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

April 23, 2019

Mr. Mano Nazar
President, Nuclear Division
and Chief Nuclear Officer
Florida Power & Light Company
Mail Stop EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING TECHNICAL SPECIFICATION CHANGES TO REDUCE THE NUMBER OF CONTROL ELEMENT ASSEMBLIES (EPID L-2018-LLA-0181)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 198 to Renewed Facility Operating License No. NPF-16 for St. Lucie Plant, Unit No. 2. The amendment changes the Technical Specifications (TSs) in response to the application from Florida Power & Light Company dated June 29, 2018, as supplemented by letters dated August 17, 2018, November 15, 2018, and February 22, 2019.

The amendment revises the TSs to reduce the total number of control element assemblies from 91 to 87. The NRC staff's safety evaluation of the amendment is enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Michael J. Wentzel, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 198 to NPF-16
2. Safety Evaluation

cc: 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. 198
Renewed License No. NPF-16

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

    A. The application for amendment by Florida Power & Light Company dated June 29, 2018, as supplemented by letters dated August 17, 2018, November 15, 2018, and February 22, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended 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 Atomic Energy Act of 1954, as amended, 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. 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. 198, 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 prior to startup from the spring 2020 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: April 23, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 198

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

Replace page 3 of Renewed Facility Operating License No. NPF-16 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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

Remove	Insert
1-2	1-2
5-3	5-3

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. <u>Maximum Power Level</u>

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

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 198, are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT VESSEL INTEGRITY

1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:

a. All containment vessel penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open on an intermittent basis under administrative control.

b. All containment vessel equipment hatches are closed and sealed,

c. Each containment vessel air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be the seal water flow supplied from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include evolutions performed with the upper guide structure (UGS) in place such as control element assembly (CEA) latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

1.9a THE COLR is the unit-specific document that provides cycle specific parameter limits for the current operating reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these limits is addressed in individual Specifications.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO™ or M5® clad fuel rods and/or poison rods, with fuel rods having an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 87 full-length control element assemblies and no part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 198

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By a license amendment request (LAR) dated June 29, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18180A094), as supplemented by letters dated August 17, 2018, November 15, 2018, and February 22, 2019 (ADAMS Accession Nos. ML18229A050, ML18319A043, and ML19053A439, respectively), Florida Power & Light Company (the licensee) proposed changes to the Technical Specifications (TSs) for St. Lucie Plant, Unit No. 2 (St. Lucie 2), which are contained in Appendix A of Renewed Facility Operating License No. NPF-16. The licensee proposed to reduce the total number of control element assemblies (CEAs) specified in the TS, from 91 to 87, to support the permanent removal of four 4-element (mini-dual) CEAs from the reactor core. The licensee also proposed to delete a reference to the 4-element CEAs in an existing TS definition.

The supplements dated November 15, 2018, and February 22, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 9, 2018 (83 FR 50696).

2.0 REGULATORY EVALUATION

2.1. System Description

The St. Lucie 2 reactor core is composed of 217 fuel assemblies and 91 CEAs. The CEAs provide reactivity control and are moved in groups to satisfy the requirements of shutdown, power level changes, and operational maneuvering. The CEAs are classified into two banks: regulating bank and shutdown bank. The CEAs in the regulating bank are used to compensate for changes in reactivity associated with routine power level changes. In addition, regulating CEAs may be used to compensate for minor variations in moderator temperature and boron concentrations during operation at power. Regulating CEAs can be used to help control the core power distribution. This function includes the suppression of xenon induced axial power oscillations. CEAs in the shutdown bank are inserted after the regulating CEAs are inserted and are withdrawn before the regulating CEAs are withdrawn.

Of the 91 CEAs installed in the St. Lucie 2 reactor core, 87 are 5-element CEAs and four are 4-element CEAs. The four 4-element CEAs are a part of the 22 CEAs in Shutdown Bank A, which remains in the fully withdrawn position during power operation. The 4-element CEAs were originally intended in the early core designs to provide additional shutdown margin during a steam line break accident.

The 4-element CEAs are unique in two aspects. First, they insert into two adjacent fuel bundles (versus one fuel bundle for the 5-element CEAs). Second, in order to refuel one bundle at a time, the 4-element CEAs are not stored in the fuel bundle during refueling operations. The 4-element CEAs are typically raised into the upper guide structure, and their extension shafts are pinned to the upper guide structure lift rig floor plate. The licensee stated that this design feature is problematic during refueling operations and during the CEA's replacement, because it is time consuming and also increases the radiological dose received by the crews during the evolution.

2.2 Licensee's Proposed Changes

The licensee proposes to reduce the total number of CEAs specified in the TS 5.3.2, from 91 to 87, to facilitate the permanent removal of four 4-element CEAs during the spring 2020 refueling outage. Additionally, the licensee proposes to delete a reference to the 4-element CEAs from the definition of Core Alterations in TS 1.9. The licensee's proposed changes to TS 1.9 and TS 5.3.2 are show below:

Current TS 1.9, "Core Alteration," of the Definitions, states:

> CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include shared (4-fingered) control element assemblies (CEAs) withdrawn into the upper guide structure (UGS) or evolutions performed with the UGS in place such as CEA latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

Revised TS 1.9, "Core Alteration", of the Definitions, states:

> CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include evolutions performed with the upper guide structure (UGS) in place such as control element assembly (CEA) latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

Current TS 5.3.2 states:

> The reactor core shall contain 91 full-length control element assemblies and no part-length control element assemblies.

Revised TS 5.3.2 states:

> The reactor core shall contain 87 full-length control element assemblies and no part-length control element assemblies.

2.3 Regulatory Review

The NRC staff reviewed the licensee's application to determine whether (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that the activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public. The NRC staff considered the following regulatory requirements, guidance, and licensing and design-basis information during its review of the proposed changes.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Paragraph 50.36(a)(1) states, in part, that each applicant for an operating license shall include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36, "Technical specifications."

Paragraph 50.36(c) of 10 CFR requires that the TSs include items in the following categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," to 10 CFR Part 50 establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The following GDC are applicable for this review:

- GDC 10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- GDC 15, "Reactor coolant system design," requires the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- GDC 20, "Protection system functions," requires that the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, and (2) to sense accident conditions and initiate the operation of systems and components important to safety.

- GDC 25, "Protection system requirements for reactivity control malfunctions," requires that the protection system shall be designed to assure that the specified acceptable fuel design limits (SAFDLs) are not exceeded for any single malfunction of the reactivity control systems.

- GDC 26, "Reactivity control system redundancy and capability," requires that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core sub critical under cold conditions.

- GDC 27, "Combined reactivity control systems capability," requires that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that the capability to cool the core is maintained.

- GDC 28, "Reactivity limits," requires that the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core.

3.0 TECHNICAL EVALUATION

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

3.1 NRC Staff Evaluation

As described in the LAR, the reason for the licensee to propose the TS change is to support permanent removal of four mini-dual CEAs, which are not required for shutdown margin considerations. The licensee identified the design bases and safety analyses that are potentially affected by the proposed TS change. The staff reviewed the LAR and other license documents and finds that the licensee has identified the appropriate potentially affected design bases and safety analyses. The affected areas include maximum controlled reactivity insertion rate, shutdown margin, rod worth assumed in the steam line break analysis and core bypass flow.

3.1.1 TS Change

The staff finds that the change from 91 to 87 CEAs has been made consistently in the TS DEFINITION and DESIGN FEATURES to support the proposed removal of the four 4-element

CEAs. The staff thus concludes that, with these modifications, the TS continue to meet the 10 CFR 50.36 requirement to specify design features and administrative controls.

3.1.2 Maximum Controlled Reactivity Insertion Rate

The NRC staff reviewed the licensee's evaluation for maximum controlled reactivity insertion rate based on GDC 25, GDC 26 and GDC 28. St. Lucie 2's current design bases to satisfy these design criteria are to design the Core, CEAs, Reactor Regulating System, and boron charging portion of the Chemical and Volume Control System so that the potential amount and rate of reactivity insertion due to normal operation and postulated reactivity accidents do not result in:

(a) Violation of the SAFDLs
(b) Damage to the reactor coolant pressure boundary
(c) Disruption of the core or other reactor internals sufficient to impair the effectiveness of emergency core cooling.

The licensee stated in its application that the reactivity accident event analyses depend on neutronic parameter limits that would remain unchanged. These parameter limits would continue to be verified each reload cycle to remain bounding for the cycle-specific core design. Further, the licensee stated that the number of CEAs is not an input to any of the reload safety analyses. Rather, these analyses evaluate various other parameters, including trip reactivity and other parameters related to the CEAs, such as:

(1) Reactivity insertion rate used in CEA withdrawal events,
(2) Ejected rod parameters applicable to rod ejection event,
(3) Dropped/misaligned CEA worth applicable to CEA misoperation event.

The NRC staff finds that the proposed change of removing four 4-element CEAs was acceptable because the reactivity insertion rate and other neutronic parameters would remain unchanged and would continue to be verified so that the design bases comply with GDC 25, GDC 26 and GDC 28.

3.1.3 Shutdown Margin

The NRC staff reviewed the licensee's evaluation for the impact of the removal of four 4-element CEAs on the shutdown margin with respect to GDC 26 and GDC 27. St. Lucie 2's current design bases that satisfy these design criteria are to ensure that the amount of reactivity available from the insertion of withdrawn CEAs under all power-operating conditions will provide for at least 1-percent shutdown margin after cooldown to hot, zero power, and any additional shutdown reactivity requirements assumed in the safety analyses. This includes an event where the highest-worth CEA fails to insert.

To evaluate the impact of the removal of the four 4-element CEAs on the total worth of reactivity systems and the subsequent available shutdown margin, the licensee performed calculations specific to St. Lucie 2. The staff requested these calculation results via Request for Information (RAI)-SRXB-1. In summary, the licensee responded as follows:

(1) The methodology used in performing the CEA reactivity worth and the subsequent shutdown margin calculation is the same as currently used in all the St. Lucie Unit 2 neutronics calculations. This methodology,

WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (ADAMS Accession No. ML080630391, non-public), is NRC approved and listed in St. Lucie Unit 2 TS 6.9.1.11.b.1. The computer code used is the PHOENIX-P/ANC code package.

(2) The neutronics calculation for total rod worth and shutdown margin included conservative methodology assumptions such as skewing xenon to the limits of Axial Shape Index (ASI) in Core Operating Limits Report (COLR) Figure 3.2-4, highest worth CEA stuck in the fully withdrawn position and 10% total rod worth uncertainty.

(3) To ensure the elimination of four 4-element CEAs does not pose problems in meeting the COLR limits, a separate sample core loading pattern was developed for Cycle 21 without using the four 4-element CEAs. The excess shutdown margin in this calculation, i.e. removal of the four 4-element CEAs, with the highest worth CEA stuck in the fully withdrawn position, would be reduced by approximately 180 pcm and 30 pcm, but still about 1000 pcm and 300 pcm above the COLR limit of 3600 pcm at the beginning-of cycle (BOC) and at the end-of-cycle (EOC), respectively. Since the worth of the highest stuck rod is loading pattern dependent, the excess shutdown margin varies from cycle to cycle and can be controlled via the core design.

(4) The calculation demonstrated that the shutdown margin requirements with the highest worth CEA stuck in the fully withdrawn position can be met without the use of the four 4-element CEAs. Additionally, if needed, excess shutdown margin can be controlled by altering the core design and fuel load during the design phase of a reload cycle.

The staff reviewed the licensee's evaluation as well as the response for SRXB-RAI-1 and finds them acceptable because the calculation is based on an NRC-approved methodology, and the calculated reduction in shutdown margin due to the removal of the four 4-element CEAs is reasonable and minimal. Therefore, the staff concludes that the proposed TS change satisfies GDC 26 and GDC 27.

3.1.4 Steam Line Break Analysis

A steam line break (SLB) transient results in an uncontrolled increase in steam flow release from the steam generators, with the flow decreasing as the steam pressure drops. This steam flow release increases the heat removal from the reactor coolant system (RCS), which decreases the RCS temperature and pressure. With the existence of a negative moderator temperature coefficient, the RCS cooldown results in a positive reactivity insertion, and consequently a reduction of the core shutdown margin for the post-trip condition. If the most reactive CEA is assumed stuck in the fully withdrawn position after reactor trip, the possibility is increased that the core may become critical and subsequently return to power (RTP). A RTP following a SLB from the post-trip condition is a concern with the high-power peaking factors that may exist when the most reactive CEA is stuck in its fully withdrawn position. Following a SLB, the core is ultimately shut down by the boric acid injected into the RCS by either the emergency core cooling system (safety injection) or the actuation of the Safety Injection Tanks.

Additionally, the event may be terminated due to reaching a dry-out condition in the affected steam generator.

The NRC reviewed the licensee's evaluation for the impact of the removal of four 4-element CEAs on the SLB based on GDC 10, GDC 15, GDC 20 and GDC 26. St. Lucie 2's current design bases to satisfy these design criteria are established by assuming that the most reactive CEA is stuck in its fully withdrawn position, and applying the most limiting single failure of one safety injection train. The analysis result demonstrates that there is no consequential damage to the primary system, and that the core remains in place and intact, by showing that the fuel thermal design basis is satisfied following a SLB.

To evaluate the impact of the removal of four 4-element CEAs on the SLB, the licensee performed calculations specific to St. Lucie 2. The staff issued RAI-SRXB-2 to request the information for the calculation in order to evaluate if the affected SLB meet the GDC 10, GDC 15, GDC 20 and GDC 26. The licensee provided the requested information as summarized below.

(1) The methodology used in performing the steam line break reactivity balance was the same as that used for other St. Lucie Unit 2 neutronics calculations. This methodology, WCAP-11596-P-A, is NRC approved and is listed in St. Lucie Unit 2 TS 6.9.1.11.b.1. The computer code used is the PHOENIX-P/ANC code package.

(2) The Cycle 21 steam line break reactivity balance calculation performed using the WCAP-11596-P-A methodology, assumed conservative methodology assumptions which include highest worth rod stuck in the fully withdrawn position in the limiting (coldest) region of the core, and also analyzing the impact of other high worth rods being stuck in the fully withdrawn position. The statepoint used is conservative with respect to parameters such as the reactor coolant system pressure, temperature and flow.

(3) The Cycle 21 steam line break reactivity balance calculation showed that the worst reactivity change was +18 pcm for the case without the four 4-element CEAs versus the case with the four 4-element CEAs. The corresponding increase in peak linear heat rate was approximately 0.74 kW/ft for the case without the four 4-element CEAs versus the case with the four 4-element CEAs. The peak linear heat rate value with the four 4-element CEAs remained well below the limit of 22 kW/ft.

(4) The calculation showed that the steam line break analysis reactivity balance impact is minimal with the four 4-element CEAs removed, using the same analysis methodology and assumptions. The removal of the four 4-element CEAs is thus justified as being acceptable from core design considerations.

The staff reviewed the licensee's assessment as well as the licensee's response to SRXB-RAI-2, provided in the November 15, 2018, application supplement and found them acceptable because the calculation is based on an NRC-approved methodology, and the impact of the removal of the four 4-element CEAs on the calculated total reactivity balance and associated peak linear heat generation rate is reasonable and minimal. Therefore, the staff concludes that the impact of the proposed TS change on the SLB satisfies GDC 10, GDC 15, GDC 20, and GDC 26.

3.1.5 Core Bypass Flow

The reactor internals are designed to direct the reactor coolant flow through the core and minimize the core bypass flow. The bypass flow is the flow that short circuits the core through the gaps, guide tubes, etc. which does not directly participate in the core cooling.

The NRC staff reviewed the licensee's evaluation for the impact of the removal of four 4-element CEAs on the core bypass flow based on GDC 10. St. Lucie 2's current design basis to satisfy GDC 10 is established by limiting the amount of core bypass flow at about 5% of total core flow as assumed in the safety analyses.

The removal of the 4-element CEAs could potentially increase the bypass flow in the respective fuel assembly corner guide tubes. The licensee provided the evaluation results for the impact of removing the four 4-element CEAs on the core bypass in its response to SRXB-RAI-3, as described below.

(1) St. Lucie Unit 2 transitioned to Framatome fuel beginning with Cycle 23 in Spring 2017. The core bypass flow prior to the implementation of fuel design change was a conservative value of 3.7% of the design reactor core flow. The transition to Framatome fuel increased the core bypass flow by approximately 0.17% to a design bypass flow value of 3.87%. The St. Lucie Unit 2 current Chapter 15 safety analysis conservatively uses a bypass flow of 4.2% of design core flow for all fuel related analyses.

(2) The removal of four 4-element CEAs will increase the core bypass flow by approximately 0.04% of the design core flow. The new bypass flow is estimated to be (3.87% + 0.04%) = 3.91%.

(3) The Framatome (current fuel vendor) reload analysis conservatively assumes a core bypass flow of 4.2% of design core flow, which bounds the bypass flow of 3.91% of design core flow resulting from the removal of the four 4-element CEAs.

The staff reviewed the licensee's assessment as well as the licensee's response to SRXB-RAI-3 provided in the November 15, 2018 application supplement and found them acceptable because the calculation result is reasonable and minimal as expected. Therefore, the staff concludes that the impact of the proposed TS change on the core bypass satisfies GDC 10, GDC 15, GDC 20 and GDC 26.

3.2 Evaluation Summary

The staff finds that the change from 91 to 87 CEAs has been made consistently in the TS DEFINITION and DESIGN FEATURES to support the proposed removal of the four 4-element CEAs. The staff thus concludes that, with these modifications, the TS continue to meet the 10 CFR 50.36 requirement to specify design features and administrative controls.

In addition, based on its evaluation as described above, the staff finds the licensee's proposed TS amendment acceptable because the licensee demonstrated through justifiable assumptions and analyses that the regulatory requirements in 10 CFR 50.36, GDC 10, GDC 15, GDC 20, GDC 25, GDC 26 GDC 27 and GDC 28 continue to be met.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Florida official (Ms. Cynthia Becker, M.P.H., Chief of the Bureau of Radiation Control, Florida Department of Health) on February 26, 2019, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding, which was published in the *Federal Register* on October 9, 2018 (83 FR 50696), that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets 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 amendment.

6.0 CONCLUSION

The Commission has concluded, based on the aforementioned considerations, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Shie-Jeng Peng

Date: April 23, 2019

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING TECHNICAL SPECIFICATION CHANGES TO REDUCE THE NUMBER OF CONTROL ELEMENT ASSEMBLIES (EPID L-2018-LLA-0181) DATED APRIL 23, 2019

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OFFICE	NRR/DORL/LPL2-2/PM	NRR/DORL/LPL2-2/LA	NRR/DSS/SRXB/BC*
NAME	RSchaaf	BClayton	JWhitman
DATE	2/26/2019	3/20/2019	1/18/2019
OFFICE	NRR/DSS/STSB/BC	OGC (NLO)*	NRR/DORL/LPL2-2/BC
NAME	VCusumano	JScro	UShoop
DATE	3/25/2019	4/11/2019	4/22/2019
OFFICE	NRR/DORL/LPL2-2/PM		
NAME	MWentzel		
DATE	4/23/2019		

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