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

August 31, 2015

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
NextEra Energy
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS TO ADOPT TSTF-426, REVISION 5, "REVISE OR ADD ACTIONS TO PRECLUDE ENTRY INTO LIMITING CONDITION FOR OPERATION 3.0.3 – RITSTF INITIATIVES 6b AND 6c" (TAC NOS. MF4631 AND MF4632)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 227 to Renewed Facility Operating License No. DPR-67 and Amendment No. 177 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively. The amendments consist of changes to the St. Lucie 1 and 2 Technical Specifications (TSs) in response to the Florida Power and Light (the licensee) amendment request letter dated August 7, 2014, as supplemented by letters dated February 20 and May 21, 2015.

The amendments change the TSs by adopting Technical Specification Task Force (TSTF) traveler TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into Limiting Condition for Operation 3.0.3 – [Risk Informed TSTF] Initiatives 6b and 6c," which is an NRC-approved change to the Standard Technical Specifications. The amendments provide an additional allowed outage time to restore an inoperable system for conditions under which existing TSs require a plant shutdown.

The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50-389

Enclosures:
1. Amendment No227 to DPR-67
2. Amendment No177 to NPF-16
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 227
Renewed License No. DPR-67

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated August 7, 2014, as supplemented by letters dated February 20 and May 21, 2015, 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. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.227, 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Operating License No. DPR-67
and the Technical Specifications

Date of Issuance: August 31, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 227

TO 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 the 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	Insert
3/4 6-15	3/4 6-15
3/4 6-27	3/4 6-27
3/4 7-20	3/4 7-20
3/4 7-20a	3/4 7-20a

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. 227 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.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: Containment Spray System: MODES 1, 2, and MODE 3 with Pressurizer Pressure $\geq$ 1750 psia.

Containment Cooling System: MODES 1, 2, and 3.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure $\geq$ 1750 psia:

   a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within 72 hours and within 10 days from initial discovery of failure to meet the LCO; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 54 hours.

   b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days and within 10 days from initial discovery of failure to meet the LCO; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   c. With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.

NOTE
Action not applicable when second containment spray train intentionally made inoperable.

   d. With two containment spray trains inoperable, within 1 hour verify TS 3.7.7, "Control Room Emergency Ventilation System," is met, and restore at least one containment spray train to OPERABLE status within 24 hours; otherwise, be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   e. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   f. With any combination of three or more trains inoperable, enter LCO 3.0.3 immediately.

2. Mode 3 with Pressurizer Pressure < 1750 psia:

   a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within the next 6 hours.

   b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately.

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent shield building ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one shield building ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second shield building ventilation system intentionally made inoperable.

b. With two shield building ventilation systems inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one shield building ventilation system to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each shield building ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 10 hours with the heaters on.

b. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

c. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

2. Verifying that the filtration system starts automatically on a Containment Isolation Signal (CIS).

3. Verifying that the filter cooling makeup air and cross connection valves can be manually opened.

4. Verifying that each system produces a negative pressure of $\geq$ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS).

PLANT SYSTEMS

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The control room emergency ventilation system shall be OPERABLE with:

a. Two booster fans,

b. Two isolation valves in each outside air intake duct,

c. Two isolation valves in the toilet area air exhaust duct,

d. One filter train,

e. At least two air conditioning units, and

f. Two isolation valves in the kitchen area exhaust duct.

NOTE

The control room envelope boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3 and 4:

a. With one booster fan inoperable, restore the inoperable fan to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one isolation valve per air duct inoperable, operation may continue provided the other isolation valve in the same duct is maintained closed; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the filter train inoperable for reasons other than an inoperable Control Room Envelope boundary:

1. Immediately initiate action to implement mitigating actions, and

2. Within 1 hour, verify LCO 3.4.8, "Specific Activity," is met, and

3. Within 24 hours restore the filter train to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With only one air conditioning unit OPERABLE, restore at least two air conditioning units to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION: (continued)

MODES 1, 2, 3 and 4: (continued)

NOTE
Action not applicable when second booster fan intentionally made inoperable.

e. With two booster fans inoperable for reasons other than an inoperable Control Room Envelope boundary:

1. Immediately initiate action to implement mitigating actions, and
2. Within 1 hour, verify LCO 3.4.8, "Specific Activity," is met, and
3. Within 24 hours restore at least one booster fan to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when third air conditioning unit intentionally made inoperable.

f. With three air conditioning units inoperable for reasons other than an inoperable Control Room Envelope boundary, restore at least one air conditioning unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

g. With the filter train inoperable due to an inoperable Control Room Envelope boundary:

1. Immediately initiate actions to implement mitigating actions, and
2. Within 24 hours, verify mitigating actions to ensure Control Room Envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, and
3. Restore Control Room Envelope boundary to OPERABLE status within 90 days.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

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.177
Renewed License No. NPF-16

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated August 7, 2014, as supplemented by letters dated February 20 and May 21, 2015, 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.177 are hereby incorporated into the license. The licensee 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Operating License No. NPF-16
and the Technical Specifications

Date of Issuance: August 31, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 177

TO 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 the 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	Insert
3/4 4-9	3/4 4-9
3/4 6-15	3/4 6-15
3/4 6-27	3/4 6-27
3/4 7-17	3/4 7-17

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. 177 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a minimum water level of greater than or equal to 27% indicated level and a maximum water level of less than or equal to 68% indicated level and at least two groups of pressurizer heaters capable of being powered from 1E buses each having a nominal capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of the above required pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE
Action not applicable when second group of required pressurizer heaters intentionally made inoperable.

b. With two groups of required pressurizer heaters inoperable, restore at least one group of required pressurizer heaters to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limits in accordance with the Surveillance Frequency Control Program.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 150 kW in accordance with the Surveillance Frequency Control Program.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power:

a. the pressurizer heaters are automatically shed from the emergency power sources, and

b. the pressurizer heaters can be reconnected to their respective buses manually from the control room after resetting of the ESFAS test signal.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: Containment Spray System: MODES 1, 2, and MODE 3 with Pressurizer Pressure $\geq$ 1750 psia.

Containment Cooling System: MODES 1, 2, and 3.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure $\geq$ 1750 psia:

   a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within 72 hours and within 10 days from initial discovery of failure to meet the LCO; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 54 hours.

   b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days and within 10 days from initial discovery of failure to meet the LCO; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   c. With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.

NOTE
Action not applicable when second containment spray train intentionally made inoperable.

   d. With two containment spray trains inoperable, within 1 hour verify TS 3.7.7, "CREACS," is met and restore at least one containment spray train to OPERABLE status within 24 hours; otherwise, be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   e. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   f. With any combination of three or more trains inoperable, enter LCO 3.0.3 immediately.

2. Mode 3 with Pressurizer Pressure < 1750 psia:

   a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within the next 6 hours.

   b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately.

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM (SBVS)

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Shield Building Ventilation Systems shall be OPERABLE.

APPLICABILITY: At all times in MODES 1, 2, 3, and 4.
In addition, during movement of recently irradiated fuel assemblies or during crane operations with loads over recently irradiated fuel assemblies in the Spent Fuel Storage Pool in MODES 5 and 6.

ACTION:

a. With the SBVS inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the Spent Fuel Pool area, only ACTION-c is required. If the SBVS is inoperable for any other reason, concurrently implement ACTION-b and ACTION-c.

b. (1) With one SBVS inoperable in MODE 1, 2, 3, or 4, restore the inoperable system to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second SBVS intentionally made inoperable.

(2) With both SBVSs inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one SBVS to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. (1) With one SBVS inoperable in any MODE, restore the inoperable system to OPERABLE status within 7 days; otherwise, suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

(2) With both SBVS inoperable in any MODE, immediately suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Shield Building Ventilation System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Performing a visual examination of SBVS in accordance with ASME N510-1989.

PLANT SYSTEMS

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (CREACS)

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room emergency air cleanup systems shall be OPERABLE with:

a. A filter train and its associated fan per system, and
b. At least one air conditioning unit per system, and
c. Two isolation valves in the kitchen area exhaust duct, and
d. Two isolation valves in the toilet area exhaust duct, and
e. Two isolation valves in each (North and South) air intake duct.

NOTE
The control room envelope boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

a. With one control room emergency air cleanup system inoperable for reasons other than an inoperable Control Room Envelope boundary, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more control room emergency air cleanup systems inoperable due to an inoperable Control Room Envelope boundary:

1. Immediately initiate actions to implement mitigating actions, and
2. Within 24 hours, verify mitigating actions to ensure Control Room Envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, and
3. Restore Control Room Envelope boundary to OPERABLE status within 90 days.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With an isolation valve in an air intake duct or air exhaust duct inoperable, operation may continue provided the other isolation valve in the same air intake or air exhaust duct is maintained closed; otherwise be in at least HOT STANDBY in the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second CREACS train intentionally made inoperable.

d. With two control room emergency air cleanup systems inoperable for reasons other than an inoperable Control Room Envelope boundary:

1. Immediately initiate action to implement mitigating actions, and
2. Within 1 hour, verify LCO 3.4.8, "Specific Activity," is met, and
3. Within 24 hours restore at least one CREACS train to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR AMENDMENT NO. 227 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-67 AND AMENDMENT NO. 177 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16 FLORIDA POWER AND LIGHT COMPANY, ET AL. ST. LUCIE PLANT, UNIT NOS. 1 AND 2 DOCKET NOS. 50-355 and 50-389

1.0 INTRODUCTION

By letter dated August 7, 2014 (Agencywide Documents Accession and Management System (ADAMS) Accession No. ML14225A630), as supplemented by letters dated February 20, 2015 (ADAMS Accession No. ML15065A235), and May 21, 2015 (ADAMS Accession No. ML 15154B051), Florida Power and Light Company (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC or Commission) for changes to the St. Lucie Nuclear Plant, Unit Nos. 1 and 2 (St. Lucie Units 1 and 2), Technical Specifications (TSs).

The proposed changes would provide an additional allowed outage time to restore an inoperable system for conditions under which existing TSs require a plant shutdown.

The revisions are consistent with Commission-approved Technical Specification Task Force Standard Technical Specifications (STSs) Change Traveler 426 (TSTF-426), Revision 5, "Revise or Add Actions to Preclude Entry into Limiting Condition for Operation (LCO) 3.0.3 – RITSTF [Risk-Informed TSTF] Initiatives 6b & 6c" (Reference 1). Revision 5 of the TSTF-426 was issued in the *Federal Register* (FR) on May 30, 2013 (78 FR 32476).

Traveler TSTF-426 incorporated the approved Westinghouse Topical Report (TR) WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (Reference 2), into NUREG-1432, "Standard Technical Specifications Combustion Engineering [CE] Plants." The TR WCAP-16125 provided the justification for risk-informed TS (RITS) Initiative 6 for nuclear plants with CE-designed nuclear steam supply systems. RITS Initiative 6 modifies selected exigent shutdown actions to allow a risk-informed operating time prior to shutdown.

The NRC staff published a no significant hazards consideration determination in the *Federal Register* on March 17, 2015 (80 FR 13908). The supplements dated February 20 and May 21, 2015, provided additional information that clarified the application but 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.

2.0 REGULATORY EVALUATION

2.1 Proposed TS Changes

TR WCAP-16125 justified modifications to various TSs to add a Condition for loss of redundant features representing a loss of safety function for a system or component included within the scope of the plant TSs. It would replace Required Actions requiring either a default shutdown or explicit LCO 3.0.3 entry with a Required Action based on the risk significance for the system's degraded condition. The Condition being added is for redundant trains discovered to be inoperable. The Condition only applies to discovery of an emergent condition resulting in redundant trains being inoperable, not from the second train intentionally made inoperable. The Allowed Outage Times (AOTs) associated with the proposed actions are specified. The AOTs are intentionally of short duration to allow for restoring the system to an operable condition, thereby avoiding the risk associated with an immediate controlled shutdown. For each TS change, TR justifies a 24-hour AOT, which is described as Completion Time in the TR. Table 1 summarizes the St. Lucie Units 1 and 2 TS changes.

Table 1

TS LCO	SYSTEM/COMPONENT	CONDITION	CURRENT AOT	PROPOSED AOT
3.4.3	Pressurizer (Unit 2 Only)	Two groups of class 1E heaters inoperable	Default shutdown - Mode 3 in 6 hours	24 Hours
3.6.2.1	Containment Spray and Cooling Systems (CSCS)	Two containment spray trains inoperable (both containment cooling trains must be operable)	Explicit LCO 3.0.3 entry	24 Hours†
3.6.6.1	Shield Building Ventilation System (SBVS)	Two trains inoperable	None/ LCO 3.0.3 – Unit 1; Explicit LCO 3.0.3 – Unit 2	24 Hours*
3.7.7.1	Control Room Emergency Ventilation System (CREVS) – Unit 1; Control Room Emergency Air Cleanup System (CREACS) – Unit 2	Two trains inoperable (Modes 1-4) for reasons other than an inoperable control room boundary	None/ LCO 3.0.3 - default shutdown – Unit 1; Explicit LCO 3.0.3 – Unit 2	24 Hours**

† Must include verification that the LCO for CREVS is met
* Must include verification that at least one train of the containment spray (CS) system is available
** Must include verification that LCO 3.4.8 "Specific Activity" is met.

By letter dated February 20, 2015 (Reference 12), the licensee withdrew changes to TS 3.1.2.2 for both units that were originally proposed as a variation from TSTF-426.

TR WCAP-16125 stated that the change to the pressurizer TS is not applicable to St. Lucie Unit 2. After examining the current TS for both units, the NRC staff concluded that the TR was in error and the change to the pressurizer TS is applicable to Unit 2. The licensee's application included proposed changes to Unit 2 pressurizer TSs.

2.2 Regulatory Requirements and Guidance

The Commission's regulatory requirements related to the content of the TS are contained in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." Pursuant to 10 CFR 50.36(c) the TSs are required to include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation at 10 CFR 50.36(c)(2) states: "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the Technical Specifications until the condition can be met."

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 3), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. RG 1.174 also provides risk acceptance guidelines for evaluating the results of such evaluations.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 4). Section 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed change meets the following key principles:

1. The proposed change meets the current regulations, unless it explicitly relates to a requested exemption.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
5. The impact of the proposed change should be monitored using performance measurement strategies.

The NRC staff reviewed the licensee's proposed changes against (1) the requirements of 10 CFR 50.36, (2) the STS changes approved for adoption in the Notice of Availability of TSTF-426 issued in the FR on May 30, 2013 (78 FR 32476), and (3) the methodology approved in TR WCAP-16125, as documented in a Safety Evaluation (SE) dated May 24, 2010 (Reference 5). The TR WCAP-16125 was reviewed against RG 1.174 and SRP Section 19.2.

3.0 TECHNICAL EVALUATION

3.1 Conformance with the Five Key Principles of SRP Section 19.2 as Summarized in the SE of TR WCAP-16125

The changes proposed in TSTF-426 are consistent with NRC-approved TR WCAP-16125. In its SE (Reference 5), the NRC staff evaluated TR WCAP-16125 for conformance with the five key principles of SRP Section 19.2.

3.1.1 Compliance with Current Regulations

The regulations at 10 CFR 50.36 permit either a plant shutdown or other remedial actions specified by TSs when an LCO is not met. The proposed changes provide new action requirements for conditions of equipment inoperability that currently require an immediate plant shutdown. Since such remedial actions are permitted per 10 CFR 50.36, the proposed changes continue to comply with current regulations, and therefore, satisfy this key principle.

3.1.2 Defense-in-Depth

The proposed changes address conditions where both trains of a system are inoperable, resulting in a loss of that system's function and a temporary reduction in the defense-in-depth capabilities of the plant. Each proposed change addresses the remaining available alternative system(s) capable of providing mitigation of events, and, where applicable, includes requirements to assure these required backup systems are operable. The reduced level of defense-in-depth is retained by verification that both trains (if applicable) of the backup system are operable. Therefore, this key principle is satisfied by the unique requirements identified for each proposed TS change.

3.1.3 Safety Margins

The proposed changes do not have any impact on the use of NRC-approved codes and standards, nor do the changes impact any acceptance criteria used in a plant's licensing basis. Under the current TSs, if an accident occurs during the 6-hour controlled shutdown time of LCO 3.0.3 caused by two trains of these systems being unavailable, it could potentially result in offsite dose limits that do not meet NRC regulatory limits. Since the changes proposed do not modify the design basis of the systems evaluated, extending the Allowed Outage Time to 24 hours would have no quantitative effect on the dose consequence as compared to the existing condition. As such, the proposed changes would not significantly reduce the plant's available safety margin, and this key principle is satisfied.

3.1.4 Performance Monitoring

The proposed changes would permit continued plant operation for short periods to address emergent equipment failures. Degradation of equipment performance could lead to excessive use of the new action requirements. This is adequately addressed by equipment performance monitoring required by 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and therefore, this key principle is satisfied.

3.1.5 Risk Assessment

The risk of each of the TS LCOs for which action requirements are proposed is evaluated in TR WCAP-16125 by three methods, as described below.

Method 1:

For calculations of ΔCDF [Delta core damage frequency], a bounding approach was applied to evaluate loss of function of a system by identifying the initiating events for which the system provides mitigation, and assuming that the event goes directly to core damage. No credit was taken for alternate mitigation strategies, and the baseline CDF was effectively assumed to be zero. The initiating event frequencies were taken from NUREG/CR-5750, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987 – 1995" (Reference 6).

The licensee verified that initiating frequencies in NUREG/CR-5750 are bounding for St. Lucie Units 1 and 2.

For ΔLERF [large early release frequency], a simplified approach using an event tree was developed to calculate the fraction of core damage events which result in large early releases. The event tree assessed containment isolation status, reactor coolant system (RCS) pressure, secondary side depressurization via the steam generators, thermally induced Steam Generator Tube Rupture (SGTR), and reactor pressure vessel (RPV) lower head failure. Assumptions related to the potential impact on LERF for each of these events, and the associated basis for probabilities used in the analysis, are discussed below:

Containment Isolated - This event defines containment integrity prior to the core damage event. If containment is not isolated, then a large early release will result concurrent with core damage. A probability of 3.0E-3 was applied for an unisolated containment, which is identified as the upper end of the range used in the CE Probabilistic Risk Assessment models in TR WCAP-16125.

RCS Pressure - High - This event defines the RCS pressure at the time of core damage. If the pressure is low, then large early releases are assumed not to occur (except via an unisolated containment); otherwise, thermally induced SGTR and high pressure melt ejection events are further evaluated. All core damage events involving loss-of-coolant accidents (LOCAs) are assumed to result in low or intermediate RCS pressure, and all other events result in high RCS pressure.

Steam Generator Depressurization - This event defines the status of the secondary side, and affects the next event which is the potential for induced SGTR. Depressurization of the secondary side occurs either due to prior operator response or due to failure of a safety relief valve. Based on NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture" (Reference 7), a probability of 0.9 is assigned for secondary depressurization.

Thermally Induced SGTR Occurs - This event represents a loss of steam generator tube integrity due to thermal stresses during a severe accident, which is assumed to result in a large early release. Two values are used, based on the status of the prior event, for

steam generator depressurization. A probability of 0.5 is assigned when the steam generators are depressurized, and 0.01 otherwise. These values are conservative, based on the assumptions regarding tube age and integrity and based on not crediting operator actions to depressurize the RCS after core damage.

RPV Lower Head Failure Results in Containment Failure - This event represents a high pressure failure of the lower head, with an energetic discharge of the molten fuel and direct containment heating, leading to failure of containment. Based on NUREG/CR--6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments" (Reference 8), the conditional containment failure probability given the event for CE-designed plants is 0.01, which is considered to be a bounding value.

None of the assessed initiating events include either SGTRs or other containment bypass events because the systems being evaluated do not mitigate these events. The NRC staff concludes that the simplified LERF event tree is reasonable and acceptable to support the evaluation of LERF for the scope of TR WCAP-16125.

Method 2:

For pressurizers, an evaluation of the increased likelihood of a plant trip due to degraded pressure control is made in order to calculate ΔCDF. The ΔLERF calculation for this TS is the same simplified approach described above for Method 1 and is acceptable.

Method 3:

The remaining systems (and associated TSs) associated with mitigation of radiological releases with magnitudes less than those associated with LERF are: CSCS (TS 3.6.2.1); SBVS (TS 3.6.6.1); and CREVS (U1 – TS 3.7.7.1) / CREACS (U2 - TS 3.7.7). There is no impact to either CDF or LERF, as the systems are provided to meet design basis dose limits. As described in TR WCAP-16125, an evaluation of the frequency of events that challenge the systems was made and compared to the acceptance guidelines of RG 1.174 applicable to ΔLERF in order to characterize the risk of these lesser releases.
TR WCAP-16125 provided additional justification based on the availability of other systems that provide a degree of defense-in-depth for prevention of these releases.

To assess the impact of the unavailability of these systems, TR WCAP-16125 examined the expected iodine releases for three categories of events:

- Beyond design basis scenarios that lead to large early releases,
- Maximum Hypothetical Accident (MHA), and
- LOCA and Non-LOCA Design-Basis Accidents (DBA).

The purpose of this assessment was to show that, using worst case assumptions, the potential accident releases anticipated under the short-term operational conditions proposed by the increased AOT for the SBVS will be well below and bounded by a large early release. For clarity, the TR WCAP-16125 evaluation was limited to the release of iodine. For each category, iodine releases were estimated assuming various combinations of system

availability. The results of this assessment are shown in Table 4.3-1 of TR WCAP-16125, supplemented by request for additional information (RAI) responses (Reference 9). The NRC staff reviewed the assumptions and methodology used to determine the bounding iodine release quantities and resulting dose consequences and found that in all cases appropriately conservative assumptions were used.

To reduce the impact of an increased AOT for the CREVS/CREACS, TR WCAP-16125 added conditions to verify that specific activity of the reactor coolant is within limits and to verify that dose mitigating actions are available in the control room. For limited durations, such as the short-term operational conditions proposed by the increased AOT for the CREVS/CREACS, the NRC staff has accepted credit for the use of respirators and potassium iodide on an interim basis to demonstrate that control room dose limits can be met.

Similarly, TR WCAP-16125 added pre-planned actions to ensure that the impact of loss of postaccident temperature control associated with an increased AOT for the CREVS/CREACS is mitigated. Actions can include use of portable fans, temporary opening of doors or use of normal heating, ventilation, and air conditioning systems. To support this change, administrative controls will be provided to monitor the control room temperature to ensure control room habitability and operability of TS equipment. If compensatory measures impact the control room envelope, the operability of containment and auxiliary building postaccident air cleanup systems will be verified. The 24-hour AOT proposed in TR WCAP-16125 for the CREVS/CREACS is consistent with the allowed 24-hour period for the evaluation of a breach of the control room envelope provided in Traveler TSTF-448 (Reference 10).

Based on an evaluation of the methods and assumptions used, the NRC staff has reasonable assurance that the postulated accident releases calculated for the short-term operational conditions proposed by the increased AOT for the SBVS will be well below the LERF releases. In addition, the NRC staff has reviewed the bases for the increased AOT for the CREVS/CREACS and has determined that the proposed conditions and compensatory measures provide reasonable assurance that control room habitability will be adequately maintained during the proposed 24-hour AOT.

External events, including internal fires and floods, were not evaluated in TR WCAP-16125. None of the systems being evaluated provide a primary mitigating function for external events, and, therefore, these events are not significant to the risk-informed decision.

TR WCAP-16125 also evaluated sensitivity studies for key areas of uncertainty in the analyses. Specifically, TR WCAP-16125 considered uncertainties in the initiating event frequencies that are the input to the CDF calculations and showed that even assuming a 95 percent upper bound frequency would not result in excessive risk. These were also propagated into the LERF calculations with similar results. TR WCAP-16125 also addressed uncertainties in the thermally induced SGTR assumptions and SG depressurization assumptions, and demonstrated that the LERF results are not significantly impacted. The NRC staff concludes that these sensitivity studies performed to evaluate the key sources of uncertainty in the risk analyses adequately demonstrate the robustness of the results to support the proposed TS changes.

3.2 NRC Staff Evaluation of the Proposed TS Changes

The licensee withdrew (Reference 12) changes that originally were proposed to TS 3.1.2.2 for both units. The remaining substantive variation from TSTF-426 Revision 5, listed as item 3 in Section 2.2, "Optional Changes and Variations," of the license amendment request, is addressed in section 3.2.4 of this safety evaluation.

3.2.1 TS 3.4.3 (Unit 2 Only)

The pressurizer and the Class 1E electrical heaters maintain a liquid-to-vapor interface to permit RCS pressure control during normal operations and in response to anticipated design basis transients. The Class 1E heaters, with their power provided by emergency AC power busses, are used to maintain RCS subcooling during a natural circulation cooldown, and the unavailability of the heaters will extend the time to reach entry conditions for the shutdown cooling system. The unavailability of the Class 1E heaters may complicate steady-state RCS pressure control and may increase the potential of an unplanned reactor trip. However, the availability of additional heaters beyond the two groups required by this TS LCO permit continued RCS pressure control.

The current LCO 3.4.3 does not provide any action for two groups of pressurizer heaters inoperable; therefore, ACTION b applies, which requires an immediate plant shutdown. The proposed change provides for a 24-hour AOT to restore at least one group of pressurizer heaters to operable status, to permit continued operation.

The unavailability of the Class 1E pressurizer heaters would not have any significant impact on plant transient response, and so there is no quantifiable impact to CDF or LERF. While mitigation of a SGTR is enhanced by the availability of pressurizer heaters, the non-Class 1E heaters can also function if offsite power is available, and plant procedures provide for mitigation of a SGTR without pressurizer heaters, if necessary.

Conservatively, the risk result due to increased likelihood of a reactor trip was calculated by assuming an order-of-magnitude increase in the reactor trip frequency when both Class 1E heater groups are inoperable. The risk result is then calculated based on the conditional core damage probability given a reactor trip with no other complications:

ΔCDF	RG 1.174 Guidance	ΔLERF	RG 1.174 Guidance
1.0E-7/yr	<1.0E-6/yr	3.8E-9/yr	<1.0E-7/yr

The ΔCDF and ΔLERF were assessed based on a bounding once per 3-year entry into the proposed action requirement from TR WCAP-16125 and assumed that the entire 24-hour duration of the AOT is used. The risk results are well below the acceptance guidelines of RG 1.174 as noted in the table.

Minimum pressurizer heater capability is supplemented by the normal availability of non-Class 1E heaters for normal plant pressure control, and the availability of plant procedures that provide plant shutdown and cooldown guidance with or without pressurizer heaters. If the available heaters are sufficient to maintain RCS pressure control, normal plant operations can continue. Because unavailability of Class 1E and non-Class 1E heaters would physically result

in plant shutdown, the NRC staff does not consider it necessary to specify additional TS or administrative requirements for the non-Class 1E heater availability.

TS LCO 3.4.3 does not contain a specific ACTION for two required groups of pressurizer heaters inoperable. As a result, ACTION b applies which specifies a plant shutdown. A new ACTION b is being added for two required pressurizer heater groups inoperable, which requires restoration of at least one required group of pressurizer heaters to operable status within 24 hours. The action is modified by a note stating it is not applicable when the second group of required pressurizer heaters is intentionally made inoperable.

As stated above, the conservatively-calculated risk result is within the acceptance guidelines of RG 1.174, and there would not have a significant impact on plant transients (i.e., plant shutdown and cooldown) without pressurizer heaters. Therefore, the NRC staff finds the proposed new action requirement and 24-hour AOT acceptable.

3.2.2 TS 3.6.2.1

The CS system and the containment coolers provide containment heat removal following accidents which release high energy steam to the containment. In addition to the heat removal function, the CS system enhances postaccident fission product removal. Each train of the CS system provides a nominal 50 percent of the cooling function, and similarly each train of the containment coolers provides 50 percent of the cooling function; thus the combined capacity of both systems is 200 percent.

LCO 3.6.2.1 provides for an explicit LCO 3.0.3 entry when less than 100 percent containment cooling capacity is available (i.e., any combination of three or more trains inoperable) or when both CS trains are inoperable, and therefore the fission product removal function is not available.

The RAI responses to the TR WCAP-16125 (Reference 11) proposed a 24-hour AOT for LCO 3.6.2.1 consistent with the other iodine removal TS changes. The RAI responses also identified that the TS-required operability of the containment coolers would provide a similar iodine removal function such that additional TS requirements for operability of other iodine removal systems would not be required. A TS action for operability of the CREVS/CREACS was proposed to assure additional defense-in-depth for control room functionality when both CS trains are inoperable during the 24-hour AOT.

Based on the information in TR WCAP-16125, the challenge frequency of the CS system for fission product removal is identical to the challenge frequency described for the SBVS described below. Similar to that analysis, it may be conservatively assumed that if both CS trains are unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event that would require the unavailable system would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per 3-year entry into the new TS would result in a frequency of a "less than LERF" release of about 9.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore, provides a context for consideration of the risk result for smaller releases.

When the function of the CS for fission product removal is unavailable, then the operability of the CREVS/CREACS, which provides for filtration to protect control room habitability, will be verified as a defense-in-depth measure.

LCO 3.6.2.1 contains existing ACTION e, which applies when two CS trains are inoperable or any combinations of three or more trains are inoperable. ACTION e required entering LCO 3.0.3 immediately. The proposed change modifies ACTION e to no longer apply when two CS trains are inoperable. Existing ACTION e is then renamed ACTION f. Existing ACTION d applies when two containment cooling trains are inoperable and requires restoration of one train within 72 hours. This action is renamed ACTION e.

A new ACTION d is added for two CS trains inoperable with required actions to verify within 1 hour that at least one train of CREVS (Unit 1)/CREACS (Unit 2) is operable and to restore at least one train of CS within 24 hours. ACTION d is modified by a note stating it is not applicable when the second CS train is intentionally made inoperable.

TR WCAP-16125 states that ACTION d is applicable when two CS trains are inoperable provided that at least one containment air cooler is operable. This restriction is imposed by revised ACTION f, which addresses any combination of three or more trains inoperable with a Required Action to enter LCO 3.0.3 immediately.

As stated above, the low risk result for severe accidents (CDF and LERF) is well below the acceptance guidelines of RG 1.174, and there is verification of operability of the CREVS/CREACS. Therefore, the NRC staff finds a new action requirement with a 24-hour AOT is acceptable for the case of both CS trains inoperable.

3.2.3 TS 3.6.6.1

The SBVS functions to assure radioactive material released from containment leakage following a DBA is filtered prior to being exhausted to the environment. Both units at St. Lucie have dual containment system consisting of a steel containment shell, and an air annulus surrounded by a concrete shield building. Each system includes two redundant trains with high efficiency particulate air filters, moisture absorbers, and charcoal adsorbers in the flowpath. The SBVS filters leakage from containment (steel shell) into the shield building for dual containment facilities. The design basis for this system is a postulated MHA involving a LOCA with a short duration uncovery of fuel, resulting from a temporary interruption, or significant degradation, of the emergency core cooling system flow. The event is assumed to result in significant iodine releases (40 – 50 percent of core inventory) from the fuel into the containment. The containment remains intact, with no more than the design basis leakage permitted by TSs. Releases associated with the MHA are significantly below the release that would occur for a postulated large early release (at least two orders of magnitude lower). This system does not provide any mitigation capability for preventing either core damage or large early releases.

The current TS for Unit 1 does not address the condition of two inoperable trains of this system; therefore, a default LCO 3.0.3 entry is required, resulting in an immediate plant shutdown. For Unit 2, ACTION b.(2) applies, which directs an immediate entry into LCO 3.0.3 resulting in a plant shutdown. The proposed change would provide a 24-hour AOT to restore at least one train of the affected system to operable status, to permit continued operation under an existing action requirement.

As noted above, this system does not provide any core damage or large early release mitigation. Therefore, the risk results are zero for these systems. However, it may be conservatively assumed that if this system is unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event that would require the unavailable system, would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per 3-year entry into the new TS would result in a frequency of a "less than LERF" release of about 9.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore provides a context for consideration of the risk result for smaller releases.

As noted in TR WCAP-16125, there are also higher frequency DBAs (e.g., rod ejection and reactor coolant pump locked rotor) that are assumed to result in fuel damage and, therefore, rely upon this system to filter any containment leakage. These accidents are associated with releases from the fuel into containment two or more orders of magnitude below those associated with the MHA described above, and four or more orders of magnitude below large early releases.

Containment spray can effectively scrub the postaccident containment atmosphere of fission products and, therefore, reduce reliance upon the downstream air cleanup systems. In order to assure additional defense-in-depth protection for the spectrum of accidents for which this system provides mitigation, the TS action will include a verification of operability of at least one train of the CS system.

TS LCO 3.6.6.1 for Unit 1 does not contain an action for both SBVS trains inoperable. As a result, this condition would require immediate entry into LCO 3.0.3. For Unit 1, a new ACTION b is added, which applies when two SBVS trains are inoperable. It requires verification within 1 hour that at least one train of CS is operable and allows 24 hours to restore at least one SBVS train to operable status. ACTION b is modified by a note stating it is not applicable when the second SBVS train is intentionally made inoperable.

TS LCO 3.6.6.1 for Unit 2 includes an immediate explicit LCO 3.0.3 entry in current ACTION b.(2) for both SBVS trains inoperable. ACTION b.(2) is modified to require verification within 1 hour that at least one train of CS is operable and allows 24 hours to restore at least one SBVS train to operable status. ACTION b.(2) is additionally modified by a note stating it is not applicable when the second SBVS train is intentionally made inoperable.

The low risk result for severe accidents is well below the acceptance guidelines of RG 1.174, and there is an additional restriction on operability of at least one CS train in the TSs. Therefore, the NRC staff finds the proposed new action requirements and the 24-hour AOTs are acceptable.

3.2.4 TS 3.7.7

According to the licensee's request for license amendment dated August 7, 2014, the CREVS for unit 1 and the CREACS for Unit 2 provide both:

- filtration of outside air delivered to the control room by the ventilation system in the event of radioactive releases of particulates or iodine from containment following an accident involving fuel failures; and
- temperature control of the control room when it is isolated during accident conditions to assure control room temperature will not exceed equipment operability requirements.

These systems assure that control room personnel are protected from potential radiation exposures in excess of regulatory limits during an accident. The system may also provide protection of control room personnel from chemical or toxic gas releases by isolating the control room air intakes.

For Unit 1, CREVS differs in design from CREACS being that CREVS is not physically a two independent train system. According to the licensee "A common filter train is shared by what otherwise are two physically and electrically independent CREVS trains." The current TS LCO 3.7.7.1 addresses the loss of the single filter train in ACTION c with 24 hours to restore the train to operable status. Currently the LCO does not address the loss of CREVS function caused by the loss of both booster fans or inoperability of all air conditioning in Modes 1, 2, 3, or 4, therefore LCO 3.0.3 would apply and result in a plant shutdown. The proposed changes include a new ACTION e for two booster fans inoperable and 24 hours to restore at least one booster fan to operable status and a new ACTION f for three air conditioning units inoperable. The current TS already provide a 24-hour AOT when both trains are inoperable due specifically to control room envelope boundary inoperability as existing Action e.

For Unit 2, the current TS LCO 3.7.7 addresses the condition of two inoperable trains with an explicit LCO 3.0.3 entry, resulting in an immediate plant shutdown. The proposed change would provide a 24-hour AOT to restore at least one train of the CREACS to operable status, to permit continued operation under an existing action requirement. The current TSs already provide a 24-hour AOT when both trains are inoperable due specifically to control room pressure boundary inoperability.

In the event of an accident involving radioactive releases without the availability of the CREVS/CREACS, there would be no direct impact on the capability of the control room staff to perform any actions required to mitigate severe core damage or large early releases, because alternative protective measures would be implemented to reduce the dose impacts. If the accident did not involve severe core damage, control room doses even without the CREVS/CREACS would be minimal, and therefore the CREVS/CREACS has no direct role in preventing core damage (i.e., $\Delta CDF = 0$).

Similarly TR WCAP-16125 stated that the unavailability of the temperature control function of these systems has a negligible impact on severe accident risk, based on long room heatup times, availability of alternate cooling strategies, and alternate means to control emergency systems locally. The NRC staff reviewed the basis for this conclusion and considered the potential plant impacts if an accident occurred that isolated the control room while the control room emergency air temperature control system (CREATCS) was inoperable.

If an accident occurred which isolated the control room without cooling, and core cooling was being maintained, then there would be negligible radiological consequences and the operators could simply un-isolate and realign the normal control room ventilation system to provide continued cooling of the CR. Therefore, the NRC staff concludes that there would be no impact on CDF (i.e., $\Delta CDF = 0$).

If a core damage accident did occur with CREVS/CREACS unavailable (including the cooling function), then the bounding impact would be to simply assume the event proceeded to a large early release based on the unavailability of the control room personnel to perform any mitigating actions. The NRC staff concludes that this assumption is conservative, because large releases occur primarily due to containment bypass accidents, and control room actions following core damage do not prevent the release from occurring.

A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event, which with the CREVS/CREACS unavailable is assumed to proceed to a large early release, would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per 3-year entry into the new TS, and assuming the entire 24-hour duration of the AOT is used, the conservatively calculated $\Delta LERF$ is about 9.0E-8/year. This $\Delta LERF$, and the zero ΔCDF, are below the acceptance guidelines of RG 1.174.

A significant contributor to control room radiological hazards was identified in TR WCAP-16125 from the release of radioactive RCS fluid from a SGTR event. A required TS action to verify LCO 3.4.8, "RCS Specific Activity," is met will be included in the new proposed action to provide additional defense-in-depth.

For the system's cooling function defense-in-depth is provided by alternative control room cooling actions and by the capability for local operation of equipment, if necessary. These actions are typically found in plant procedures, and are not required to be implemented by TS controls. The licensee confirmed in the license amendment request that plant procedures can establish temporary alternate means of control room cooling.

TR WCAP-16125 also addressed a TS action to require initiation of mitigating actions to lessen the effects of potential hazards of smoke, chemical, radiological, or toxic gas releases. The NRC staff considers the specific hazards and compensatory measures to be plant-specific, and did not find sufficient information to conclude that the proposed changes are acceptable for these events without a plant-specific evaluation. The RAI response (Reference 11) identifies that these mitigating actions were previously reviewed and approved by the NRC staff for Traveler TSTF-448 (Reference 10). TSTF-448 authorizes a generic TS change to permit a

24-hour AOT when the control room boundary is inoperable, and includes the same mitigating actions to assure protection of the control room staff from non-radiological hazards. For St. Lucie Units 1 and 2 license amendments for the incorporation of TSTF-448 were approved via letter dated September 30, 2008 (ADAMs Accession No. ML082630416). Those amendments established more effective and appropriate actions, surveillance, and administrative TS requirements related to ensuring the habitability of the control room envelope in accordance with the NRC-approved TSTF-448, Revision 3.

The TR WCAP-161254 justifies a 24-hour AOT for two CREVS/CREACS trains inoperable for any reason provided that mitigating actions are implemented immediately and it is verified that LCO 3.4.8, "RCS Specific Activity," is met within 1 hour.

For Unit 1 TS LCO 3.7.7.1, ACTION c applies when the CREVS filter train is inoperable for reasons other than an inoperable control room boundary in Modes 1, 2, 3, or 4 and requires restoration of the filter train within 24 hours. This action is revised to immediately implement mitigating actions, and verify LCO 3.4.8, "Specific Activity" is met within 1 hour. A new ACTION e is added to apply when two booster fans are inoperable and ACTION f is added to apply when three air conditioning units are inoperable in Modes 1, 2, 3 and 4. Both actions apply for reasons other than an inoperable control room envelope boundary, which is addressed in existing ACTION e. Both new ACTIONs e and f are modified by a note stating that the actions are not applicable when the second booster fan or third air conditioning unit is intentionally made inoperable. Existing ACTION e is renamed ACTION g.

For Unit 2 TS LCO 3.7.7, ACTION d applies when two CREACS trains are inoperable due to any reason other than an inoperable control room boundary in Modes 1, 2, 3, or 4 and requires entering LCO 3.0.3 immediately. ACTION d is revised to: immediately implement mitigating actions, verify LCO 3.4.8, "Specific Activity" is met within 1 hour and require restoration of at least one CREACS train to operable status within 24 hours. ACTION d is additionally modified by a note stating it is not applicable when the second CREATCS train is intentionally made inoperable.

Based on the risk result being below the acceptance guidelines of RG 1.174 and the additional restriction on meeting RCS specific activity limits in the TSs, the NRC staff finds the proposed new action requirements and 24-hour AOT acceptable.

3.3 TS Bases Changes

TSTF-426 included, and the licensee submitted the following TS bases changes, for information purposes:

- A reference to the NRC-approved TR WCAP-16125 has been added to the reference section of the TS Bases for each TS affected in TSTF-426.

- Revisions to reflect the changes to the TS.

- For all affected TS, a Note on each applicable condition was added that states: "Not applicable when second [system or component name] intentionally made inoperable." The Bases are revised to provide additional explanation of the Note: "The Condition is

modified by a Note stating it is not applicable if the second [system or component name] is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one [system or component name] is inoperable for any reason and the second [system or component name] is discovered to be inoperable, or if both [system or component name] are discovered to be inoperable at the same time."

The NRC staff determined that TS Bases changes are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission's Final Policy Statement on TSs Improvements for Nuclear Power Reactors, dated July 2, 1993 (58 FR 39132). As stated in 10 CFR 50.36(a)(1), TS bases are not part of TSs.

3.4 Technical Evaluation Summary

The NRC staff has reviewed the proposed changes against approved Traveler TSTF-426, which was based on approved TR WCAP-16125 (using the five key principles of risk-informed decision making) and finds that the proposed changes are acceptable. Additionally, appropriate TS notes are provided that assure that the loss of safety function action requirements are not applicable for operational convenience and that voluntary entry into these action requirements in lieu of other alternatives that would not result in redundant systems or components being inoperable are prohibited.

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 July 20, 2015,[1] 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 has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on March 17, 2015 (80 FR 13908). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

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[1] The NRC staff notified the State official by telephone and by e-mail. The e-mail is in ADAMS under Accession No. ML15202A008.

6.0 CONCLUSION

Based on the aforementioned considerations, the NRC staff concluded 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. TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into LCO 3.0.3 – RITSTF Initiatives 6b & 6c," dated November 22, 2011 (ADAMS Accession No. ML113260461)

2. TR WCAP-16125-NP-A, Revision 2, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," dated August 2010 (ADAMS Package Accession No. ML110070498)

3. Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," NRC, dated May 2011 (ADAMS Accession No. ML100910006)

4. NUREG-0800, Standard Review Plan, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007 (ADAMS Accession No. ML071700658)

5. Final SE of Pressurized Water Reactor Owners' Group TR WCAP-16125-NP, Revision 2, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," dated May 24, 2010 (ADAMS Package Accession No. ML093560466)

6. NUREG/CR-5750, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987 –1995," dated February 1999 (ADAMS Accession No. ML070580080)

7. NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," dated March 1998 (ADAMS Accession No. ML070570094)

8. NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," dated February 1996 (ADAMS Accession No. ML081920672)

9. Responses to the NRC RAI on TR WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," dated August 10, 2009 (ADAMS Accession No. ML092260399)

10. TSTF-448-A, Revision 3, "Control Room Habitability," dated August 8, 2006, and corrected pages dated December 29, 2006 (ADAMS Accession Nos. ML062210095 and ML063630467)

11. Responses to the NRC RAI #2 on TR WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," dated July 8, 2009 (ADAMS Accession No. ML091940063)

12. Letter dated February 20, 2015, from Christopher R. Costanzo, Site Vice President, St. Lucie Nuclear Plant, Florida Power and Light Company, to NRC, "Response to Requests for Additional Information (RAIs) Regarding License Amendment Request to Adopt TSTF-426, "Revise or Add Actions to Preclude Entry into LCO 3.0.3 - RITSTF Initiatives 6b and 6c" (ADAMS Accession No. ML15065A235)

Principal Contributor: Peter J. Snyder

Date: August 31, 2015

The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50-389

Enclosures:
1. Amendment No. 227 to DPR-67
2. Amendment No. 177 to NPF-16
3. Safety Evaluation

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