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

November 25, 2015

Mr. Eric McCartney
Site Vice President
NextEra Energy Point Beach, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS FOR THE STEAM GENERATOR TECHNICAL SPECIFICATIONS, TO REFLECT ADOPTION OF TSTF-510 RE: (TAC NOS. MF6043 AND MF6044)

Dear Mr. McCartney:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 254 and 258 to Renewed Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant (Point Beach), Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated March 27, 2015.

These amendments incorporate the guidance of Technical Specification Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." The guidance of TSTF-510 revises TS 3.4.17, "Steam Generator (SG) Tube Integrity," TS 5.5.8, "Steam Generator (SG) Program," and TS 5.6.8, "Steam Generator Tube Inspection Report."

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 254 to DPR-24
2. Amendment No. 258 to DPR-27
3. Safety Evaluation

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

NEXTERA ENERGY POINT BEACH, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 254
License No. DPR-24

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

   A. The application for amendment by NextEra Energy Point Beach, LLC (the licensee), dated March 27, 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 Technical Specifications as indicated in the attachment to this license amendment, and paragraph 4.B of the Renewed Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 254, are hereby incorporated in the renewed operating license. NextEra Energy Point Beach shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of issuance: November 25, 2015

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

NEXTERA ENERGY POINT BEACH, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 258
License No. DPR-27

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

   A. The application for amendment by NextEra Energy Point Beach, LLC (the licensee), dated March 27, 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 Technical Specifications as indicated in the attachment to this license amendment, and paragraph 4.B of the Renewed Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 258, are hereby incorporated in the renewed operating license. NextEra Point Beach shall operate the facility in accordance with Technical Specifications.

Enclosure 2

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of issuance: November 25, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 254

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-24

AND LICENSE AMENDMENT NO. 258

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Replace the following pages of Renewed Facility Operating License Nos. DPR-24 and DPR-27, and Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
3.4.17-1	3.4.17-1
3.4.71-2	3.4.71-2
5.5-7	5.5-7
5.5-8	5.5-8
5.5-8a	5.5-8a
5.5-9	5.5-9
5.6-6	5.6-6
5.6-7	5.6-7

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NextEra Energy Point Beach to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, NextEra Energy Point Beach to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

4. 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 Levels

NextEra Energy Point Beach is authorized to operate the facility at reactor core power levels not in excess of 1800 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 254, are hereby incorporated in the renewed operating license. NextEra Energy Point Beach shall operate the facility in accordance with Technical Specifications.

C. Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NextEra Energy Point Beach to receive, possess and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed source for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NextEra Energy Point Beach to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, NextEra Energy Point Beach to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

4. 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 Levels

NextEra Energy Point Beach is authorized to operate the facility at reactor core power levels not in excess of 1800 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 258 are hereby incorporated in the renewed operating license. NextEra Energy Point Beach shall operate the facility in accordance with Technical Specifications.

C. Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee=s application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

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

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.8 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 cool down), and 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 a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

for all SGs and leakage rate for an individual SG. Leakage is not to exceed 500 gallons per day per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS 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.

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 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 tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what location.

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

2. i. Unit 1 (alloy 600 Thermally Treated tubes): 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

5.5 Programs and Manuals

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.

ii. Unit 2 (alloy 690 Thermally Treated tubes): After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third 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, c and d 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

5.5 Programs and Manuals

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 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

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

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

3. If crack indications are found in any SG tube, 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 to secondary LEAKAGE.

5.6 Reporting Requirements

5.6.6 PAM Report

When a report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

Any abnormal degradation of the containment structure identified during the engineering evaluation of abnormal conditions shall be reported to the Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, 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, 5.6 Reporting Requirements

5.6.8 Steam Generator Tube Inspection Report (continued)

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

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

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

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 254 and 258

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

NEXTERA ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated March 27, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15086A378), NextEra Energy Point Beach, LLC (the licensee) submitted a license amendment request (LAR) to revise the Point Beach Nuclear Plant (PBNP), Units 1 and 2 technical specifications (TSs). The amendment proposes to incorporate the guidance of Technical Specification Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." The guidance of TSTF-510 revises TS 3.4.17, "Steam Generator (SG) Tube Integrity," TS 5.5.8, "Steam Generator (SG) Program," and TS 5.6.8, "Steam Generator Tube Inspection Report."

The licensee stated that this LAR is consistent with TSTF-510, Revision 2, which was noticed as being available for use as part of the consolidated line item improvement process in the *Federal Register* (FR) on October 27, 2011 (76 FR 66763). Because the LAR includes TS revisions for both implementation of TSTF-510 and removal of unnecessary information related to the original PBNP, Unit 1 SGs, this LAR is not being processed under the consolidated line item improvement process.

1.1 SG TS Background

The SG tubes of a pressurized water reactor (PWR) have a number of important safety functions. They 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 to transfer heat from the primary to the secondary system. In addition, SG tube integrity is relied upon to prevent uncontrolled fission product release under conditions resulting from core damage during severe accidents.

The current SG TS requirements for PBNP were based on TSTF-449, Revision 4, "Steam Generator Tube Integrity" (U.S. Nuclear Regulatory Commission (NRC) *FR* Notice of Availability (70 FR 24126)). The TSTF-449 changes to the Standard Technical Specifications (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 TS for all PWRs. The changes included in TSTF-510 addressed implementation issues associated with the inspection periods, and addressed other administrative changes and clarifications.

2.0 REGULATORY EVALUATION

The regulation at 10 CFR establishes the requirements with respect to the integrity of 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 gross rupture" (GDC 14),
- "shall be designed with sufficient margin" (GDC 15 and 31),
- shall be of "the highest quality standards possible" (GDC 30), and
- shall be designed to permit "periodic inspection and testing...to assess...structural and leak tight integrity" (GDC 32).

Point Beach Nuclear Plant was licensed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. As such, PBNP is not licensed to the GDC in Appendix A. The PBNP Final Safety Analysis Report (FSAR) lists the plant-specific GDC to which the plant was licensed. The PBNP GDC addressing the RCPB are PBNP GDC 9 "Reactor Coolant Pressure Boundary," GDC 33 "Reactor Coolant Pressure Boundary Capability," GDC 34 "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention", and GDC 36 "Reactor Coolant Pressure Boundary Surveillance." The applicable criteria for this system are discussed in FSAR Section 4.1 "Reactor Coolant System – Design Basis." Point Beach Nuclear Plant GDC 9, 33, 34, and 36 are similar to Appendix A GDC 14, 15, 31, and 32.

As specified by 10 CFR 50.55a(c), 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 Boiler and Pressure Vessel Code (ASME Code). Section 50.55a(g) further requires that components and supports, which are classified as ASME Code Class 1, must be designed and provided with access to enable performance of inservice examination of these components to the extent practical, and must meet the pre-service examination requirements set forth in the editions and addenda of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code incorporated by reference in 10 CFR 50.55a(b) that were applied to the construction of the particular component. Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional SG tube surveillance requirements in TSs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture and main steam line break. These analyses consider the primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 100.11 guidelines for offsite doses (or 10 CFR 50.67, as appropriate), GDC-19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis.

The regulation at 10 CFR 50.36, "Technical specifications," establishes the requirements related to the content of the TS. 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.

The LCOs (and accompanying action statements) and the SRs in the STS that are relevant to SG tube integrity are in Specification 3.4.13 "Reactor Coolant System Operational Leakage," and Specification 3.4.20 (SR 3.4.20.2), "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 PBNP, Units 1 and 2, TS 3.4.13 and 3.4.17 respectively, address requirements similar to those specified in STS sections above.

The regulation at 10 CFR 50.36(c)(5) 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 TS. The SG Program is defined in TS 5.5.8, while the reporting requirements relating to implementation of the SG Program are in TS 5.6.8.

Technical Specification 5.5.8, "Steam Generator (SG) Program" for PBNP, Units 1 and 2, requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Tube integrity is maintained by meeting specified performance criteria for structural and leakage integrity consistent with the plant design and licensing bases. Technical Specification 5.5.8.a requires a condition monitoring assessment be performed during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met. Technical Specification 5.5.8.d includes provisions regarding the scope, frequency, and methods of SG tube inspections.

3.0 TECHNICAL EVALUATION

3.1 TSTF-510 Technical Evaluation

The changes in TSTF-510, Revision 2, reflect licensees' early implementation experience with their current TSs. The changes in TSTF-510, Revision 2, are 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 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 approved TSTF-510, Revision 2, for use with the consolidated line item process on October 19, 2011 (ADAMS Accession No. ML112101604). The staff's model safety evaluation is available under ADAMS Accession No. ML112101513. Other than the variations or deviations discussed below, the licensee is not proposing any variations or deviations from the TS changes described in the TSTF-510, Revision 2. As a result, the staff's evaluation is focused on these differences, since the other changes were previously evaluated in the model safety evaluation (ADAMS Accession No. ML112101513).

The PBNP, Units 1 and 2 TSs utilize different numbering and titles than the STS on which TSTF-510, Revision 2, was based.

> For PBNP, Unit 1 and Unit 2, the "Steam Generator (SG) Program" in the TS is numbered 5.5.8 rather than 5.5.9, the "Steam Generator Tube Integrity" TS is numbered 3.4.17 rather than 3.4.20, and the "Steam Generator Tube Inspection Report" is numbered 5.6.8 rather than 5.6.9.

These differences are administrative and do not affect the applicability of TSTF-510, Revision 2, to the PBNP Units 1 and 2 TSs. As a result, the NRC staff finds the differences between what was approved for TSTF-510, Revision 2, and what is being proposed acceptable.

3.2 <u>Interim Alternate Repair Criteria Technical Evaluation</u>

The licensee also proposed removing wording from TS 5.5.8, "Steam Generator (SG) Program" and TS 5.6.8, "Steam Generator (SG) Tube Integrity." The wording is associated with inspections and implementation of an alternate repair criteria (ARC) for the replacement SGs in Unit 1, which were only approved for refueling outage 31 (RFO31) and the subsequent operating cycle (fall 2008 to spring 2010). The proposed changes would revise the following:

1. Removal of most of 5.5.8.c from "The following alternate tube repair criteria..." to the end of TS 5.5.8.c "...which is not expanded the full length of the tubesheet."

2. Removal of TS 5.6.8.i, 5.6.8.j, and 5.6.8.k in their entirety.

The ARC for PBNP Unit 1 was issued on October 7, 2008 (ADAMS Accession No. ML082540883). The NRC staff approved the ARC to be "applicable only to RFO31 and the subsequent operating cycle." Four refueling outages have transpired since RFO31: RFO32, RFO33, RFO34, and RFO35, which occurred in spring 2010, fall 2011, spring 2013, and fall 2014, respectively. Therefore, the ARC is no longer applicable to the PBNP, Unit 1 SGs. Since the ARC expired in RFO32 (spring 2010), removal of the ARC wording from the TS is an administrative effort that the staff finds acceptable.

The NRC staff also notes that TS 5.5.8.b.1 contains an extra "and." The revised TS provided as Attachment 3 to the licensee's amendment request, currently states:

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 cool down), <u>and</u> all anticipated transients included in the design specification, and design basis accidents.

The "and", indicated above with an underline, is not contained in TSTF-510 but does not change the meaning and therefore is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change a surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (80 FR 32627). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

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 Contributor: A. Johnson, NRR/DE/ESGB

Date: November 25, 2015

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Mahesh L. Chawla, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 254 to DPR-24
2. Amendment No. 258 to DPR-27
3. Safety Evaluation

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ADAMS Accession No.: ML15293A457 ***via memorandum**

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