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

September 19, 2016

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
Florida Power & Light Co.
Mail Stop: NT3/JW
15430 Endeavor Drive
Jupiter, FL 33478

SUBJECT: ST. LUCIE PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATION CHANGE TO ELIMINATE THE MODERATOR TEMPERATURE COEFFICIENT SURVEILLANCE TEST AT THE END OF CYCLE (CAC NOS. MF7269 AND MF7270)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment Nos. 235 and 185 to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2, respectively. These amendments consist of changes to the Technical Specifications (TSs) in response to Florida Power & Light Company's application dated January 19, 2016, as supplemented by a letter dated May 6, 2016.

The amendments revise the TS Surveillance Requirement 4.1.1.4.2 to implement elimination of the near end-of-cycle Moderator Temperature Coefficient measurement as supported by Topical Report CE NPSD-911-A and Amendment 1-A. Topical Report WCAP-16045-P-A for the PARAGON code would also be added to TS 6.9.1.11.b Core Operating Limits Report analytical methods.

The NRC staff's safety evaluation of the amendments is enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Perry H. Buckberg, Senior Project Manager
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50-389

Enclosures:
1. Amendment No. 235 to DPR-67
2. Amendment No. 185 to NPF-16
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 235
Renewed License No. DPR-67

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

A. The application for amendment by Florida Power & Light Company (FPL, the licensee), dated January 19, 2016, as supplemented by a letter dated May 6, 2016, 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. DPR-67 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 235, 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 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeanne A. Dion

Jeanne A. Dion, Acting Chief
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: September 19, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 235

ST. LUCIE PLANT UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace Page 3 of Renewed Facility Operating License DPR-67 with the attached Page 3.

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

Remove Page	Insert Page
3/4 1-6	3/4 1-6
6-19b	6-19b

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 or incorporated below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 3020 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 235 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

Appendix B, the Environmental Protection Plan (Non-Radiological), contains environmental conditions of the renewed license. If significant detrimental effects or evidence of irreversible damage are detected by the monitoring programs required by Appendix B of this license, FPL will provide the Commission with an analysis of the problem and plan of action to be taken subject to Commission approval to eliminate or significantly reduce the detrimental effects or damage.

C. Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 28, 2003, describes certain future activities to be completed before the period of extended operation. FPL shall complete these activities no later than March 1, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on March 28, 2003, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71 (e)(4), following issuance of this renewed license. Until that update is complete, FPL may make changes to the programs described in such supplement without prior Commission approval, provided that FPL evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Sustained Core Uncovery Actions

Procedural guidance shall be in place to instruct operators to implement actions that are designed to mitigate a small-break loss-of-coolant accident prior to a calculated time of sustained core uncovery.

SURVEILLANCE REQUIREMENTS (continued)

4.1.1.4.2*** Verify MTC is within the lower limit specified in the COLR.****

Each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup.

*** If MTC is more negative than the lower limit specified in the COLR when extrapolated to the end of cycle, 4.1.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.

**** Only required if the MTC determined in SR 4.1.1.4.1 is not within ±1.6 pcm/°F of the corresponding design value.

CORE OPERATING LIMITS REPORT (continued)

20. EMF-1961 (P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors."

21. EMF-231O(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Revision 1, as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0.

22. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," Revision 0, as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0.

23. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 0, as supplemented by ANP-2903(P), "St. Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA summary Report with Zr-4 Fuel Cladding," Revision 1.

24. BAW-10240(P)(A) Revision 0, "Incorporation of M5 Properties in Framatome ANP Approved Methods."

25. WCAP-16045-P-A, Revision 0, “Qualification of the Two-Dimensionall Transport Code PARAGON,” August 2004.

UNITED STATES
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FLORIDA POWER AND 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. 185
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 (FPL, the licensee), dated January 19, 2016, as supplemented by a letter dated May 6, 2016, 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 185, 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 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeanne A. Dion

Jeanne A. Dion, Acting Chief
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: September 19, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 185

ST. LUCIE PLANT UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace Page 3 of Renewed Facility Operating License NPF-16 with the attached Page 3.

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

Remove Page	Insert Page
3/4 1-5	3/4 1-5
6-20e	6-20e
6-20ea	6-20ea

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 3020 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 185 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC} shall be maintained within the limits specified in the COLR. The maximum upper limit shall be +5 pcm/°F at ≤70% of RATED THERMAL POWER, with a linear ramp from +5 pcm/°F at 70% of RATED THERMAL POWER to 0 pcm/°F at 100% RATED THERMAL POWER.

APPLICABILITY: MODES 1 AND 2*#.

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 Verify MTC is within the upper limit specified in LCO 3.1.1.4.

a. Prior to entering MODE 1 after each fuel loading, and

b. Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup. **

4.1.1.4.2*** Verify MTC is within the lower limit specified in the COLR.****

Each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup.

See Special Test Exception 3.10.2 and 3.10.5.

* With K_{eff} greater than or equal to 1.0.

** Only required to be performed when MTC determined prior to entering MODE 1 is verified using adjusted predicted MTC.

*** If MTC is more negative than the lower limit specified in the COLR when extrapolated to the end of cycle, 4.1.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.

**** Only required if the MTC determined in SR 4.1.1.4.1 is not within ±1.6 pcm/°F of the corresponding design value.

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

61. WCAP-11397-P-A, (Proprietary), 'Revised Thermal Design Procedure," April 1989.

62. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

63. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

64. Letter, W. Jefferson Jr. (FPL) to Document Control Desk (USNRC), "St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit," L-2003-276, December 2003 (NRC SER dated January 31, 2005, Letter B.T. Moroney (NRC) to J.A. Stall (FPL), TAC No. MC1566).

65. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses", April 1999.

66. WCAP-7908-A, Rev. 0, "FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod", December 1989.

67. WCAP-7979-P-A, Rev. 0, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code", January 1975.

68. WCAP-7588, Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods", January 1975.

69. WCAP-16045-P-A, Revision 0, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

70. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs, Volume 1 Methodology Description, Volume 2 Benchmarking Results," Siemens Power Corporation, January 1997.

71. XN-NF-78-44 (NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, Inc.," October 1983.

72. XN-75-27(A) and Supplements 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P).

73. XN-NF-82-06 (P)(A), Rev. 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, Inc., October 1986.

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

74. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, Inc., November 1986.

75. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, December 1991.

76. EMF-92-116(P)(A), Rev. 0, "Generic Mechanical Design Criteria for PWR Fuel Design," Siemens Power Corporation, February, 1999.

77. BAW-10240(P)(A), Rev.0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," Framatome ANP, Inc., May 2004.

78. XN-NF-82-21(P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, September 1983.

79. EMF-92-153(P)(A), Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," January 2005.

80. EMF-1961(P)(A), Revision 0, "Statistical/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000.

81. EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, Inc., May 2004.

82. XN-75-32(P)(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," October 1983.

83. BAW-10231P-A Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004.

84. EMF-2103(P)(A) Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

85. EMF-2328 (P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle on the NRC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 235 AND 185

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-67 AND NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By application dated January 19, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16033A472), as supplemented by letter dated May 6, 2016 (ADAMS ML16133A045), Florida Power and Light Company (the licensee) submitted a license amendment request (LAR) for the St. Lucie Plant Unit Nos. 1 and 2 (St. Lucie 1 & 2). The amendment would revise the Technical Specification (TS) Surveillance Requirement (SR) 4.1.1.4.2 to allow the near end-of-cycle (EOC) Moderator Temperature Coefficient (MTC) measurement to not be performed if the conditions in Combustion Engineering, Inc., (CE) Topical Report (TR) CE NPSD-911-A and Amendment 1-A, "Analysis of Moderator Temperature Coefficients in Support of a Change in the Technical Specifications End-of-Cycle Negative MTC Limits" (Reference 1) are met. The licensee also requests adding Westinghouse TR WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON" (Reference 2) to the Core Operating Limits Report (COLR) methods in TS 6.9.1.11.b. This will support future use of PARAGON in licensing applications for St. Lucie 1 & 2.

The supplemental letter dated May 6, 2016, was in response to a request for additional information (RAI) issued by U.S. Nuclear Regulatory Commission (NRC) staff on March 28, 2016. The additional information clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 29, 2016 (81 FR 17501).

2.0 REGULATORY EVALUATION

2.1 Description of Moderator Temperature Coefficient Requirements

The moderator in a nuclear reactor is the water flowing through the core. As any operating condition of the moderator or fuel changes, the reactivity of the core changes accordingly. Reactivity is a function of the fission neutron population and describes the state of the reactor core. A nuclear reactor is critical (i.e., no change in neutron population from one generation to the next) when reactivity of the core is zero, subcritical when reactivity is negative, and supercritical when reactivity is positive. Changing a reactor core operating parameter affects other properties of the core. Once a change has been made to the core, it is necessary to make some compensating change to maintain criticality at the same power.

The properties of a reactor system that result in positive or negative reactivity additions with changes in certain parameters are generally described by reactivity coefficients. A reactivity coefficient is defined as the change of reactivity per unit change in some operating parameter of the reactor. The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response to abnormal or accidental transients, is evaluated by means of detailed plant calculations. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables that are set by conditions external to the core. Because reactivity coefficients change during the life of the reactor core, a range of coefficient values are established to ensure the correct response of the plant throughout its life. The MTC is one of these reactivity coefficients and is one of the controlling parameters for power and reactivity changes. MTC is defined as the change in reactivity in percent millirho (pcm) per degree Fahrenheit (°F) change in moderator temperature (pcm/°F).

2.2 Licensee's Proposed Changes

The licensee proposed to revise TS SR 4.1.1.4.2 to eliminate the requirement for the MTC measurement at 2/3 of expected core burnup if the measured beginning-of-cycle (BOC) MTC is within +/-1.6 pcm/°F of the predicted value. The following note is proposed to be added to SR 4.1.1.4.2:

> "Only required if the MTC determined in SR 4.1.1.4.1 is not within +/-1.6 pcm/°F of the corresponding design value."

The licensee also proposed to revise TS 6.9.1.11.b to add the following TR to the COLR list of references:

> WCAP-16045-P-A, Revision 0, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

2.3 Regulatory Review

Per Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(b), each license authorizing operation of a utilization facility will include TSs, which will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34. The Commission may include such additional TSs as the Commission finds appropriate.

Per 10 CFR 50.90, whenever a holder of a license desires to amend the license, application for an amendment must be filed with the Commission fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. Concerning TSs, 10 CFR 50.36(a)(1) states that each applicant for a license authorizing operation of a utilization facility shall include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TSs

Section 50.36(c) requires, in part, that TSs will include items in the category of SRs. Per 10 CFR 50.36(c)(3) "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." As stated in 10 CFR 50.36(c)(2)(i), limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

Section 50.36(c) requires, in part, that TSs will include items in the category of Administrative Controls. Per 10 CFR 50.36(c)(5), "Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

As stated in 10 CFR 50.92, in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The considerations include 10 CFR 50.40 and 10 CFR 50.57, require the Commission to find, among other things, that (1) there is reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (2) there is reasonable assurance that such activities will be conducted in compliance with the regulations in this chapter; and (3) the issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

The St. Lucie Unit No. 1 (St. Lucie 1) design is based on the proposed General Design Criteria (GDC) published by the Atomic Energy Commission in the *Federal Register* (32 FR 10213) on July 11, 1967. Section 1.3.2, "Comparison of Preliminary and Final Design," and Chapter 3, "Design Criteria - Structures, Components, Equipment and Systems," of the St. Lucie 1 Updated Final Safety Analysis Report (UFSAR) describe the St. Lucie 1 design. Section 3.1.11 "CRITERION 11 - REACTOR INHERENT PROTECTION" of the Unit 1 UFSAR states: "The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity." Section 3.1.11 further states: "In the power operating range, the combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor power in the power operating range is a decrease in reactivity; i.e., the inherent nuclear feedback characteristics is negative."

St. Lucie 2 was designed and constructed in compliance with the GDC, relevant to the evaluation of this LAR. Section 3.1.11, "CRITERION 11 - REACTOR INHERENT PROTECTION" of the Unit 2 UFSAR states: "The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity." Section 3.1.11 further states: "In the power operating range, the combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor power in the power operating range is a decrease in reactivity; i.e., the inherent nuclear feedback characteristics are not positive."

NRC Generic Letter (GL) 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions," dated June 24, 1999 (ADAMS ML031080345), provides guidance on licensee qualification for using safety analysis codes in performing their own licensing analyses using methods that have been reviewed and approved by the NRC.

NRC GL 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988 (ADAMS ML031130447) provides guidance on relocating cycle-specific parameter limits from the TSs to the COLR. The COLR controls the values of cycle-specific parameters and methodologies.

By letter dated May 17, 1988 (ADAMS 8805250188 (non-public)), the NRC staff approved the Westinghouse TR WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" (Reference 3). PHOENIX-P is a two-dimensional, multi-group transport theory code used to calculate lattice physics parameters and provide cross-sections as input to the ANC code.

By letter dated June 14, 2000 (within ADAMS ML003752592), the NRC staff approved the CE Owners Group TR CE NPSD-911-A. CE NPSD-911-A provides a methodology for licensees operating CE plants to be exempted from performing the near-EOC MTC measurement, provided the measured BOC MTCs are within +/- 1.6 percent millirho (pcm) per degree Fahrenheit (°F) change in moderator temperature (pcm/°F) ($0.16*10^{-4}\Delta\rho/^{o}F$) of the predicted value. The conclusion of the NRC's Safety Evaluation (SE) included five conditions on the use of the methodology.

On March 18, 2004, the NRC staff approved the Westinghouse TR WCAP-16045-P-A (ADAMS ML040780402). PARAGON is a lattice physics code and is a replacement for PHOENIX-P. The conclusion to the staff's SE for WCAP-16045-P-A explicitly endorsed this replacement, stating, "... the staff considers the new PARAGON code to be well qualified as a stand-alone code replacement for the PHOENIX-P lattice code, wherever the PHOENIX-P code is used in NRC-approved methodologies. The staff concludes that it is acceptable for licensing applications."

3.0 TECHNICAL EVALUATION

3.1 Evaluation of TS Changes

Topical reports CE NPSD-911-A and WCAP-16045-P-A have been reviewed and approved by the NRC staff. Therefore, the focus of this review will be verification that the methodologies, conditions, and limitations of the generic approvals are satisfied for the St. Lucie 1 & 2 specific application.

3.2 Application of CE NPSD-911-A and Amendment 1-A

The licensee proposed implementation of the methodology described in CE NPSD-911-A and Amendment 1-A, which was approved by the NRC with the following conditions:

1. In order to ensure that the moderator temperature coefficient will not exceed the Technical Specification limit with a confidence/tolerance of 95/95 percent, the cycle must be designed, using the CE methodology, such that the best estimate MTC is:

    a. more negative than the BOC Technical Specification limit by the design margin

    b. more positive than the EOC Technical Specification limit by the design margin

2. The design margin is determined to be 1.6 pcm/°F at all times in life.

3. The analysis of the revised data base, including the most recent measured and calculated MTCs, has established that if the measured beginning-of-cycle moderator temperature coefficients are within 1.6 pcm/°F of the best estimate prediction, then it can be assumed that the end-of-cycle coefficient will also be within 1.6 pcm/°F of the prediction and its measurement is not required.

4. The measured data reduction must be based on the current CE methodology as described in the report.

5. If the beginning-of-cycle measurements fail the acceptance criteria of ±1.6 pcm/°F and the discrepancy cannot be resolved, the end-of cycle surveillance test must be performed.

In the LAR, the licensee provided dispositions to these conditions for which they concluded that the conditions will be met. Compliance with each of these conditions is discussed below.

3.2.1 Condition 1

Condition 1 of CE NPSD-911-A requires the MTC design margin must be based on the use of the CE methodology (DIT and ROCS codes) for the design calculations for MTC. The licensee stated that St. Lucie 1 & 2 nuclear design calculations are currently performed with the Westinghouse PHOENIX-P/ANC codes for St. Lucie 1 & 2. In the future the licensee may transition to using PARAGON/ANC for St. Lucie 1 & 2, but there is no planned date or cycle to transition from PHOENIX-P to PARAGON.

To support use of PHOENIX-P/ANC with CE NPSD-911-A, the licensee provided benchmarking results that were performed by Westinghouse for CE Nuclear Steam Supply System plants. Westinghouse also performed benchmarking between DIT/ROCS and PARAGON/ANC specifically to support the use of PARAGON/ANC with CE NPSD-911-A (ADAMS ML051740481 and ML051740484 (non-public)). Additionally, the staff requested the licensee to provide any bias used with PHOENIX-P/ANC and PARAGON/ANC to calculate the best estimate design MTC values for NPSD-911-A. In its letter dated May 6, 2016, the licensee responded to the staff's RAI by providing the bias used with PHOENIX-P/ANC and PARAGON/ANC to calculate the St. Lucie 1 & 2 best estimate design MTC values for CE NPSD-911-A, and that used with the MTC data in Tables 1 and 2 of the LAR.

The staff determined that an appropriate bias is used for the predicted best estimate MTC with PHOENIX-P/ANC and PARAGON/ANC, since it is consistent with the CE NPSD-911-A methodology. From the review, the staff determined that all three nuclear design code systems show similar MTC benchmarking results and support the MTC design margin of 1.6 pcm/°F as specified in CE NPSD-911-A. Therefore the staff concludes this condition is satisfied with the use of either PHOENIX-P/ANC or PARAGON/ANC for the St. Lucie 1 & 2 MTC design calculations.

3.2.2 Condition 2

As discussed in Section 4.2.1, the design margin of 1.6 pcm/°F is supported for all three Westinghouse nuclear design code systems. Thus the staff determined that Condition 2 is satisfied since the licensee will use 1.6 pcm/°F as the St. Lucie 1 & 2 design margin with either PHOENIX-P/ANC or PARAGON/ANC.

3.2.3 Condition 3

The licensee provided comparison of measured and predicted MTC data for the past three cycles for both St. Lucie 1 & 2 in Tables 1 and 2 of the LAR. In all cycles, the BOC and EOC measured-to-predicted differences are less than 1.6 pcm/°F. The staff determined that the BOC and EOC data provide additional confidence that 1.6 pcm/°F MTC design margin developed in CE NPSD-911-A remains applicable to St. Lucie 1 & 2 when using the PHOENIX-P/ANC codes.

The proposed changes to TS SR 4.1.1.4.2 specifically require that MTC measurement at 2/3 of expected burnup is only required if the measured MTC at BOC is not within +/-1.6 pcm/°F of the corresponding design value for MTC.

The staff concludes that Condition 3 is satisfied for St. Lucie 1 & 2 since the BOC measured-to-predicted differences are within 1.6 pcm/°F.

3.2.4 Condition 4

As discussed in Section 4.2.1 of this SE, use of either PHOENIX-P/ANC or PARAGON/ANC is a suitable replacement for the CE methodology (DIT/ROCS codes) originally used in CE NPSD-911-A. The staff determined that code benchmarking of MTC data shows that all three codes support the design margin of 1.6 pcm/°F, therefore Condition 4 is satisfied for St. Lucie 1 & 2.

3.2.5 Condition 5

The proposed changes to TS SR 4.1.1.4.2 specifically state that the MTC measurement at 2/3 of expected burnup is only required if the measured MTCs at BOC, as required in TS SR 4.1.1.4.1, are not within +/-1.6 pcm/°F of the corresponding design value for MTC. Failure to meet the +/-1.6 pcm/°F acceptance criteria for the BOC MTC will require performance of the MTC surveillance at 2/3 of expected core burnup. The staff concludes that Condition 5 is satisfied for St. Lucie 1 & 2 since the proposed TS SR changes require 2/3 of expected burnup surveillance tests if the BOC measurements are not within +/-1.6 pcm/°F.

3.3 Use of CE NPSD-911-A with Startup Test Activity Reduction (STAR)

The NRC staff approved the Westinghouse TR WCAP-16011-P-A, "Startup Test Activity Reduction Program" (Reference 4) by letter dated January 14, 2005 (ADAMS ML050180327).

STAR allows one of the BOC MTC measurements (either at hot zero power or hot full power (HFP)) required by TS SR 4.1.1.4.1 not to be performed if certain criteria are met. Based on the data provided in LAR Tables 1 and 2, neither St. Lucie 1 nor 2 has performed the HFP, BOC, MTC measurement in recent cycles when STAR is used.

The methodology in CE NPSD-911-A assumes BOC MTC measurements at both zero power and full power will be performed each cycle. The TR specifically states, "If the isothermal temperature coefficients [ITC] measured at zero power during the cycle startup program, and at power during the first power ascension, fall within the design margin (acceptance criteria) of $\pm 0.16*10^{-4}\Delta\rho/^{\circ}F$, then the end-of-cycle best estimate prediction will also be within $\pm 0.16*10^{-4}\Delta\rho/^{\circ}F$ of the true MTC."

The requirements in CE NPSD-911-A are still met if only one BOC ITC/MTC measurement is performed when using STAR. Pages 3 and 4 of the January 14, 2005, SE for STAR include, "The elimination of the middle-of-cycle (MOC) at power ITC measurement to verify end-of-cycle (EOC) MTC compliance with the technical specifications is acceptable for plants that eliminated this measurement using the methodology in Amendment 1 of TR CE NPSD-911-P-A." This is also supported by precedent with Palo Verde in which both STAR and CE NPSD-911-P-A were implemented in the same license amendment (ADAMS ML15070A124).

3.4 Changes to SR 4.1.1.4.2

Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. SR 4.1.1.4.2 at St. Lucie 1 & 2 provides for testing of the MTC lower limit specified in the COLR.

The licensee modified SR 4.1.1.4.2 based on CE NPSD-911-P-A and technical specification task force-406-T (TSTF-406-T) Revision 2. TSTF-406-T is a Technical Specification Task Force Traveler that revised NUREG-1432, Revision 2 (Standard TSs for CE Plants) to incorporate the allowances approved in CE NPSD-911-P-A. The staff reviewed the changes to ensure that the SR would continue to provide reasonable assurance that the value of the MTC remains within the limiting conditions assumed in the UFSAR accident analyses.

As discussed in the previous section, all conditions of CE NPSD-911-P-A were found by the NRC staff to be satisfied for St. Lucie 1 & 2. The NRC staff, therefore, determined that the use of CE NPSD-911-P-A and TSTF-406-T Revision 2 are acceptable as the basis for modifying SR 4.1.1.4.2 and concluded that the limitations on MTC contained in TS 3.1.1.4 and verified by SR 4.1.1.4 will provide reasonable assurance that the value of the coefficient remains within the limiting conditions assumed in the UFSAR accident analyses.

3.5 Changes to TS 6.9.1.11.b

The licensee proposed adding TR WCAP-16045-P-A, Revision 0 to the TS 6.9.1.11.b COLR list of references. The licensee currently uses PHOENIX-P/ANC for St. Lucie 1 & 2 nuclear design licensing applications and may transition to PARAGON/ANC in the future. The change to the COLR list of references in TS 6.9.1.11.b will support future use of PARAGON for St. Lucie 1 & 2 nuclear design and licensing calculations. As stated in Section 2.0 of this evaluation, WCAP-6045-P-A was approved by the NRC (ADAMS ML040780402) and the SE concluded that the PARAGON code can be used as a replacement for the PHOENIX-P lattice code wherever the PHOENIX-P code is used with NRC approved methodologies. PARAGON has also been benchmarked and approved for use in another CE analog plant, Calvert Cliffs Units 1 and 2 (ADAMS ML050550010).

Identification of the physics code in the COLR consistent with GL 88-16 will identify which code (either PHOENIX-P or PARAGON) is being used for St. Lucie 1 & 2 licensing applications each cycle.

The Administrative Controls section of the TSs, in part, contains the provisions relating to the reporting necessary to assure operation of the facility in a safe manner. TS 6.9.1.11.b is found in the Administrative Controls section of the TSs and it establishes what parameters are to be reported in the COLR and requires a list of the analytical methods used to establish those parameters. Given the generic approval of WCAP-16045-P-A and the use of PARAGON at other CE plants, the NRC staff finds the addition of WCAP-16045-P-A to TS 6.9.1.11.b to the list of the analytical methods to be acceptable for St. Lucie 1 & 2 since its addition will continue to support the reporting requirements necessary to assure operation of the facility in a safe manner. The use of PARAGON by the licensee for St. Lucie 1 & 2 licensing applications remains subject to the guidelines in GL-83-11 Supplement 1, "Licensee Qualification for Performing Safety Analyses."

3.6 SUMMARY

The NRC staff reviewed the information provided by the licensee and determined that CE NPSD-911-A and Amendment 1-A is applicable for St. Lucie 1 & 2 with the following physics code systems:

1. WCAP-11596-P-A, "Qualification of the PHOENIX P/ANC Nuclear Design System for Pressurized Water Reactor Cores"
2. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON"

Use of the PARAGON code for St. Lucie 1 & 2 licensing applications is subject to the technical positions and requests in GL 83-11 Supplement 1 "Licensee Qualification for Performing Safety Analyses."

The NRC staff concludes that the proposed changes to SR 4.1.1.4.2 are acceptable based on the applicability of the CE NPSD-911-A methodology and the licensee meeting all conditions specified in the NRC's safety evaluation for CE NPSD-911-A. Thus the staff finds that 10 CFR 50.36(c)(3) is satisfied since the proposed change to the SR will continue to provide adequate testing of the MTC lower limit.

Because the NRC staff determined that use of CE NPSD-911-A was acceptable at St. Lucie 1 & 2, and that the COLR will identify which physics codes from TS 6.9.1.11.b "Core Operation Limits Report (COLR)" are used each cycle consistent with GL 88-16, the NRC staff finds that the proposed changes to TS 6.9.1.11.b are acceptable. Thus, the staff finds that 10 CFR 50.36(c)(5) is satisfied since that addition of WCAP-16045-P-A to TS 6.9.1.11.b to the list of the analytical methods will continue to support the reporting requirements necessary to assure operation of the facility in a safe manner.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, on June 14, 2016, the NRC staff notified the State of Florida official (Ms. Cynthia Becker, M.P.H., Chief of the Bureau of Radiation Control, Florida Department of Health) of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change inspection or surveillance requirements or requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff determined that the amendments involve no significant change in the types, or significant increase in, the amounts of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. By *Federal Register* notice dated March 29, 2016 (81 FR 17506), the Commission previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on these findings. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. CE NPSD-911-A and Amendment 1-A, "Analysis of Moderator Temperature Coefficients in Support of a Change in the Technical Specifications End-of-Cycle Negative MTC Limits," CE Nuclear Power, September 2000 (ADAMS ML003752592).

2. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," Westinghouse Electric Company, August 2004 (ADAMS ML042250322 non-proprietary and ML042250345 proprietary).

3. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (ADAMS ML080630391 proprietary)

4. WCAP-16011-P-A, "Startup Test Activity Reduction Program," Westinghouse Electric Company, February 2005 (ADAMS ML050660118 non-proprietary and ML13329A701 proprietary).

Principal Contributors: Nicholas Domenico
Joshua Borromeo
Perry Buckberg

Date: September 19, 2016

The NRC staff's safety evaluation of the amendments is enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Perry H. Buckberg, Senior Project Manager
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50-389

Enclosures:
1. Amendment No. 235 to DPR-67
2. Amendment No. 185 to NPF-16
3. Safety Evaluation

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