{eff}$ limits, with the largest reduction being 0.0052 $\Delta k$. This nonconservatism is addressed in Section 3.3.5 of this report.

3.3.1.2 Computer Code Validation

Since the NCS analysis credits fuel burnup, it is necessary to consider validation of the computer codes and data used to calculate burned fuel compositions and the computer code and data that utilize the burned fuel compositions to calculate $k_{eff}$ for systems with burned fuel.

Consistent with the guidance provided in the Kopp Memo (Reference 9), the analysis has incorporated a "5 percent of the reactivity decrement" uncertainty to cover lack of validation of fuel composition calculations. This uncertainty is calculated as 0.05 times the change in $k_{eff}$ from the fresh fuel to the credited final fuel burnup. This uncertainty was calculated by the licensee and applied correctly.

The study used to support validation of $k_{eff}$ calculations using MCNP-5 was documented in Holtec International report HI-2104790 (Reference 10). The validation set included 291 critical configurations that included the French Haut Taux de Combustion (HTC) critical experiments (Reference 11), mixed uranium and plutonium critical experiments and some low enrichment uranium fuel pin experiments. During the review it was noted that the validation set included some of the HTC critical experiments that were not recommended for use, that the trending analysis did not evaluate trends in the calculated bias and bias uncertainty associated with variation of plutonium content or with uranium enrichment, and that the wrong value was used for the fresh fuel bias. Thae licensee performed additional analyses to support its response to RAI SRXB-145, showing the impact of excluding some of the HTC critical experiments, and evaluating the variation of the bias with plutonium content and uranium enrichment. This analysis showed that the bias used in the analysis was about 0.0010 too low. Considering the margins available to the regulatory limit, this 0.0010 $\Delta k$ nonconservatism is acceptable.

Appropriate critical experiment data was not available to validate $k_{eff}$ calculations crediting minor actinides (uranium-236, neptunium-237, and americium-243) or fission products. This deficiency was identified in the analysis. To address this validation deficiency, an uncertainty equal to **[ ]** of the worth of the minor actinides and fission products was adopted. Recent work published in NUREG/CR-7109 indicates that an uncertainty of about 3 percent of the minor actinide and fission product worth should be sufficient to conservatively bound biases that may be associated with minor actinides and fission products. Therefore, the **[ ]** uncertainty value adequately covers the $k_{eff}$ validation deficiencies.

3.3.2 SFP and Fuel Storage Racks

3.3.2.1 SFP Water Temperature

NRC guidance provided in the Kopp memorandum states the NCS analysis should be done at the temperature corresponding to the highest reactivity. Analysis was performed with water densities of 0.9787 g/cm$^3$ and 1.00 g/cm$^3$ and at water temperatures of 300 °K and 400 °K and at soluble boron concentrations of 0, 500, and 1000 PPM by weight. The water densities used reflect the water density range as the water temperature varies from the temperature where maximum water density occurs, near 39 °F, up to 155 °F, which is the assumed maximum normal operating temperature. The 300 °K and 400 °K temperatures were used because the MCNP continuous energy cross sections are available only at a few specific temperatures and MCNP does not do interpolation or temperature corrections. The product of the analysis was a storage configuration-dependent temperature bias to be applied to the maximum $k_{eff}$ value for that storage configuration.

The NRC staff identified two issues with the SFP water temperature bias determination. The first is that the analysis did not examine water densities between 1.00 and 0.9787 g/cm$^3$ to ensure that $k_{eff}$ did not peak at some value between the maximum and minimum water density values. Instead, engineering judgment was cited by the licensee as the basis for concluding that the calculations were conservative and acceptable. The second issue is that there is currently no way to validate the impact on $k_{eff}$ of the variation of cross sections with temperature. There is some uncertainty associated with utilizing 400 °K cross sections to quantify the impact of cross section variation at 342 °K on the $k_{eff}$ value calculated by MCNP.

From Table 7.3 of HI-2104753, after discarding bias values that would reduce the maximum $k_{eff}$ value, the calculated temperature bias values range up to 0.0034 $\Delta$k for case 10 with no soluble boron. It is unlikely that the uncertainty on the temperature corrections is as large as 100 percent. Consequently, for purposes of evaluating a balance of conservatisms and nonconservatisms, the SFP water temperature bias determination may be nonconservative by as much as 0.0034 $\Delta$k. This nonconservatism is addressed in Section 3.3.5 of this report.

3.3.2.2 SFP Storage Rack Models

Three fuel storage rack variations are used in the St. Lucie 2 SFP. All three are manufactured by fabricating boxes that are then welded together at their corners, creating a checkerboard arrangement of storage cells inside fabricated boxes and between boxes. In response to RAI SRXB-130, the licensee stated that the filler panels and corners that are used to complete the periphery of each rack module have the same thickness as the fabricated boxes and are attached such that the minimum cell inner dimension is maintained.

The CPR includes BORAL™ panels held to the exterior side of each box using a steel wrapper. A detailed representation of the model rack that explicitly models the box, BORAL™ panel, and wrapper was utilized in the analysis. The CPR model used in cases 1 and 8 was a laterally infinite array of fabricated and formed cells. Use of this laterally infinite model, which does not credit neutron leakage from the sides of the CPR rack modules, produces some unquantified margin for the CPR calculations.

The Region 1 and 2 rack modules are also constructed steel boxes that are welded together at the corners, creating formed and fabricated storage cells. However, these rack modules do not include BORAL™ panels and wrappers. A simplified model was used for the Region 1 and 2 rack modules that utilizes a single average repeated cell. This model maintains the center-to-center spacing between assemblies and the amount of steel between assemblies. This simplification should have little to no effect on the calculated $k_{eff}$ values. The only differences between the Region 1 and 2 rack modules is that an L-shaped steel insert has been placed in all Region 1 cells and that L-shaped METAMIC™ inserts may be installed in some Region 2 cells. In both the CPR and the Region 2 racks, with METAMIC™ inserts, the minimum boron-10 loading is used in the model.

The staff finds that the design basis models used to model the CPR and Region 1 and 2 racks are appropriate and conservative.

3.3.2.3 SFP Storage Rack Models Manufacturing Tolerances and Uncertainties

The minimum material thicknesses of the box walls and sheathing were used in all design basis calculations. Sensitivity calculations were performed for the CPR rack that determined the maximum center-to-center spacing yielded higher $k_{eff}$ values. For the Region 1 and 2 rack modules, the minimum cell inner dimension was used, which is conservative and therefore, acceptable.

3.3.3 Fuel Assembly

3.3.3.1 Bounding Fuel Assembly Design

The fuel assemblies used at St. Lucie 2 are all Combustion Engineering (CE) 16x16 or equivalent assemblies manufactured by other suppliers. In some of the older CE 16x16 assemblies, fuel rods were replaced with rods containing $B_4C$. More recently, the fuel assemblies have included some fuel rods with $Gd_2O_3$ mixed in with the $UO_2$ in the fuel pellets. When $Gd_2O_3$ is mixed in the fuel pellets with the $UO_2$, the uranium-235 enrichment is lower than the maximum planar average enrichment for the fuel pellets that do not have $Gd_2O_3$. In both new and old designs, the initial uranium enrichment may be varied from pin-to-pin to control fuel assembly power peaking factors. The fuel assembly design model used in the St. Lucie 2 SFP NCS analysis does not include any $B_4C$ or $Gd_2O_3$ rods and utilizes the maximum planar average enrichment for all fuel pins. This increases the amount of uranium-235 present relative to the actual fuel assemblies. Sensitivity calculations were performed to estimate the bias introduced by using the planar average enrichment rather than the actual pin-dependent enrichment variations. The licensee performed sensitivity studies that indicate that modeling the fuel assemblies as having all fuel pellets at the maximum planar average enrichment rather than explicitly modeling the $Gd_2O_3$ fuel rods and the replacement of some fuel rods with $B_4C$ rods adds additional unquantified margin to the analysis.

Some fuel assemblies stored at St. Lucie 2 include 6-inch 2.6 weight-percent enriched uranium-235 blankets on the ends of the fuel assemblies. The licensee's calculations were performed both with and without enriched uranium blankets to ensure that the most reactive design was used for each case and, where appropriate, each point on each burnup dependent loading curve. Blankets of different sizes and enrichment were not included in the analysis and therefore are not part of the methodology.

The fuel assembly model included only the active length of the fuel rods. The bounding model did not include fuel assembly grids, nozzles, or the nonfuel ends of the fuel rods. The licensee's sensitivity calculations were performed both with and without soluble boron to show that modeling the grids either reduced reactivity or, when soluble boron was present, caused only a small increase in $k_{eff}$. Not modeling the grids is conservative at unborated conditions, but at some point the amount of soluble boron in the water would make it nonconservative to not model the grids. At 500 ppm of soluble boron, the value used for normal conditions with soluble boron, the impact may be as large as 0.0008 $\Delta k$. It is expected that this value will increase as soluble boron concentrations are increased for accident conditions. However, this increase in $\Delta k$ is offset by the residual uncredited soluble boron.

3.3.3.2 Fuel Assembly Manufacturing Tolerances and Uncertainties

The licensee used an uncommon technique to address fuel storage rack and fuel assembly manufacturing tolerances. In its approach, the licensee used the worst case dimensions for key parameters in the model, effectively incorporating the uncertainty as a bias. It then performed an analysis to show that the margin provided by using the bounding values for a few key parameters was larger than the margin produced by doing detailed uncertainty analysis around the nominal conditions, combining the uncertainty contributions. According to the analysis presented in HI-2104753, including worst case dimensions produces conservative margins varying from 0.0014 to 0.0163 $\Delta k$ compared to using nominal dimensions and including a more conventional detailed uncertainty analysis. Due to the basis for the method and range of margins observed for the St. Lucie 2 analysis, any future licensing basis changes utilizing this method will require the same comparison of detailed uncertainty analysis results to the simplified method results. For the St. Lucie 2 analysis, the method does generate reasonably conservative margins compared to the more conventional detailed uncertainty analysis.

3.3.3.3 Spent Fuel Characterization

Characterization of fresh fuel is based primarily on uranium-235 enrichment and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, as discussed above, or are bounded by values used in the analysis. These tolerances and bounding values would also carry through to the spent nuclear fuel, common industry practice has been to treat the uncertainties as unaffected by the fuel depletion. The characterization of spent nuclear fuel is more problematic. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. That characterization has three main areas: a burnup uncertainty, the axial and radial apportionment of the burnup, and the core operation that achieved that burnup.

3.3.3.4 Burnup Uncertainty

In the Kopp Memo (Reference 9), the NRC staff provided its method for evaluating burnup uncertainty, "A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." The licensee used this approach in HI-2104753 to address the uncertainty in the burned fuel compositions.

3.3.3.5 Axial Apportionment of the Burnup or Axial Burnup Profile

Another important aspect of fuel characterization is the selection of the axial burnup profile. At the beginning of life, a PWR fuel assembly will be exposed to a near-cosine axial-shaped flux, which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the fuel assembly ends, burnup is suppressed due to leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis" (Reference 12), has shown that, at assembly burnups above about 10 to 20 GWd/MTU, this results in an underprediction of $k_{eff}$; generally the underprediction becomes larger as burnup increases. This is what is known as the "end effect." Proper selection of the axial burnup profile is necessary to ensure $k_{eff}$ is not underpredicted due to the end effect.

NUREG/CR-6801 provides recommendations for selecting an appropriate axial burnup profile. With respect to the axial burnup profile, HI-2104753 did not use the axial burnup profiles from NUREG/CR-6801. A description of how the axial burnup profiles were derived is provided in Sections 2.3.5.6 and 7.11 of HI-2104753. The NRC staff requested additional information regarding the derivation of the axial burnup profiles, which the licensee provided in its response to RAI SRXB-139 (Reference 6).

The axial burnup profiles used were derived from 744 St. Lucie 2 fuel assembly burnup profiles. These profiles include 620 pre-EPU profiles and 124 profiles from design calculations for post-EPU cycles. The 744 profiles include 558 profiles for assemblies with enriched uranium blankets and 188 profiles for nonblanketed fuel. Separate profiles were prepared for blanketed and unblanketed fuel. Blanketed and unblanketed fuel assemblies are treated differently because application of the relatively lower burnups associated with axial blanket zones to unblanketed assemblies would be significantly and unnecessarily over-conservative.

The process described in HI-2104753 and the RAI SRXB-139 response results in relative burnup-dependent axial burnup profiles that are more reactive than all of the 744 profiles. One conservative feature of the bounding axial burnup profiles is that they are not renormalized to yield an assembly average relative burnup of 1.0. Instead, use of the profiles conservatively reduces the assembly average burnup by 1.2 to 3.5 percent. The use of these conservative fuel-assembly-design-specific and plant-specific axial burnup profiles is acceptable.

The burnup credit limit curves were derived using polynomial fits that bound the most limiting result obtained using a uniform profile, nonblanket profile and a blanketed fuel profile. Consequently, separate loading curves for blanketed and nonblanketed fuel are not needed.

3.3.3.6 Planar Burnup Distribution

Due to the neutron flux gradients in the reactor core, assemblies can show a radially tilted burnup distribution (i.e., differences in burnup between portions or quadrants of the cross section of the assembly). The HI-2104753 analysis did not consider the effect of planar burnup distribution on reactivity. The impact of radial burnup gradients may be estimated by comparing the distribution of radial burnup tilt information provided in Figure 3-4 of DOE/RW-0496 (Reference 13) with information on the sensitivity of $k_{eff}$ to radial burnup tilt provided in Section 6.1.2 of NUREG/CR-6800 (Reference 14). From DOE/RW-0496, the maximum quadrant deviation from assembly average burnup had been observed to be less than 25 percent at low (burnup < 20 GWd/MTU) assembly average burnups and was observed to decrease with burnup, generally being less than 10 percent at burnups above 20 GWd/MTU. Combining these radial tilt bounding estimates with the $k_{eff}$ sensitivity information provided in NUREG/CR-6800, the NRC staff's review of the radial burnup tilts could raise the $k_{eff}$ value as much as 0.002 $\Delta k$. With the information available, the staff finds that it is conservative to consider this effect as a bias, its potential impact is small, and it is accommodated within the analysis margins.

3.3.3.7 Burnup History/Core Operating Parameters

NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," (Reference 15) provides some discussion on the treatment of depletion analysis parameters that determine how the burnup was achieved. While NUREG/CR-6665 is focused on NCS analysis in storage and transportation casks, the basic principals with respect to the depletion analysis apply generically, since the phenomena occur in the reactor as the fuel is being used. The results have some applicability to St. Lucie 2 NCS analyses. The basic strategy is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field resulting in maximum plutonium-239/241 production. NUREG/CR-6665 discusses six parameters affecting the depletion analysis: fuel temperature, moderator temperature, soluble boron, specific power and operating history, fixed burnable poisons, and integral burnable poisons. While the mechanism for each is different, the effect is similar: Doppler broadening/spectral hardening of the neutron field resulting in increased plutonium-239/241 production. NUREG/CR-6665 provides an estimate of the reactivity worth of these parameters. The largest effect appears to be due to moderator temperature. NUREG/CR-6665 approximates the moderator temperature effect, in an infinite lattice of high burnup fuel, to be 90 pcm/°K. Thus, a 10 °F change in moderator temperature used in the depletion analysis would result in 0.005 $\Delta k_{eff}$. The effects of each core operating parameter typically have a burnup or time dependency.

For fuel and moderator temperatures, NUREG/CR-6665 recommends using the maximum operating temperatures to maximize plutonium-239/241 production. For moderator temperature, the HI-2104753 analysis used the post-EPU exit water temperature for the peak power assembly and used a conservatively high fuel temperature. The moderator and fuel temperatures used are therefore acceptable.

For boron concentration, NUREG/CR-6665 recommends using a conservatively high cycle-average boron concentration. The licensee's analysis used a cycle average soluble boron concentration of 750 ppm for pre-EPU cycles and 1000 ppm for post-EPU cycles. The data for St. Lucie 2 cycles 13 through 18 were used to correctly calculate the average soluble boron concentration for each cycle, yielding a maximum pre-EPU value of 650 ppm, which is well below the 750 ppm value used for pre-EPU fuel burnup calculations. For post-EPU cycles, a value of 1000 ppm was adopted. This value is 300 ppm higher than the predicted boron letdown curve provided in Table F.8 of HI-2104753 and is to bound future operations. The boron concentrations used are therefore acceptable.

Specific power and operating history are related. Operating history is essentially the history of time spent at various specific power levels. Specific power is a second order effect compared to moderator temperature and soluble boron concentration. The HI-2104753 analysis used a pre-EPU specific power of 33.5 MW/MTU and a post-EPU specific power of 37.5 MW/MTU. As is described in Section 7.4 of HI-2104753 and presented in Tables 7.25 through 7.31, sensitivity calculations were performed for ± 5 MW/MTU. These calculations demonstrated that the St. Lucie 2 SFP rack $k_{eff}$ values are only weakly dependent on the specific power level. The uncertainties associated with specific power and operating history are very small compared to the other uncertainties explicitly included in the analysis and therefore have negligible impact on the overall uncertainty. The specific power and operating history used are therefore acceptable.

3.3.3.8 Integral and Fixed Burnable Absorbers

St. Lucie 2 has in the past used fuel assemblies in which some of the fuel rods were replaced with rods filled with $B_4C$. St. Lucie 2 has also used fuel rods in which some of the rods contained fuel plus $Gd_2O_3$. Rather than explicitly modeling the $B_4C$ or $UO_2$+$Gd_2O_3$ fuel rods, the HI-2104753 analysis used models where all fuel rods were assumed to be at maximum planar average enrichment. As discussed in Section 3.3.3.1 of this evaluation sensitivity studies for St. Lucie 2 SFP indicate this is a conservative modeling simplification.

Removable neutron absorbing rods have not been used in the St. Lucie 2 reactor.

3.3.4 Determination of Soluble Boron Requirements

Section 50.68 of 10 CFR requires that the $k_{eff}$ of the St. Lucie 2 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 unborated water. By considering the double contingency principal, the 1900 ppm of soluble boron that is required by the St. Lucie 2 post-EPU TS can be credited to ensure the St. Lucie 2 $k_{eff}$ does not exceed 0.95, provided that the accident and a boron dilution are independent events. Since the minimum required soluble boron has increased from 1720 ppm to 1900 ppm and the soluble boron credited under normal conditions has decreased from 520 to 500 ppm, the pre-EPU soluble boron dilution analysis bounds the post-EPU configuration.

HI-2104753 considered the following accidents: abnormal temperature, dropped, mislocated, and misloaded fuel assemblies, missing or damaged required METAMIC™ insert, missing required control rods, use of an incorrect loading curve, misalignment between the active fuel region and the neutron absorber. The licensee's analysis demonstrated that the fuel storage racks will be subcritical under accident conditions with 1500 ppm of soluble boron, which is below the revised TS required 1900 ppm of soluble boron.

3.3.5 Margin Analysis and Comparison with Remaining Uncertainties

This section provides evaluation of additional conservatism in the analysis and evaluation of items that may have been treated nonconservatively.

3.3.5.1 Potential Nonconservatisms

The response to RAI SRXB-121 (Reference 6) indicates vthat results from some nonconverged MCNP calculations were used in the analysis. Those new calculations improved convergence and the case-specific uncertainty increased by 0.001 $\Delta k$ for some cases. This small increase is covered by the conservative margins described below. The nonconverged cases also affected the determination of the loading curves, increasing the $k_{eff}$ by up to 0.0052 $\Delta k$ for some of the points. Since the curves were generated using a target $k_{eff}$ + bias + uncertainties of 0.99, the $k_{eff}$ of the normal conditions associated with the loading curves may be as high as 0.9952. Considering the conservative margins described below, the NRC staff finds this acceptable.

As was noted in Section 3.3.3.6, the evaluation of the planar burnup distribution could increase the estimated $k_{eff}$ from the St. Lucie 2 NCS analysis by as much as 0.002 $\Delta k$.

As was noted in Section 3.3.2.1, the evaluation of the temperature dependence of the calculated $k_{eff}$ values did not look at water densities and temperature between the normal condition extremes. It is possible that the calculated $k_{eff}$ values may, in some cases, have peaked between the extremes. It is unlikely that the nonconservatism, if it exists, could be as large as 0.0034 $\Delta k$. Considering the conservative margins described below, the NRC staff finds this acceptable.

The analysis incorporates a 2.5-percent uncertainty in the fuel assembly average burnup. The response to RAI SRXB-128 (Reference 5) indicates that 1.5 percent of the 2.5 percent covers uncertainty in plant secondary calorimetric power. This leaves 2-percent uncertainty on the assembly relative power. Uncertainty is introduced because only approximately 25 percent of the assemblies are instrumented and that there are uncertainties associated with neutron flux measurement, detector calibration, integration of power over the life of the assembly, and extrapolation of measured data to uninstrumented assemblies. A value more typically used is 5 percent of burnup. Because there is conservatism in the derivation of the axial burnup profiles, and the other conservatisms described in the next subsection, use of the low 2.5-percent uncertainty in the St. Lucie 2 NCS analysis is, therefore, acceptable in the context of the available margins.

3.3.5.2 Potential Analysis Conservatisms

The analysis includes several aspects that add margin to the analysis.

These include the following:

- Axial burnup profiles not renormalized

  Application of the axial burnup profiles artificially reduces the assembly average burnups by 1.2 to 3.5 percent. This represents additional margin of about 0.003 Δk for an unblanketed assembly burned to 50 GWd/MTU.

- Burnup coefficient loading curve defined at $k_{eff}$ + uncertainties + biases equal to 0.99

  Section 50.68 of 10 CFR requires that, when soluble boron in the spent fuel pool is credited, the $k_{eff}$ must be below 1.0 when no soluble boron is present. Generating loading curves with a $k_{eff}$ of 0.99, without soluble boron, includes 0.01 Δk margin to the analysis.

- Burnup coefficient loading curve fits either match or conservatively bound the calculated burnup coefficient curve points

  The final loading curves are second-order polynomial fits that conservatively bound the calculated acceptable loading points. This introduces a small amount of additional margin for most of the range.

- Conservative treatment of uncertainties

  An unusual treatment of manufacturing tolerances was used that, according to the analysis, introduced additional margin varying from 0.0014 to 0.0163 Δk.

- Modeling all fuel rods at the maximum planar average enrichment rather than explicitly modeling $B_4C$ or $UO_2$+$Gd_2O_3$ fuel rods

  Many of the fuel assemblies include either fuel rods with $UO_2$+$Gd_2O_3$ pellets or with $B_4C$ rods replacing fuel rods. The presence of these Gd rods and $B_4C$ rods is not credited in the analysis. This represents significant unquantified margin. From a generic study presented in Reference 15, not modeling the $Gd_2O_3$ represents margin ranging from about 0.08 Δk at zero burnup to 0.002 Δk at high burnups. From Figure 36 in the Reference 15, not modeling the $B_4C$ rods in a CE 14x14 assembly results in margin ranging from around 0.15 Δk at low burnups to 0.002 Δk at burnups around 30 GWd/MTU.

- An uncertainty equal to [ ] of the minor actinide and fission product worth was used

  Recent work documented in Reference 8 indicates that use of an uncertainty as low as 1.5 percent of the minor actinide and fission product worth may be appropriate. Use of the [ ] uncertainty adds margin to the uncertainty analysis. The "[ ] of the worth" uncertainty results in uncertainties ranging from 0.0054 Δk at low burnups to 0.026 Δk at high burnups. At high burnups, use of the [ ] uncertainty value increased the total uncertainty by about 0.01 Δk.

3.3.5.3 Conclusion on Analysis of Margins

Considering both the potential nonconservatisms identified in Section 3.3.5.1 and the conservatisms identified in Section 3.3.5.2, it is concluded that the available margins offset the potential nonconservatisms.

3.4 The NFV NCS Analysis

During the review of the NCS analysis of the SFP it was determined that NCS analysis supporting fresh fuel in the NFV was needed to support related TS changes. Specifically, the TS limiting the fuel stored to a maximum uranium-235 enrichment of 4.5 weight percent was changed to permit a maximum lattice average uranium-235 enrichment of 4.6 weight-percent uranium-235. A NFV NCS analysis, documented in Holtec International report HI-2094416, Revision 1 (Reference 16), was provided as attachment 4 to Reference 6.

This section documents the review of the NFV NCS analysis.

3.4.1 New Fuel Vault (NFV) NCS Analysis Method

The analysis is to demonstrate compliance with the following requirements from 10 CFR 50.68:

Paragraph 50.68(b)(2) of 10 CFR requires, "The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95-percent probability, 95-percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used."

Paragraph 50.68(b)(3) of 10 CFR requires, "If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95-percent probability, 95-percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used."

Compliance with these requirements is demonstrated in HI-2094416 by modeling a simplified representation of fresh fuel stored in the NFV submitted by the licensee as part of its application. The licensee's analysis demonstrates that the $k_{eff}$ value, including bias and uncertainties, for both full density water and optimum moderation conditions is more than 0.03 Δk below the applicable limits from 10 CFR 50.68.

The MCNP4a computer code and nuclear data was used in their analysis to calculate the $k_{eff}$ value for fresh fuel in the NFV fuel storage racks. MCNP4a has a long history of use for this type of analysis and is, therefore, acceptable.

MCNP4a and its associated nuclear data were validated in Holtec International report HI-2094486R0, "MCNP Benchmark Calculations," and Section 2 of HI-2094416 states the bias and 95/95 uncertainty from the validation are 0.0013 $\Delta k \pm 0.0086$. This bias and bias uncertainty are similar to values reported for other analyses. Considering that the values are consistent with similar analyses and that there is more than 0.03 $\Delta k$ margin to the limits, further review of the validation was deemed unnecessary.

The NRC staff reviewed the computational method and supporting validation and finds them acceptable.

3.4.2 NFV Fuel Storage Racks

The steel structures that comprise the NFV fuel storage racks were conservatively not modeled. Without the steel, the rack model simplifies down to constraints on the spacing and location of the fuel assemblies. All rack structures were modeled as water at the calculation-specific water density.

Section 7 of HI-2094416 states that rack tolerances are not included since the racks are not modeled. Rack tolerances include tolerances on spacing between assemblies and between assemblies and the NFV floor and walls. Not modeling the storage rack steel provides enough margin to cover deficiencies in the uncertainty analysis.

3.4.3 Fuel Assemblies

HI-2094416 states that the design basis fuel assembly is a Westinghouse 16x16 and describes the assembly in Table 5.1 of that report. From Table 5.1, the 16x16 assembly is equivalent to a CE 16x16 fuel assembly. The assembly used in the analysis presented in HI-2094416 is consistent with the fuel assembly design used for the spent fuel storage analysis documented in HI-2104753, review of which was covered in Section 3.3 of this safety evaluation.

The fuel assembly design is conservative in that natural uranium blankets and fuel rods containing $Gd_2O_3$ are modeled as unpoisoned 4.6 weight-percent uranium-235 $UO_2$. The analysis did not address the impact of using the planar average enrichment simplification. As was shown in Table 7.34 of HI-2104753, the impact of using average enrichments rather than distributed enrichments is small and mixed in direction. Any nonconservatism related to this effect can be covered by the large margins to the limits.

Other than the active lengths of the fuel rods, other fuel assembly components were replaced with water. This is conservative for NVF analysis.

Based on the NRC staff's review of the above, the fuel assembly model used in the NFV NCS analysis is adequately conservative.

3.4.4 Analysis of Margins

The licensee determined that the maximum $k_{eff}$ value, including biases and uncertainties, for the full density flooding case was 0.9152. The applicable limit is 0.95. The margin to this limit is 0.035 $\Delta k$. The licensee determined that the maximum $k_{eff}$ value, including biases and uncertainties, for the optimum moderation case was 0.9358. The applicable limit is 0.98. Thus, the margin to this limit is 0.044 $\Delta k$ and the NRC staff finds this acceptable..

A few small potential nonconservatisms were identified in the review. The margins to the limits are large enough to cover potential nonconservatisms associated with deficiencies in the analysis of tolerances and failure to evaluate the impact of some modeling simplifications.

3.5 Conclusions

The NRC staff review of the St. Lucie 2 spent fuel storage racks NCS analysis, documented in HI-2104753, and of the St. Lucie 2 NFV NCS analysis documented in HI-2094416, Revision 1, identified some nonconservative items.

Those items were evaluated against the margin to the regulatory limit and what the NRC considers an appropriate amount of margin attributable to conservatisms documented in the analyses.

It was noted that the results from nonconverged Monte Carlo calculations were used in the analysis. The response to RAI SRXB-121 indicated that upon rerunning many of the cases, the reported maximum $k_{eff}$ values may be nonconservative by as much as 0.005 $\Delta k$. The NRC staff finds this acceptable because the burnup credit loading curves were designed to have a maximum $k_{eff}$, including biases and uncertainties, of 0.99. This left adequate margin to cover the use of results from poorly converged Monte Carlo calculations.

The uncertainty analysis included an allowance of only 2.5 percent of the assembly average burnup used to compare to the burnup credit loading curves. This value is lower than the more typical value used of 5 percent of assembly average burnup.

The effects of $k_{eff}$ variation with SFP water temperature and density were quantified only at the extremes of the parameter ranges. Because the analysis did not include intermediate water densities to show that $k_{eff}$ did not peak between the extremes of the parameter ranges, the nonconservatism associated with this part of the analysis is likely between 0 and 0.0034 $\Delta k$.

The licensee's use of a method for evaluating manufacturing tolerances and uncertainties in its SFP analysis was supported with calculations demonstrating that the revised method yielded conservative total uncertainties with additional margins ranging from 0.0014 to 0.0163 $\Delta k$.

Following review of the supporting analysis reports and based on the margins to regulatory limits, crediting some analysis conservatism, and including consideration of the identified potential nonconservatisms, the NRC staff concludes that there is a reasonable assurance that the St. Lucie 2 SFP and NFV fuel storage racks meet the applicable NCS regulatory requirements.

Because of the need to evaluate offsetting effects in the licensee's analysis, its analysis constitutes a methodology of which any change would be a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

## 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, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

## 5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact was prepared and published in the *Federal Register* on January 6, 2012 (77 FR 813). The draft Environmental Assessment provided a 30-day opportunity for public comment. The NRC staff received comments that were addressed in the final environmental assessment. The final Environmental Assessment was published in the *Federal Register* on July 6, 2012 (77 FR 40092). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

## 6.0 CONCLUSION

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

## 7.0 REFERENCES

1. Florida Power and Light Company letter L-2011-021, R.L. Anderson, Site Vice President, St. Lucie Plant to USNRC document control desk, re: "St. Lucie Plant Unit 2, Docket No. 50-389, Renewed Facility Operating License No. NPF-16, License Amendment Request for Extended Power Uprate," February 25, 2011 (ADAMS ML110730116).

2. Florida Power and Light Company letter L-2011-466, R.L. Anderson, Site Vice President, St. Lucie Plant to USNRC document control desk, re: "St. Lucie Plant Unit 2, Docket No. 50-389, Renewed Facility Operating License No. NPF-16, Revision to Extended Power Uprate License Amendment Request Proposed Technical Specification 5.6, Design Features – Fuel Storage – Criticality," November 4, 2011 (ADAMS ML11314A111).

3. Florida Power and Light Company letter L-2012-181, R.L. Anderson, Site Vice President, St. Lucie Plant to USNRC document control desk, re: "St. Lucie Plant Unit 2, Docket No. 50-389, Renewed Facility Operating License No. NPF-16, Extended Power Uprate License Amendment Request – Supplement to Proposed Technical Specification Changes Related to Spent Fuel Storage Requirements," April 30, 2012 (ADAMS ML12124A065).

4. Florida Power and Light Company letter L-2011-489, R. L. Anderson, Site Vice President, St. Lucie Plant to USNRC document control desk, re: "St. Lucie Plant Unit 2, Docket No. 50-389, Renewed Facility Operating License No. NPF-16, Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment," December 8, 2011 (ADAMS ML11350A245).

5. Florida Power and Light Company letter L-2012-182, R.L. Anderson, Site Vice President, St. Lucie Plant to USNRC document control desk, re: "St. Lucie Plant Unit 2, Docket No. 50-389, Renewed Facility Operating License No. NPF-16, Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request," May 4, 2012 (ADAMS ML12130A478).

6. Florida Power and Light Company letter L-2012-201, R.L. Anderson, Site Vice President, St. Lucie Plant to USNRC document control desk, re: "St. Lucie Plant Unit 2, Docket No. 50-389, Renewed Facility Operating License No. NPF-16, Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request," May 7, 2012 (ADAMS ML12132A414).

7. S. Anton, "Lumped Fission Products and Pm148m Cross Sections for MCNP," HI-2033031, Holtec International, January 2011.

8. J.M. Scaglione, D.E. Mueller, J.C. Wagner, and W.J. Marshall, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality ($k_{eff}$) Predictions," NUREG/CR-7109 (ORNL/TM-2011/514), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, April 2012.

9. NRC Memorandum from L. Kopp to T. Collins, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998 (ADAMS ML003728001).

10. V. Makodym, "Nuclear Group Computer Code Benchmark Calculations," HI-2104790, Holtec International, January 2011 (ADAMS ML11350A245).

11. D.E. Mueller, K.R. Elam, and P.B. Fox, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," NUREG/CR-6979, ORNL/TM-2007/083, U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, September 2008.

12. J.C. Wagner, M.D. DeHart, and C.V. Parks, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis," NUREG/CR-6801 (ORNL/TM-2001/273), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2003.

13. C. Kang, et al., "Horizontal Burnup Gradient Datafile for PWR Assemblies," DOE/RW-0496, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, May 1997.

14. J.C. Wagner and C.E. Sanders, "Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs," NUREG/CR-6800 (ORNL/TM-2002/6), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2003.

15. C.V. Parks, M.D. DeHart, and J.C. Wagner, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665 (ORNL/TM-1999/303), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, February 2000.

16. B. Brickner, "Criticality Analysis of the St. Lucie Unit 2 New Fuel Vault," HI-2094416, Revision 1, Holtec International, June 2010.

Principal Contributor: Kent Wood

Date: September 19, 2012

The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations,* Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Accordingly, the NRC staff has also prepared a redacted, publicly-available, non-proprietary version of the SE. Copies of theproprietary and non-proprietary versions of the SE are enclosed.

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/***

Tracy J. Orf, Project Manager
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 162 to NPF-16
2. Non-Proprietary Safety Evaluation
3. Proprietary Safety Evaluation

cc w/ Enclosures 1 and 2: Distribution via Listserv

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