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

March 29, 2016

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

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATIONS FOR COMMUNICATIONS AND THE MANIPULATOR CRANE (CAC NOS. MF6423 AND MF6424)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 269 to Renewed Facility Operating License (RFOL) No. DPR-31 and Amendment No. 264 to RFOL No. DPR-41 for Turkey Point Nuclear Generating Unit Nos. 3 and 4, respectively. The amendments change the Technical Specifications (TSs) in response to the application from Florida Power & Light Company (the licensee) dated July 2, 2015, as supplemented by letter dated November 17, 2015.

The amendments remove TS 3/4.9.5, "Communications," and TS 3/4.9.6, "Manipulator Crane," from the TSs and require inclusion of those specifications in the Updated Final Safety Analysis Report and related procedures, which the licensee is required to control by the provisions set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, "Changes, tests, and experiments." The removal of TS 3/4.9.5 and TS 3/4.9.6 is consistent with the requirements in 10 CFR, Section 50.36, "Technical specifications."

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

Sincerely,

Audrey L. Klett, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 269 to DPR-31
2. Amendment No. 264 to DPR-41
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 269
Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated July 2, 2015, as supplemented by letter dated November 17, 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, the license is amended by changes to the Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-31 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 269 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report and related procedures as described in the licensee's letter dated July 2, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: March 29, 2016

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 264
Renewed License No. DPR-41

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated July 2, 2015, as supplemented by letter dated November 17, 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, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-41 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 264 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report and related procedures as described in the licensee's letter dated July 2, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: March 29, 2016

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 269 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 264 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace page 3 of Renewed Facility Operating License No. DPR-31 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace page 3 of Renewed Facility Operating License No. DPR-41 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 pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
xii	xii
3/4 9-5	3/4 9-5
3/4 9-6	3/4 9-6

E. Pursuant to the Act and 10 CFR Parts 40 and 70 to receive, possess, and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units Nos. 3 and 4.

3. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified below:

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 269 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than July 19, 2012.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. Pursuant to the Act and 10 CFR Parts 40 and 70 to receive, possess, and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units Nos. 3 and 4.

3. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified below:

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 264 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than April 10, 2013.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.9.5 DELETED

REFUELING OPERATIONS

3/4.9.6 DELETED

3/4.9.7 DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR AMENDMENT NO. 269 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31 AND AMENDMENT NO. 264 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41 FLORIDA POWER & LIGHT COMPANY TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated July 2, 2015,[1] as supplemented by letter dated November 17, 2015,[2] Florida Power & Light Company (the licensee) requested changes to the Technical Specifications (TSs) for Turkey Point Nuclear Generating Unit Nos. 3 and 4 (Turkey Point), which are contained in Appendix A of Renewed Facility Operating License Nos. DPR-31 and DPR-41. The licensee proposed to remove communications and manipulator crane requirements from the TSs and maintain them in licensee-controlled documents. The licensee requested the removal of TS 3/4.9.5, "Communications," and TS 3/4.9.6, "Manipulator Crane," from the TSs and the inclusion of these requirements in the Turkey Point Updated Final Safety Analysis Report (UFSAR) and related procedures. The licensee also requested conforming changes to the TS Index.

By electronic mail (e-mail) dated November 13, 2015,[3] the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff sent the licensee a request for additional information (RAI). The licensee responded to the RAI by letter dated November 17, 2015. The licensee's letter dated November 17, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination, which was published in the *Federal Register* (FR) on October 27, 2015 (80 FR 65813).

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML15198A153.
[2] ADAMS Accession No. ML16006A355.
[3] ADAMS Accession No. ML15320A374.

2.0 REGULATORY EVALUATION

2.1. Description of Turkey Point Refueling Operations Communications and Manipulator Crane Design and Requirements

The UFSAR states that the manipulator crane and auxiliary hoist are part of the fuel handling system. In the refueling cavity, fuel is removed from the reactor vessel, transferred through the water, and placed in the fuel transfer system by a manipulator crane. The manipulator crane, which is located inside containment, is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered from the mast to grip the fuel assembly. The upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position. The manipulator crane can lift only one fuel assembly at a time.

The UFSAR states that the suspended weight on the gripper tool is monitored by an electric load cell indicator mounted on the control console. A load in excess of 110 percent of a fuel assembly weight stops the winch drive from moving in the up direction. The gripper is interlocked through a weight sensing device and also a mechanical spring lock so that it cannot be opened when supporting a fuel assembly.

The UFSAR states that after the vessel head is removed, the rod cluster control drive shafts are removed from their respective assemblies using the auxiliary hoist on the manipulator crane and the drive shaft unlatching tool. The fuel handling manipulators and hoists are designed so that fuel can only be raised up to positions that provide adequate shield water depth for the safety of operating personnel.

The UFSAR states that direct communication between the control room and the refueling cavity manipulator crane is required whenever changes in core geometry that affect criticality are taking place. This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe conditions detected from the control board indicators during fuel movement.

Section 3/4.9, "Refueling Operations," of the Turkey Point TSs contains TS 3/4.9.5 and TS 3/4.9.6. The requirements of TS 3/4.9.5 for communication capability between the refueling personnel and the control room apply only during core alterations, which can only be conducted with the reactor head removed and the reactor coolant system depressurized. The components covered by this limiting condition for operation (LCO) include radios and associated power and transmission equipment necessary to establish and maintain communications between the control room and the refueling station.

TS 3/4.9.5, LCO 3.9.5 requires direct communications be maintained between the control room and personnel at the refueling station. When these direct communications cannot be maintained, this TS requires the licensee to suspend all core alterations. TS 3/4.9.5 also contains a surveillance requirement (SR) to demonstrate these direct communications within 1 hour prior to the start of, and in accordance with, the Surveillance Frequency Control

Program, during core alternations. The Turkey Point TS Bases, which are documented in the licensee's letter dated April 22, 2015,[4] state that the requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during core alterations.

The requirements of TS 3/4.9.6 for an operable manipulator crane and auxiliary hoist apply only during movement of drive rods or fuel assemblies within the reactor vessel, which can only take place with the reactor head removed and the reactor coolant system depressurized. This is less restrictive than applicability during core alterations. The refueling manipulator crane is used for movement of control rods and fuel assemblies within the reactor vessel and adjacent refueling cavity. The manipulator crane has adequate load capacity to lift a control rod or fuel assembly and is equipped with overload limits that ensure the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during refueling operations.

TS 3/4.9.6, LCO 3.9.6 requires the manipulator crane and auxiliary hoist be used for movement of drive rods or fuel assemblies and be operable with specified minimum capacity and overload limits. When the operability is not satisfied, this TS requires the licensee to suspend use of the crane or hoist for movement of drive rods and fuel assemblies within the reactor vessel. TS 3/4.9.6 also contains two SRs for load testing the crane and hoist. The TS Bases state that the requirements for the manipulator crane ensure that the manipulator crane will be used for movement of drive rods and fuel assemblies; that the crane has sufficient load capacity for lifting drive rods and fuel assemblies; and that the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. The TS Bases also state that the requirement that the auxiliary hoist load indicator be used to prevent lifting excessive loads requires manual action. The auxiliary hoist load indicator does not include any automatic mechanical or electrical interlocks that prevent lifting loads in excess of 600 pounds.

2.2 Licensee's Proposed Changes

The licensee requested to delete TS 3/4.9.5 and TS 3/4.9.6 from the TSs. The licensee also requested that these requirements be relocated to the UFSAR and related procedures and be controlled in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, "Changes, tests and experiments." The licensee also requested conforming changes to the TSs Index.

2.3 Regulatory Review

The NRC staff reviewed the licensee's application to ensure 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) activities proposed 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 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.

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[4] ADAMS Accession No. ML15139A080.

Section 182a of the Atomic Energy Act of 1954, as amended (the Act), requires applicants for nuclear power plant operating licenses to include the TSs as part of the license. Section 50.36, "Technical specifications," of 10 CFR contains the requirements for items that must be in the TSs, which include LCOs and SRs. Paragraph 50.36(c)(2)(ii) of 10 CFR states that a TS LCO of a nuclear reactor must be established for each item meeting one or more of the following criteria:

> Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

In the FR notice that announced the final rulemaking that added these four criteria to 10 CFR 50.36 (60 FR 36953; July 19, 1995), the NRC staff stated that the rule codifies criteria for determining the content of TSs, and that each licensee covered by these regulations may voluntarily use the criteria as a basis to propose the relocation of existing TSs that do not meet any of the criteria from the facility license to licensee-controlled documents. The staff also stated in this FR notice that related SRs and actions would be retained for each LCO that remains in the TSs.

In its letter dated May 9, 1988,[5] from the Director of the Office of Nuclear Reactor Regulation, Thomas E. Murley, to Walter S. Wilgus of the Babcock and Wilcox Owners Group, the NRC staff documented its conclusions as to which specifications must be retained in the Standard Technical Specifications (STSs) and which specifications could be relocated to other licensee-controlled documents, based on the staff's review of the Commission's Interim Policy Statement on Technical Specification Improvements, dated February 6, 1987 (52 FR 3788). The May 9, 1988, letter enclosed Appendix B, "Results of the NRC Staff Review Westinghouse Owners Group's Submittal Retention and Relocation of Specific Technical Specifications," which lists Westinghouse STS LCO 3.9.5, "Communications," and LCO 3.9.6, "Manipulator Crane," as LCOs that may be relocated to licensee-controlled documents.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 16.0, "Technical Specifications," contains guidance for

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[5] ADAMS Accession No. ML11264A057.

reviews of TSs. The NRC staff prepared STSs for each of the light-water reactor nuclear steam supply systems and associated balance-of-plant equipment systems. The staff applies this guidance in its reviews to help it determine if the content and format of proposed changes are consistent with the applicable STSs. The staff reviewed the licensee's application against the STSs in NUREG-1431, Revision 4, "Standard Technical Specifications – Westinghouse Plants," Volumes 1 and 2.[6]

Paragraph 50.34(b) of 10 CFR states that the Final Safety Analysis Report (FSAR) shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole. This regulation also states that the FSAR shall include a description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements; the bases, with technical justification therefore, upon which such requirements have been established; and the evaluations required to show that safety functions will be accomplished. Such items as fuel handling systems shall be discussed insofar as they are pertinent.

Paragraph 50.59(c)(1) of 10 CFR states that a licensee can make changes in the facility or procedures as described in the UFSAR and conduct tests or experiments not described in the UFSAR without obtaining a license amendment pursuant to 10 CFR 50.90 if none of the criteria in 10 CFR 50.59(c)(2) are met. Paragraph 50.59(c)(3) of 10 CFR states that the UFSAR is considered to include FSAR changes resulting from evaluations performed pursuant to 10 CFR 50.59 and analyses performed pursuant to 10 CFR 50.90 since submittal of the last UFSAR pursuant to 10 CFR 50.71.

NRC human factors reviews address programs, procedures, training, plant design features, and operator manual actions related to operator performance during normal and accident conditions. The NRC staff conducted a human factors evaluation to confirm that operator performance would not be adversely affected as a result of the licensee's proposed changes to the TSs. The review was based on the following regulatory guidance:

(1) NUREG-0800, Chapter 13.5.1.1, "Administrative Procedures – General," Revision 1, dated December 2011,[7] and
(2) NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, dated November 2012.[8]

Chapter 13.5.1.1 of NUREG-0800 provides guidance for reviewing changes involving administrative controls of cranes. NUREG-0711 provides guidance for reviewing procedures. The purpose of the human factors review is to ensure operators have the equipment and information necessary for safe operation and to minimize the likelihood of errors while operating the plant. Therefore, the human factors review focuses on integrated performance of operators with equipment, procedures, and administrative controls and, therefore, relies on this regulatory guidance.

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[6] ADAMS Accession Nos. ML12100A222 and ML12100A228.
[7] ADAMS Accession No. ML112730402.
[8] ADAMS Accession No. ML12324A013.

3.0 TECHNICAL EVALUATION

TS 3/4.9.5 ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during core alterations. This TS provides assurance that communication systems are available so control room staff can warn staff at the refueling station of potential safety issues during core alterations. The associated SR provides assurance that communication systems are operable prior to fuel movement and at specified intervals during fuel movement. TS 3/4.9.6 ensures that the manipulator crane will be used for movement of drive rods and fuel assemblies, the crane has sufficient load capacity for lifting drive rods and fuel assemblies, and the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. This TS prescribes weight limits and interlock settings to prevent overloading of the manipulator crane and auxiliary hoist. The associated SRs provide assurance that both systems are operable during movement of drive rods and fuel assemblies.

In its application dated July 2, 2015, the licensee requested to delete TS 3/4.9.5 and TS 3/4.9.6 from the TSs and to relocate the information contained in those TSs to the UFSAR and related procedures. The licensee stated that the proposed changes are consistent with the STSs for Westinghouse Plants. The licensee also stated that the refueling operation communications, manipulator crane, and auxiliary hoist are not equipment that satisfies the criteria of 10 CFR 50.36(c)(2)(ii) for establishment of an LCO. The NRC staff compared the proposed deletion of these TSs against the regulatory requirements of 10 CFR 50.36 and the guidance in the staff's letter dated May 9, 1988; NUREG-1431; NUREG-0711; and NUREG-0800, Chapter 13.5.1.1.

Section 50.36 of 10 CFR contains the requirements for items that must be in the TSs. Paragraph 50.36(c)(2)(ii) provides four criteria for determining when a TS LCO must be established. The NRC staff reviewed TS 3/4.9.5 and TS 3/4.9.6 against each of these four criteria to determine whether these TSs need to continue to be included in the TSs.

The NRC staff determined that the refueling operations communications, manipulator crane, and auxiliary hoist do not meet Criterion 1 of 10 CFR 50.36(c)(2)(ii). The requirements related to refueling communications and the manipulator crane and auxiliary hoist operability are included in TS 3/4.9.5 and TS 3/4.9.6, respectively. These requirements apply only when the reactor coolant pressure boundary is opened by removing the reactor vessel head to support core alterations. Therefore, the requirements proposed for removal from the TSs do not apply to instrumentation used to detect, and indicate in the control room, a significant, abnormal degradation of the reactor coolant pressure boundary.

The NRC staff determined that the refueling operations communications, manipulator crane, and auxiliary hoist do not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii). The design-basis accident (DBA) associated with core alterations and movement of material in or near the spent fuel pool is a postulated fuel handling accident (FHA). The assumptions and radiological consequences of a design-basis FHA occurring inside containment and in the fuel handling building are described in Section 14.2.1 of the Turkey Point UFSAR. The Turkey Point UFSAR states that the radiological consequences of the FHA were determined in accordance with the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis

Accidents at Nuclear Power Reactors," dated July 2000.[9] The important process variables, design features, and initial conditions for the FHA consequence analysis are the fuel design, peak fuel burn-up, decay time, number of fuel pins damaged, and water depth available to scrub the postulated release. The Turkey Point FHA analysis assumed all fuel rods in a peak power fuel assembly are breached as a result of the drop. The failure of all fuel rods is a conservative assumption that provides a bounding source term for determining the potential radiological consequences resulting from any level of damage to a single fuel assembly.

The requirements related to refueling communications and manipulator crane operability apply when handling fuel assemblies inside containment. Communications capability is not a process variable, design feature, or initial condition of an FHA. However, the FHA postulated to occur in containment may involve the manipulator crane. The limiting condition for operation associated with the manipulator crane specifies that the crane be operable with an adequate lifting capacity to handle fuel assemblies and an overload cut-off to protect the reactor vessel internals from excessive uplift force. Since the FHA assumes the manipulator crane drops a fuel assembly, the capacity of the crane is not an initial condition associated with the design-basis FHA. Similarly, operation of the overload cutoff is not an initial condition of the design-basis FHA, because the assumption that all fuel pins are damaged in a single assembly bounds the potential damage to a fuel assembly that could result from excessive uplift forces.

The assumed fuel assembly damage used in the FHA analysis is conservative and not mechanistically linked to damage that could result from specific fuel handling accident scenarios. Although the manipulator crane could be involved in events that result in damage to a fuel assembly, the physical arrangement of the fuel handling areas provides built-in protection against fuel assembly damage that could reasonably exceed that assumed in the FHA analysis. LCOs 3.9.5 and 3.9.6 are unrelated to inadvertent fuel loading accidents and do not present an initial condition for fuel drop accident scenarios. The consequences of such scenarios are bounded by those already considered in the UFSAR. Therefore, the proposed removal of the refueling communications, manipulator crane, and auxiliary hoist LCOs from the TSs is acceptable with respect to Criterion 2.

The NRC staff determined that the refueling operations communications, manipulator crane, and auxiliary hoist do not meet Criterion 3 of 10 CFR 50.36(c)(2)(ii). The equipment encompassed by the requirements proposed for removal from the TSs do not function to mitigate any DBA. Control room isolation, which is credited in Section 14.2.1 of the UFSAR to mitigate control room operator dose following an FHA within containment, is assumed to be automatically initiated by high radiation indicated on the containment radiation monitors. Therefore, the equipment used by personnel to establish and maintain communication between the control room and the refueling station within containment is not relied on to mitigate DBAs. The manipulator crane and auxiliary hoist do not actuate to mitigate a DBA or transient analysis. Therefore, the equipment addressed by the communication and manipulator crane TSs does not satisfy Criterion 3.

The NRC staff determined that the refueling operations communications, manipulator crane, and auxiliary hoist do not meet Criterion 4 of 10 CFR 50.36(c)(2)(ii). None of the equipment addressed by the communication and manipulator crane TSs has been shown to be significant

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[9] ADAMS Accession No. ML003716792.

to public health and safety through either operating experience or probabilistic risk assessments. Therefore, the equipment addressed by the communication and manipulator crane TSs does not satisfy Criterion 4.

Because the refueling operations communications, manipulator crane, and auxiliary hoist do not meet any of these criteria, the NRC staff determined that the licensee is not required to retain this LCO and its associated applicability, actions, and SRs in the TSs. This determination is also consistent with the NRC letter dated May 9, 1988, from Thomas E. Murley, to Walter S. Wilgus. In addition, NUREG-1431 provides equipment and systems that are typically required in the TSs for Westinghouse plants. Communications equipment for refueling, manipulator crane, and auxiliary hoist specifications are not listed in NUREG-1431. Therefore, the NRC staff has determined it is acceptable to delete TS 3/4.9.5 and TS 3/4.9.6 from the TSs.

In its application dated July 2, 2015, and in its supplement dated November 17, 2015, the licensee requested that TS 3/4.9.5 and TS 3/4.9.6 be relocated in their entirety to the UFSAR and related procedures. In its letter dated November 17, 2015, the licensee responded to the staff's RAI and indicated that it intends to relocate the information currently in TS 3/4.9.5 and TS 3/4.9.6 in its entirety to the UFSAR and related plant procedures. The staff determined that the licensee's response to the staff's RAI ensures that relevant information will be available to operators as a result of granting these amendments. The staff determined that implementation of the amendments will not cause changes or deletions to the information available to operators; rather, the amendments change the location of the information. The staff determined that the licensee's request does not change communication system equipment or other physical equipment used during manipulator crane or auxiliary hoist operation. The staff does not expect job tasks completed by operators to change, with the exception that the information currently found in TS 3/4.9.5 and TS 3/4.9.6 will now be found in the UFSAR and related procedures rather than in TSs. Similarly, these amendments do not change the SR intervals, thereby maintaining the current level of protection with the communications and crane systems. The NRC staff determined that relocation of these TS requirements into the UFSAR and related procedures provides reasonable assurance that the licensee will maintain adequate communications and operation of the manipulator crane and auxiliary hoist and that plant operation conforms to the design bases. The staff also determined that the proposed changes preserve the information needed to conduct refueling operations and communications, do not reduce any safety margins, and are consistent with the guidance in NUREG-0711 and Chapter 13.5.1.1 of NUREG-0800.

In its application dated July 2, 2015, the licensee stated that following NRC approval of this proposed amendment, changes to the relocated requirements will be controlled by the provisions of 10 CFR 50.59 to determine if prior NRC approval is required. Section 50.59 of 10 CFR requires the licensee to determine whether any changes to the requirements relocated in the UFSAR will need prior NRC approval by a license amendment. An NRC-approved license amendment is required if the changes to the UFSAR may result in an increase in occurrence or frequency of accidents, or a failure or malfunction of an SSC. This provides reasonable assurance that information necessary to the operators will not be inadvertently removed from the UFSAR and related procedures without required analyses. The staff determined that the proposed changes are adequate to ensure the affected equipment is adequately controlled to reliably perform its intended functions.

Based on the aforementioned considerations and conclusions, the NRC staff has determined that it is acceptable to delete TS 3/4.9.5 and TS 3/4.9.6 from the TSs and relocate such requirements to the UFSAR and related procedures to be controlled by the licensee in accordance with 10 CFR 50.59. As such, the NRC staff also concludes that the proposed changes to the TS Index that reflect deletion of these TSs are also acceptable.

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 8, 2016,[10] of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the 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, which was published in the FR on October 27, 2015 (80 FR 65813), that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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.

Principal Contributors: Brian Green
Steve Jones
Audrey Klett

Date: March 29, 2016

[10] The NRC staff notified the State official by telephone and by e-mail (ADAMS Accession No. ML16039A258).

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

Sincerely,

/RA/

Audrey L. Klett, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 269 to DPR-31
2. Amendment No. 264 to DPR-41
3. Safety Evaluation

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