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

November 6, 2012

Mr. Mano Nazar
Executive Vice President and
Chief Nuclear Officer
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: TURKEY POINT NUCLEAR GENERATING STATION UNIT NOS. 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING ADOPTION OF TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION" (TAC NOS. ME9106 AND ME9107)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 255 to Renewed Facility Operating License No. DPR-31 and Amendment No. 251 to Renewed Facility Operating License No. DPR-41 for the Turkey Point Nuclear Generating Station, Unit Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 16, 2012, as supplemented by letter dated August 10, 2012.

The amendments revise TS 3/4.4.5, "Steam Generator (SG) Tube Integrity," TS 6.8.4.j, "Steam Generator (SG) Program," and TS 6.9.1.8, "Steam Generator Tube Inspection Report," and include TS Bases changes that summarize and clarify the purpose of the TS in accordance with TS Task Force Traveler (TSTF)-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

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. 50-250 and 50-251

Enclosures:
1. Amendment No. 255 to DPR-31
2. Amendment No. 251 to DPR-41
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. 50-250

TURKEY POINT NUCLEAR GENERATING STATION UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 255
Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated July 16, 2012, as supplemented by letter dated August 10, 2012, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the 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. 255 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 7 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: November 6, 2012

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING STATION UNIT NO. 4

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 251
Renewed License No. DPR-41

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated July 16, 2012, as supplemented by letter dated August 10, 2012, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the 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.251 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 7 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: November 6, 2012

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 255 RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 251 RENEWED FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace Page 3 of Renewed Operating License DPR-31 with the attached Page 3.

Replace Page 3 of Renewed Operating License DPR-41 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 numbers and contain marginal lines indicating the areas of changes.

Remove pages	Insert pages
3/4 4-11	3/4 4-11
6-18	6-18
6-18a	6-18a
6-18b	6-18b
6-18c	6-18c
6-22a	6-22a

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. 255 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. <u>Maximum Power Level</u>

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

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 251 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. <u>Final Safety Analysis Report</u>

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.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the SG Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION*:

a. With one or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program;

1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

b. With the requirements and associated allowable outage time of Action a above not met or SG tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection.

*Separate Action entry is allowed for each SG tube.

- The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than 0.60 L_a, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.

- The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than 0.6 L_a, at all times when containment integrity is required.

3) Overall air lock leakage acceptance criteria is $\leq$ 0.05 L_a, when pressurized to P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.

i. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. Change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.8.4 i.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

j. Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:

1. Tubes with service-induced flaws located greater than 18.11 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.11 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The portion of the tube below 18.11 inches from the top of the tubesheet is excluded from inspection. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period beings at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;

   b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

   c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

3. If crack indications are found in any portion of a SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-secondary leakage.

k. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the CREVS, operating at the flow rate required by Surveillance Requirement 4.7.5.d, at a Frequency of 18 months. Additionally, the supply fans (trains A and B) will be tested on a staggered test basis (defined in Technical Specification definition 1.29 every 36 months). The results shall be trended and the CRE boundary assessed every 18 months.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of Specification 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

6.8.5 DELETED

STEAM GENERATOR TUBE INSPECTION REPORT (Cont'd)

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

i. The calculated accident induced leakage rate from the portion of the tubes below 18.11 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 255 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-31 AND

AMENDMENT NO. 251 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING STATION UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated July 16, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12215A011), as supplemented by letter dated August 10, 2012 (ML12242A346), Florida Power and Light (the licensee) proposed an amendment to the Technical Specifications (TSs) for Turkey Point Nuclear Generating Station, Unit Nos. 3 and 4. The proposed amendments would adopt U.S. Nuclear Regulatory Commission (NRC)-approved Revision 2 to Technical Specifications Task Force (TSTF) Standard Technical Specifications (STSs) Change Traveler TSTF-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" (ADAMS No. ML110610350). The proposed changes revise TS 3/4.4.5, "Steam Generator (SG) Tube Integrity," TS 6.8.4.j, "Steam Generator (SG) Program," and TS 6.9.1.8, "Steam Generator Tube Inspection Report," and include TS Bases changes that summarize and clarify the purpose of the TSs. The plant specific changes concern SG inspection periods, and address applicable administrative changes and clarifications.

The licensee stated that the license amendment request (LAR) is consistent with the Notice of Availability of TSTF-510, Revision 2, announced in the *Federal Register* on October 27, 2011 (76 FR 66763) as part of the consolidated line item improvement process.

The current STS requirements in the above specifications were established in May 2005 with the NRC staff's approval of TSTF-449, Revision 4, "Steam Generator Tube Integrity" (NRC *Federal Register* Notice of Availability (70 FR 24126)). The TSTF-449 changes to the STS incorporated a new, largely performance-based approach for ensuring the integrity of the SG tubes is maintained. The performance-based requirements were supplemented by prescriptive requirements relating to tube inspections and tube repair limits to ensure that conditions adverse to quality are detected and corrected on a timely basis. As of September 2007, the TSTF-449, Revision 4, changes were adopted in the plant TSs for all pressurized-water reactors (PWRs)

The proposed changes in TSTF-510, Revision 2, reflect licensees' early implementation experience with respect to the TSTF-449, Revision 4. TSTF-510 characterizes the changes as editorial corrections, changes, and clarifications intended to improve internal consistency, consistency with implementing industry documents, and usability without changing the intent of the requirements. The licensee stated that the proposed changes are an improvement to the existing SG inspection requirements and continue to provide assurance that the plant licensing basis will be maintained between SG inspections.

The NRC staff's original proposed no significant hazards consideration determination was published in the *Federal Register* on September 4, 2012 (77 FR 53929).

2.0 REGULATORY EVALUATION

The SG tubes in PWRs have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage during severe accidents.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the requirements with respect to the integrity of the SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB shall have "an extremely low probability of abnormal leakage...and of gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32).

The GDC included in Appendix A to 10 CFR Part 50 did not become effective until February 20, 1971. During the initial plant licensing of Turkey Point, it was demonstrated that the design of the RCPB met the regulatory requirements in place at that time, the Atomic Energy Commission's GDC proposed in 1967. Turkey Point's Updated Final Safety Analysis Report Section 4.1.3, "Principal Design Criteria," discusses the following design bases for the RCPB:

> Reactor Coolant Pressure Boundary
>
> Criterion: The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. (1967 Proposed GDC 9)
>
> Monitoring Reactor Coolant Leakage
>
> Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (1967 Proposed GDC 16)

Reactor Coolant Pressure Boundary Capability

Criterion: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic load imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition. (1967 Proposed GDC 33)

Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes. (1967 Proposed GDC 34)

Reactor Coolant Pressure Boundary Surveillance

Criterion: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided. (1967 Proposed GDC 36)

To this end, 10 CFR 50.55a specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Section 50.55a further requires, in part, that throughout the service life of a PWR facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection [(ISI)] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code.

The regulation at 10 CFR 50.36, "Technical specifications," establishes the requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. LCOs and accompanying action statements and SRs in the STSs relevant to SG tube integrity are in Specification 3.4.13, "RCS [reactor coolant system] Operational Leakage," and Specification 3.4.20, "Steam Generator (SG) Tube Integrity." The SRs in the "Steam Generator (SG) Tube Integrity" specification reference the SG Program, which is defined in the STS administrative controls.

The 10 CFR 50.36(c)(5) regulation defines administrative controls as "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." Programs established by the licensee to operate the facility in a safe manner, including the SG Program, are listed in the administrative controls section of the TSs. The SG Program is defined in Specification 6.8.4.j, while the reporting requirements relating to implementation of the SG Program are in Specification 6.9.1.8.

Specification 6.8.4.j requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. SG tube integrity is maintained by meeting the performance criteria specified in TS 6.8.4.j.b for structural and leakage integrity, consistent with the plant design and licensing basis. Specification 6.8.4.j.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. Specification 6.8.4.j.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that (1) may be present along the length of a tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet and (2) may satisfy the applicable tube repair criteria. The applicable tube repair criteria, specified in TS 6.8.4.j.c, are that tubes found during ISI to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness shall be plugged, unless the tubes are permitted to remain in service through application of the alternate repair criteria provided in TS 6.8.4.j.c.1.

3.0 TECHNICAL EVALUATION

Each proposed change to the TSs is described individually below, followed by the NRC staff's assessment of the change.

3.1 Specification 6.8.4.j, "Steam Generator (SG) Program"

Proposed Change: The last sentence of the introductory paragraph currently states: "In addition, the Steam Generator Program shall include the following provisions:" The change would delete the word "provisions" such that the sentence would state: "In addition, the Steam Generator Program shall include the following:" The basis for this change is that subsequent paragraphs in Specification 6.8.4.j start with "Provisions for ..." and the word "provisions" in the introductory paragraph is duplicative.

Assessment: The NRC staff has reviewed Specification 6.8.4.j and agrees that the word, "provisions," in the introductory paragraph is duplicative. The NRC staff agrees that the change is administrative in nature, and therefore is acceptable.

3.2 Paragraph 6.8.4.j.b.1, "Structural integrity performance criterion"

The first sentence currently states:

> Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents.

Proposed Change: Revise the sentence as follows:

> Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents.

The basis for the change is that this sentence inappropriately includes anticipated transients in the description of normal operating conditions.

Assessment: The NRC staff agrees the current wording is incorrect and that anticipated transients should be differentiated from normal operating conditions. Therefore, the NRC staff finds the change acceptable.

3.3. Paragraph 6.8.4.j.c, "Provisions for SG tube repair criteria," Paragraph 6.8.4.j.d, "Provisions for SG tube inspections," LCO 3.4.5, "Steam Generator (SG) Tube Integrity," SR 4.4.5.2, "Steam Generator (SG) Tube Integrity"

Proposed Change: Change all references to "tube repair criteria" to "tube plugging criteria." This change is intended to be consistent with the treatment of SG tube repair throughout Specification 6.8.4.j.

Assessment: The NRC staff finds that the proposed change provides a more accurate label of the criteria and, therefore, adds clarity to the specification. This is because one of two actions must be taken when the criteria are exceeded. One action is to remove the tube from service by plugging the tube at both tube ends. The alternative action is to repair the tube, but only if such a repair is permitted by paragraph 6.8.4.j.c. Therefore, the NRC staff finds the change acceptable.

3.4 Paragraph 6.8.4.j.d, "Provisions for SG tube inspections"

Proposed Change: Change the term "an assessment of degradation" to "a degradation assessment" to be consistent with the terminology used in industry program documents.

Assessment: The NRC staff agrees that the terminology should be consistent and finds the change acceptable.

3.5 Paragraph 6.8.4.j.d.1

Proposed change: The paragraph currently states: "Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement." The change would replace "SG replacement" with "SG installation." The basis for the change is that it will allow the SG Program to apply to both existing plants and new plants.

Assessment: The NRC staff agrees the SG Program can apply to both existing and new plants. Therefore, the NRC staff finds the change acceptable.

3.6 Paragraph 6.8.4.j.d.2 for plants with SGs with alloy 600 thermally treated (TT) tubes

The paragraph currently states:

> Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

Proposed Change: Revise paragraph 6.8.4.j.d.2 as follows:

> After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.
>
> a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;

b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

Assessment: Paragraph 6.8.4.j.d.2 in its current form and with the proposed changes is similar for each of the tube alloy types, but with differences that reflect the improved resistance of alloy 600 TT to stress corrosion cracking relative to alloy 600 mill annealed (MA) and the improved resistance of alloy 690 TT relative to both alloy 600 MA and alloy 600 TT. These differences include progressively larger maximum inspection interval requirements and sequential inspection periods (during which 100 percent of the tubes must be inspected) for alloy 600 MA, 600 TT, and alloy 690 TT tubes, respectively. In addition, because of the longer maximum inspection intervals allowed for alloy 600 TT and 690 TT tubes, paragraph 6.8.4.j.d.2 includes a restriction on the distribution of sampling over each sequential inspection period for alloy 600 TT and 690 TT tubes that is not included for alloy 600 MA tubes.

The licensee proposes to move the first two sentences of paragraph 6.8.4.j.d.2 to the end of the paragraph and make editorial changes to improve clarity. The NRC staff finds these changes to be of a clarifying nature, not changing the current intent of these two sentences. However, the LAR also includes two changes to when inspections are performed as follows:

- The second inspection period would be revised from 90 to 96 effective full power months (EFPMs).
- The third and subsequent inspection periods would be revised from 60 to 72 EFPMs.

The licensee characterizes these changes as marginal increases for consistency with typical fuel cycle lengths that better accommodate the scheduling of inspections. The NRC staff notes that plants with alloy 600 TT SG tubes typically inspect at 18- or 36-month intervals (one or two fuel cycles, respectively) depending on whether stress corrosion crack activity was observed during the most recent inspection. With these intervals, the last scheduled inspection during the first inspection period would occur at 108 months after the first refueling outage following SG installation. This is 12 months before the end of the first 120-EFPM inspection period. However, with the proposed changes to the length of the second and subsequent inspection periods, the NRC staff finds that the last scheduled inspections in the second and subsequent inspection periods will coincide exactly with the end of these periods.

The proposed changes would generally increase the number of inspections in each of the second and subsequent inspection periods by up to one additional inspection. This could reduce the required average minimum sample size during these periods. However, inspection sample sizes will continue to be subject to paragraph 6.8.4.j.d, which states that in addition to meeting the requirements of paragraphs 6.8.4.j.d.1, d.2, and d.3, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure SG tube integrity is maintained until the next scheduled inspection. Therefore, the NRC staff concludes that with the proposed changes to the length of the second and subsequent inspection periods, compliance with the SG program requirements in Specification 6.8.4.j will continue to ensure both adequate inspection scopes and tube integrity.

For each inspection period, paragraph 6.8.4.j.d.2 currently requires that at least 50 percent of the tubes be inspected by the refueling outage nearest to the mid-point of the inspection period and the remaining 50 percent by the refueling outage nearest the end of the inspection period. The NRC staff notes that if there are not an equal number of inspections in the first half and second half of the inspection period, the average minimum sampling requirement may be markedly different for inspections in the first half of the inspection period compared to those in the second half, even when there are uniform intervals between each inspection. For example, a plant in the first (120 EFPM) inspection period with a scheduled 36-month interval (two fuel cycles) between each inspection would currently be required to inspect 50 percent of the tubes by the refueling outage nearest the midpoint of the inspection, which would be the third refueling outage in the period, six months before the mid-point. However, since no inspection is scheduled for that outage, then the full 50 percent sample must be performed during the inspection scheduled for the second refueling outage in the period. Two inspections would be scheduled to occur in the second half of the inspection period, at 72 and 108 months into the inspection period. Thus, the current sampling requirement could be satisfied by performing a 25 percent sample during each of these inspections or other combinations of sampling (e.g., 10 percent during one and 40 percent in the other) totaling 50 percent. The NRC staff finds there is no basis to require the minimum initial sample size to vary so much from inspection to inspection. The licensee proposes to revise this requirement such that the minimum sample size for a given inspection in a given inspection period is 100 percent divided by the number of scheduled inspections during that inspection period. For the above example, the proposed change would result in a uniform initial minimum sample size of 33.3 percent for each of the three scheduled inspections during the inspection period. The NRC staff concludes this proposed revision to be an improvement to the existing requirement since it provides a more consistent minimum initial sampling requirement.

The proposed changes to paragraph 6.8.4.j.d.2 include two new sentences addressing the prorating of required tube sample sizes if a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria. For example, new information from another similar plant becomes available indicating the potential for circumferential cracking at a specific location on the tube. Previous degradation assessments had not identified the potential for this type of degradation at this location. Thus, previous inspections of this location had not been performed with a technique capable of detecting circumferential cracks. However, now that the potential for circumferential cracking has been identified at this location, paragraph 6.8.4.j.d requires a method of inspection to be performed with the objective of detecting circumferential cracks that may be present at this location and that may satisfy the applicable tube plugging criteria. Suppose this inspection is performed for the first time during the third of four SG inspections scheduled for one of the inspection periods. Paragraph 6.8.4.j.d.2 currently does not specify whether this location needs to be 100 percent inspected by the end of the inspection period, or whether a prorated approach may be taken. The NRC staff addressed this question in Issue 1 of NRC Regulatory Information Summary (RIS) 2009-04, "Steam Generator Tube Inspection Requirements," dated April 3, 2009 (ADAMS No. ML083470557), as follows:

> Issue 1: *A licensee may identify a new potential degradation mechanism after the first inspection in a sequential period. If this occurs, what are the expectations concerning the scope of examinations for this new potential degradation mechanism for the*

> *remainder of the period (e.g., do 100 percent of the tubes have to be inspected by the end of the period or can the sample be prorated for the remaining part of the period)?*
>
> [NRC Staff Position:] The TS contain requirements that are a mixture of prescriptive and performance-based elements. Paragraph "d" of these requirements indicates that the inspection scope, inspection methods, and inspection intervals shall be sufficient to ensure that SG tube integrity is maintained until the next SG inspection. Paragraph "d" is a performance-based element because it describes the goal of the inspections but does not specify how to achieve the goal. However, paragraph "d.2" is a prescriptive element because it specifies that the licensee must inspect 100 percent of the tubes at specified periods.
>
> If an assessment of degradation performed after the first inspection in a sequential period results in a licensee concluding that a new degradation mechanism (not anticipated during the prior inspections in that period) may potentially occur, the scope of inspections in the remaining portion of the period should be sufficient to ensure SG tube integrity for the period between inspections.
>
> In addition, to satisfy the prescriptive requirements of paragraph "d.2" that the licensee must inspect 100 percent of the tubes within a specified period, a prorated sample for the remaining portion of the period is appropriate for this potentially new degradation mechanism. This prorated sample should be such that if the licensee had implemented it at the beginning of the period, the TS requirement for the 100 percent inspection in the entire period (for this degradation mechanism) would have been met. A prorated sample is appropriate because (1) the licensee would have performed the prior inspections in this sequential period consistently with the requirements, and (2) the scope of inspections must be sufficient to ensure that the licensee maintains SG tube integrity for the period between inspections.

The NRC staff finds that proposed Sentences 3 and 4 clarify the existing requirement consistent with the NRC staff's position from RIS 2009-04 quoted above and are, therefore, acceptable.

The proposed fifth sentence in paragraph 6.8.4.j.d.2 states, "Each inspection period defined below may be extended up to 3 EFPMs to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage." Allowing extension of the inspection periods by up to an additional 3 EFPMs potentially impacts the average tube inspection sample size to be implemented during a given inspection in that period. For example, if three SG inspections are scheduled to occur within the nominal 60-EFPM period, the minimum sample size for each of the three inspections could average as little as 33.3 percent of the tube population. If a fourth inspection can be included within the period by extending the period by 3 EFPMs, then the minimum sample size for each of the four inspections could average as little as 25 percent of the tube population. Since the subsequent period begins at the end of the included SG inspection outage, the proposed change does not impact the required frequency of SG inspection.

Required tube inspection sample sizes are also subject to the performance-based requirement in paragraph 6.8.4.j.d, which states, in part, that in addition to meeting the requirements of paragraph 6.8.4.j.d.1, d.2, and d.3, "the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection." This requirement remains unchanged under the proposal. The NRC staff concludes the proposed fifth sentence, by allowing the potential for smaller sample sizes, involves only a relatively minor relaxation to the existing sampling requirements in paragraph 6.8.4.j.d.2. However, the performance-based requirements in 6.8.4.j.d ensure that adequate inspection sampling will be performed to ensure tube integrity is maintained. Thus, the NRC staff concludes that the proposed change is acceptable.

Finally, the first sentence of the proposed revision to paragraph 6.8.4.j.d.2 replaces the last sentence of the current paragraph 6.8.4.j.d.2. This sentence establishes the minimum allowable SG inspection frequency as at least every 48 EFPMs or at least every other refueling outage (whichever results in more frequent inspections). This minimum inspection frequency is unchanged from the current sentence. The NRC staff finds that the wording changes in the sentence are of an editorial and clarifying nature and are not material, such that the current intent of the requirement is unchanged. Thus, the NRC staff concludes the first sentence of proposed paragraph 6.8.4.j.d.2 is acceptable.

3.7 Paragraph 6.8.4.j.d.3 (for plants with SG tubing fabricated from alloy 600 TT)

The first sentence of paragraph 6.8.4.j.d.3 currently states:

> If crack indications are found in any portion of a SG tube not excluded above, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

Proposed Change: Revise this sentence as follows:

> If crack indications are found in any portion of a SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections).

The proposed change is replacing the words "for each SG" with the words "for each affected and potentially affected SG." The licensee states that the existing wording can be misinterpreted. The licensee further states that the intention is that those SGs that are affected and those SGs that are potentially affected must be inspected for the degradation mechanism that caused the crack indication. However, some licensees have questioned whether the current reference to "each SG" requires only the SGs that are affected to be inspected for the degradation mechanism. The proposed revision is intended to clarify the intent of the requirement.

Assessment: Paragraph 6.8.4.j.d.2 permits SG inspection intervals to extend over multiple fuel cycles for SGs with alloy 600 TT tubing, assuming that such intervals can be implemented while ensuring tube integrity is maintained in accordance with paragraph 6.8.4.j.d. However, stress corrosion cracks may not become detectable by inspection until the crack depth approaches the tube repair limit. In addition, stress corrosion cracks may exhibit high growth rates. For these reasons, once cracks have been found in any SG tube, paragraph 6.8.4.j.d.3 restricts the allowable interval to the next scheduled inspection to 24 EFPMs or one refueling outage (whichever is less). The intent of this requirement is that it applies to the affected SG and to any other SG that may be potentially affected by the degradation mechanism that caused the known crack(s). For example, a root cause analysis in response to the initial finding of one or more cracks might reveal that the crack(s) are associated with a manufacturing anomaly that causes locally high residual stress, which in turn caused the early initiation of cracks at the affected locations. If it can be established that the extent of condition of the manufacturing anomaly applies only to one SG and not the others, then the NRC staff agrees that only the affected SG needs to be inspected within 24 EFPMs or one refueling cycle in accordance with paragraph 6.8.4.j.d.2. The next scheduled inspections of the other SGs will continue to be subject to all other provisions of paragraph 6.8.4.j.d. The NRC staff finds the proposed change to paragraph 6.8.4.j.d.3 acceptable.

3.8 Specification 6.9.1.8, "Steam Generator Tube Inspection Report"

This specification lists items a. through k. to be included in a report which shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.8.4.j, "Steam Generator (SG) Program."

Proposed Change: Item b. currently reads: "Active degradation mechanisms found..." to be revised to read: "Degradation mechanisms found..."

Item e. currently reads: "Number of tubes plugged during the inspection outage for each active degradation mechanism..." to be revised to read: "Number of tubes plugged during the inspection outage for each degradation mechanism..."

Item f. currently reads, "Total number and percentage of tubes plugged to date..." to be revised to read: "The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator..."

Assessment: This proposal would delete the word "Active" in items b. and e. above. Thus, all degradation mechanisms found, whether deemed to be active or not, would now be reportable. The NRC staff finds the proposed change acceptable.

Other changes proposed in this section consist of combining items f. and h., and renumbering items i., j., and k. to h., i., and j., respectively. These are editorial changes that do not materially change the reporting requirements. The NRC staff finds these changes acceptable.

4.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component 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 (77 FR 53929). 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 the amendments.

6.0 CONCLUSION

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

Principal Contributors: Khadijah Hemphill
Kristy Bucholtz

Date: November 6, 2012

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Farideh E. Saba, Senior 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. 255 to DPR-31
2. Amendment No. 251 to DPR-41
3. Safety Evaluation

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