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

January 9, 2015

Mr. Benjamin C. Waldrep
Site Vice President
Shearon Harris Nuclear Power Plant
Duke Energy
5413 Shearon Harris Road
New Hill, NC 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT ON ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE 510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION" (TAC NO. MF4112)

Dear Mr. Waldrep:

The Nuclear Regulatory Commission (NRC) has issued Amendment No. 145 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). This amendment changes the HNP Technical Specifications (TSs) in response to your application dated April 24, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14114A743).

The amendment modifies Shearon Harris TS 6.8.4.l, "Steam Generator Program," TS 6.9.1.7, "Steam Generator Tube Inspection Report," and TS 3/4.4.5, "Steam Generator Tube Integrity." The amendment addresses implementation issues associated with the steam generator inspection periods, and addresses applicable administrative changes and certain clarifications. The changes are in accordance with NRC approved Technical Specifications Task Force (TSTF), Revision 2, Standard Technical Specifications (STSs) Change Traveler TSTF-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" (ADAMS Accession No. ML110610350).

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

Sincerely,

/RA/

Martha Barillas, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 145 to NPF-63
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

DUKE ENERGY PROGRESS, INC.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 145
License No. NPF-63

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

   A. The application for amendment by Duke Energy Progress, Inc. (the licensee), dated April 24, 2014, 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.

Enclosure 1

2. Accordingly, the license is amended by changes to the TS, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 145, are hereby incorporated into this license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Lisa M. Regner, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License No. NPF-63
and the Technical Specifications

Date of Issuance: January 9, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 145

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page of the renewed facility operating license with the revised page. The revised page is identified by amendment number and contains a line in the margin indicating the area of change.

Remove	Insert
Page 4	Page 4

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
TSs	TSs
3/4 4-13	3/4 4-13
6-19d	6-19d
6-19e	6-19e
6-19f	6-19f
6-24c	6-24c
6-24d	6-24d

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I 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 or incorporated below.

(1) Maximum Power Level

Duke Energy Progress, Inc. is authorized to operate the facility at reactor core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 145 , are hereby incorporated into this license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, Inc. shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

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[1] The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

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

   AND

b. With the requirements and associated allowed 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 steam generator tube integrity in accordance with the Steam Generator Program.

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

* Separate ACTION entry is allowed for each SG tube.

PROCEDURES AND PROGRAMS (Continued)

l. Steam Generator (SG) Program

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

1. 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 a 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.

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

   a) Structural integrity performance criterion. All inservice 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), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 (3deltaP) 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.

   b) 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 for all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm total for all three SGs.

   c) The operation leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

PROCEDURES AND PROGRAMS (Continued)

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

4. 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 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 4a, 4b, and 4c 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 locations.

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

   b) 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 1), 2), 3), and 4) 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.

      1) 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;

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

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

PROCEDURES AND PROGRAMS (Continued)

4) 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.

c) 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.

5. Provisions for monitoring operational primary-to-secondary leakage.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

o. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.l. 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,

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT (Continued)

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 the effective plugging percentage in each steam generator, and

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED By Amendment No.92)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO RENEWED FACILITY

OPERATING LICENSE NO. NPF-63

DUKE ENERGY PROGRESS, INC.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By application dated April 24, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14114A743), Duke Energy Progress, Inc. (Duke Energy, or the licensee) requested changes to the Technical Specifications (TSs) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed changes concern Shearon Harris steam generator inspection periods, and address applicable administrative changes and certain clarifications. The plant-specific changes concern TS 6.8.4.l, "Steam Generator (SG) Program" 6.9.1.7, "Steam Generator Tube Inspection Report," and TS 3/4.4.5, "Steam Generator Tube Integrity."

Per the licensee, the specific changes are in accordance with 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 (SG) Program Inspection Frequencies and Tube Sample Selection" (ADAMS Accession No. ML110610350).

TSTF Travelers, such as TSTF-510, evaluate changes to the STSs. The STSs applicable to the HNP-1 Nuclear Steam Supply System is NUREG-1431, "Standard Technical Specifications Westinghouse Plants." The current STS provisions related to SG programs 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 STSs incorporated a new, largely performance-based approach for ensuring that the integrity of the SG tubes is maintained. The performance-based provisions were supplemented by prescriptive provisions relating to tube inspections and tube repair limits to ensure that conditions adverse to quality are detected and corrected on a timely basis. By letter dated March 16, 2007 (ADAMS Accession No. ML070510525), the NRC approved TSTF-449 for implementation in the HNP-1 TSs.

After the issuance of TSTF-449, TSTF-510 was developed to reflect the industry's early implementation experience with respect to TSTF-449. 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. Further, according to the licensee's application, 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 following section details the regulatory requirements and guidance used by the NRC staff to evaluate the application.

2.0 REGULATORY EVALUATION

The SG tubes in pressurized-water reactors (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.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) establish 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 to assure that the design conditions ... are not exceeded ..." (GDC 15), "shall be designed with sufficient margin that when stressed ... (1) the boundary behaves in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized" (GDC 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). These GDC are referred to in TSTF-510. The HNP Updated Final Safety Analysis Report (UFSAR) Section 3.1 provides an evaluation of the design bases of HNP against the GDC discussed above. The staff's review of this section shows how the licensee meets these GDC requirements.

Paragraph 50.55a(c)(1) of 10 CFR specifies that components that 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). Paragraph 50.55a(g)(4) of 10 CFR 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 preservice 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.

Section 50.36 of 10 CFR, "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. As described in TSTF-510, LCOs and accompanying action statements and SRs in the STSs 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 licensee states, "The HNP TS utilizes different numbering than the Standard Technical Specifications (STS) on which TSTF-510-A was based. Specifically, the STS and corresponding HNP numbering is as follows."

STS Numbering	**HNP Numbering**
TS 3.4.20, "Steam Generator (SG) Tube Integrity"	TS 3/4.4.5, "Steam Generator (SG) Tube Integrity"
Limiting Condition for Operation (LCO) 3.4.20	LCO 3.4.5
LCO 3.4.20, Condition A	LCO 3.4.5, Action a
Surveillance Requirement (SR) 3.4.20.2	SR 4.4.5.2
Specification 5.5.9, "Steam Generator (SG) Program"	Specification 6.8.4.l, "Steam Generator (SG) Program." Note: HNP also uses a different numbering scheme within Specification 6.8.4.l.
Specification 5.6.7, "Steam Generator Tube Inspection Report"	Specification 6.9.1.7, "Steam Generator Tube Inspection Report"

The HNP sections listed in the above table contain requirements similar to those specified in STS sections listed above.

Paragraph 50.36(c)(5) of 10 CFR 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. For HNP, the SG Program is defined in Specification 6.8.4.l, while the reporting requirements relating to implementation of the SG Program are in Specification 6.9.1.7.

Specification 6.8.4.l requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Specification 6.8.4.l.1 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. SG tube integrity is maintained by meeting the performance criteria specified in TS 6.8.4.l.2 for structural and leakage integrity, consistent with the plant design and licensing basis. The applicable tube repair criteria specified in TS 6.8.4.l.3 are that tubes found during ISI to contain flaws with a depth equal to or exceeding 40 percent (%) of the nominal wall thickness shall be plugged. Specification 6.8.4.l.4 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.

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.l, "Steam Generator (SG) Program"

The last sentence in the introductory paragraph of this TS currently states, "In addition, the Steam Generator Program shall include the following provisions."

Proposed Change: Delete the word "provisions", thus changing the sentence to "In addition, the Steam Generator Program shall include the following:" The change is needed to remove the duplication since the subsequent paragraphs in the TS starts with "Provisions for."

Evaluation: The NRC staff reviewed the licensee's proposed change to Specification 6.8.4.l and has determined that the word "provisions" in the introductory paragraph is duplicative. The NRC staff has determined that the editorial change is corrective or minor in nature, changes no technical requirements, and, therefore, is acceptable.

3.2 Specification 6.8.4.l, Paragraph 6.8.4.l.2.a, "Structural integrity performance criterion"

The first sentence of this TS currently states:

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

Proposed Change: Revise the sentence by moving the ")", after cooldown, removing "and" after cooldown, and inserting a comma after specification as follows:

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

Evaluation: The basis for the change is that the sentence inappropriately includes anticipated transients in the description of normal operating conditions. The NRC staff determined that the change is corrective in nature in that the current wording is incorrect because anticipated transients should be differentiated from normal operating conditions because each refers to separate and distinct parameters. Therefore, the NRC staff finds the change acceptable.

3.3. Paragraph 6.8.4.I.3, "Provisions for SG tube repair criteria," Paragraph 6.8.4.I.4, "Provisions for SG tube inspections," TS 3/4.4.5, "Steam Generator (SG) Tube Integrity," and SR 4.4.5.2 for LCO 3.4.5

Proposed Change: Change all instances of the term "tube repair criteria" to "tube plugging criteria." This change is intended to be consistent with the treatment of SG tube plugging throughout TS 3/4.4.5 and TS 6.8.4.I.1.

Evaluation: The NRC staff finds that the proposed changes more accurately labels the criteria and, therefore, adds clarity to the specification. Generally, 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 in the TSs by paragraph 6.8.4.I.3. HNP does not have any approved alternate repair criteria, and thus plugging is the only available option if the criteria are exceeded. Therefore, the NRC staff finds these corrective changes acceptable.

3.4 Paragraph 6.8.4.I.4, "Provisions for SG tube inspection"

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

Evaluation: The proposed editorial change does not alter technical requirements. Therefore, the NRC staff finds the change acceptable.

3.5 Paragraph 6.8.4.I.4.a

The paragraph currently states: "Inspect 100 % of the tubes in each SG during the first refueling outage following SG replacement."

Proposed change: 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.

Evaluation: The NRC staff finds that the SG Program can apply to both existing and new plants and the wording change allows for consistency between HNP and other plants. Since this wording modification does not alter any technical or functional requirements for HNP, the NRC staff finds it acceptable.

3.6 Paragraph 6.8.4.I.4.b (SGs with alloy 690 thermally treated tubes)

TSTF-510 is written to accommodate plants with several variations of SG tubing material. As described in the HNP UFSAR, Amendment 59, Chapter 5.4.2, the HNP SGs employ a thermally treated (TT) alloy 690 tubing design.

Paragraph 6.8.4.I.4.b currently states:

> Inspect 100% of the tubes at sequential periods of 144, 108, 72, 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 outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

Proposed Change: Replace paragraph 6.8.4.I.4.b with the following insert:

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 1), 2), 3), and 4) 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.

1) 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;

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

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

4) 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.

Evaluation: Regarding paragraph 6.8.4.I.4.b, the proposed change relocates the first two sentences, "Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs," of paragraph 6.8.4.I.4.b to the inspection periods specified in 1 through 4 of the revised paragraph, and clarifies existing inspection requirements for the sequential periods. The NRC staff finds the relocation of these two sentences and editorial changes to be clarifying in nature, do not change the current intent of these two sentences, and are acceptable.

In addition to the relocation and editorial changes, the licensee proposed three changes to the inspection periods. The duration of the inspection periods would be changed as stated below:

- The second inspection period would be revised from 108 to 120 EFPM [effective full power months].
- The third inspection period would be revised from 72 to 96 EFPM.
- The fourth and subsequent inspection periods would be revised from 60 to 72 EFPM.

The licensee characterizes these changes as marginal increases for consistency with typical fuel cycle lengths that better accommodate the scheduling of refueling outage inspections. The NRC staff finds that, depending on the actual plant inspection schedule, these changes could impact the number of inspections in a given period, as well as the sample size. However, inspection sample sizes will continue to be subject to paragraph 6.8.4.I.4.b, which states that in addition to meeting the requirements of paragraph 6.8.4.I.4.b, 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 extensions to the length of the second and subsequent inspection periods, compliance with the SG program requirements in Specification 6.8.4.I.4.b will continue to ensure both adequate inspection scopes and tube integrity for the reasons addressed below.

For each inspection period, paragraph 6.8.4.I.4.b 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, as compared to those in the second half, even when there are uniform intervals between each inspection. For example, a plant in the second (120 EFPM) inspection period with a scheduled 36-month interval (two 18-month 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 period, which would be the third refueling outage in the period (after 54 EFPM), 6 months before the mid-point (assuming an inspection was performed at the very end of the 144 EFPM inspection period). However, since no inspection is scheduled for that outage (because inspections take place every other outage – once every 36 months), 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. Also, the current TS allows variation in sample sizes from inspection to inspection within a given period. 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 provides more consistency in the refueling outage inspection minimum initial sampling requirement and is acceptable.

The proposed third and fourth sentences of paragraph 6.8.4.I.4.b state,

> 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.

These sentences address the possibility that a degradation assessment in accordance with paragraph 6.8.4.I.4.b will indicate that the tubing may be susceptible to a type of degradation at a location not previously inspected with a technique capable of detecting that type of degradation at that location (for example, new information from another similar plant becomes available), indicating the potential for circumferential cracking at a specific location on the tube. Thus, previous degradation assessments would not have identified the potential for this type of degradation at this location and previous inspections of this location would not have been performed with a technique capable of detecting circumferential cracks. However, once the potential for circumferential cracking is identified at this location, revised paragraph 6.8.4.I.4.b would require an inspection with a method capable of detection of a crack that may satisfy the applicable tube plugging criteria.

Furthermore, if this inspection is performed for the first time during the third of fourth SG inspections scheduled for the 144 EFPM inspection period, the current paragraph 6.8.4.I.4.b does not specifically identify whether 100 percent of the tubes at this location need to be inspected by the end of the 144 EFPM inspection period using a method capable of detection, 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 Accession 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 [5.5.9.]"d" [Standard Technical Specifications for Westinghouse Plants (NUREG-1431)] 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 5.5.9.d is a performance-based element because it describes the goal of the inspections but does not specify how to achieve the goal. [However, this] paragraph "d.2" is a prescriptive element because it specifies that the licensee must inspect 100 percent of the tubes at specified periods. [HNP-1 TS 6.8.4.l.4.b contains information similar to Paragraph 5.5.9.d.]
>
> 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 [5.5.9.] "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 relocation of information in proposed sentences 3 and 4, as described above, clarifies the existing requirement, such that it is consistent with the NRC staff's position from RIS 2009-04, and is, therefore, acceptable.

The proposed fifth sentence in paragraph 6.8.4.l.4.b states, "Each inspection period defined below may be extended up to 3 effective full power months (EFPM) 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 EFPM potentially impacts the average tube inspection sample size to be implemented during a given inspection in that period. For example, if four SG inspections are scheduled to occur within the nominal 144 EFPM period, the minimum sample size for each of the four inspections could average as little as 25 percent of the tube population. If a licensee chooses to include a fifth inspection within the period by extending the period by 3 EFPM, then the minimum sample size for each of the five inspections could average as little as 20 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.l.4.b, which states, in part, that in addition to meeting the requirements of paragraph 6.8.4.l.4.b, "the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next scheduled SG inspection." This requirement remains unchanged under the proposal. The NRC staff concludes the proposed fifth sentence, which allows for smaller sample sizes, involves only a minor relaxation to the existing sampling requirements in paragraph 6.8.4.l.4.b. In addition, these requirements are enhanced by the performance-based requirements in 6.8.4.l.4.b which ensure that adequate inspection sampling will be performed and 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.l.4.b, "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)," replaces the last sentence of the current paragraph 6.8.4.l.4.b, "No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected." Because the minimum allowable SG inspection frequency of at least every 72 EFPM or at least every third refueling outage (whichever results in more frequent inspections) remains unchanged from the current requirement in the HNP TSs, the NRC staff finds that the changes in the sentence are editorial in nature and do not substantially change the existing requirements. Thus, the NRC staff concludes the proposed change is acceptable.

3.7 Paragraph 6.8.4.l.4.c The first sentence of paragraph 6.8.4.l.4.c currently states:

> If crack indications are found in any SG tube, 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 by replacing the words "for each SG" with words "for each affected and potentially affected SG":

> 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).

The change is proposed to clarify the intent of this statement.

Evaluation: The proposed changes in paragraph 6.8.4.l.4.b permit SG inspection intervals to extend over multiple fuel cycles for SGs with alloy 600 TT and 690 TT tubing, assuming that such intervals can be implemented while ensuring tube integrity is maintained in accordance with paragraph 6.8.4.l. However, stress-corrosion cracks may not become detectable by inspection until the crack depth approaches the tube plugging criteria. In addition, stress-corrosion cracks may exhibit high growth rates. Once cracks have been found in any SG tube, current paragraph 6.8.4.l.4.c restricts the allowable interval to the next scheduled inspection to 24 EFPM or one refueling outage (whichever is less). The licensee states this requirement is intended to apply to the affected SG and to any other SG at that unit, which may be potentially affected by the degradation mechanism that caused the known crack(s).

For example, if a root cause analysis in response to the initial finding of one or more cracks reveals that the crack(s) are associated with a manufacturing anomaly which causes locally high residual stress, which in turn, caused the early initiation of cracks at the affected locations and 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 finds it reasonable for the licensee to inspect only the affected SG within 24 EFPM or one refueling cycle in accordance with revised paragraph 6.8.4.I.4.c. Conversely, if it cannot be established that the manufacturing anomaly applies to just one SG, then all potentially affected SGs would have to be inspected. The next scheduled inspections of the other SGs would continue to be subject to all other provisions of paragraph 6.8.4.I.4.c. The NRC staff finds the proposed change to paragraph 6.8.4.I.4.c acceptable because it requires inspections be performed to ensure tube integrity consistent with scope of the suspected degradation mechanism.

3.8 Specification 6.9.1.7, "Steam Generator Tube Inspection Report"

This specification lists Items a through g to be included in a report that must be submitted within 180 days after the average reactor coolant temperature exceeds 200 degrees Fahrenheit following completion of an inspection performed in accordance with the Specification 6.8.4.I, "Steam Generator (SG) Program."

Proposed Change: Delete the word "Active" in Items b and e as follows:

Item b: "Active degradation mechanisms found." Would be revised to: "Degradation mechanisms found."

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

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

Evaluation: The proposed revisions to Items b and e would require that any degradation mechanisms found, whether deemed to be active or not, be reportable. The NRC staff finds these changes acceptable because the revised TS are more restrictive. In addition, the NRC staff finds the added reporting requirement regarding the effective percentage of tube plugging is more restrictive and acceptable.

3.9 Technical Conclusion

The NRC staff has reviewed the licensee's proposed changes and concludes that they are acceptable for the reasons described above.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and an inspection or surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (79 FR 42543). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributor: R. Grover
K. Karwoski

Date: January 9, 2015

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

Sincerely,

/RA/

Martha Barillas, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 145 to NPF-63
2. Safety Evaluation

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