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

February 2, 2016

Mr. Fadi Diya
Senior Vice President and
Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-510, REVISION 2 (CAC NO. MF5826)

Dear Mr. Diya:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 215 to Renewed Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 9, 2015, as supplemented by letters dated April 8, August 12, and December 10, 2015.

The amendment revises TS requirements regarding steam generator tube inspections and reporting as described in TS Task Force (TSTF) traveler TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," with some minor administrative differences.

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

Sincerely,

L. John Klos, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 215 to NPF-30
2. Safety Evaluation

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 215
License No. NPF-30

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

   A. The application for amendment by Union Electric Company (UE, the licensee), dated March 9, 2015, as supplemented by letters dated April 8, August 12, and December 10, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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 2.C.(2) of Renewed Facility Operating License No. NPF-30 is hereby amended to read as follows:

    (2) <u>Technical Specifications and Environmental Protection Plan</u>*

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 215 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance, and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. NPF-30 and
Technical Specifications

Date of Issuance: February 2, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 215

RENEWED FACILITY OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Replace the following pages of the Renewed Facility Operating License No. NPF-30 and 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.

Renewed Facility Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
TOC 4	TOC 4
3.4-44	3.4-44
3.4-46	3.4-46
5.0-10 to 5.0-26	5.0-10 to 5.0-31

(3) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed 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

UE is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 215 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Environmental Qualification (Section 3.11, SSER #3)**

Deleted per Amendment No. 169.

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* Amendments 133, 134, & 135 were effective as of April 30, 2000 however these amendments were implemented on April 1, 2000.

** 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 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

<u>AND</u>

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with 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 inspection.	7 days
	<u>AND</u>		
	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

(continued)

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 (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.9 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

(continued)

5.5.9 Steam Generator (SG) Program (continued)

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.

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

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 (3DP) 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 for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total for all four steam generators.

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

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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 locations.

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

2. After the first refueling outage following SG installation, inspect each SG at least every 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 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

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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.

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Rev. 2, and uses the test procedure guidance in Regulatory Guide 1.52, Revision 2, Positions C.5.a, C.5.c and C.5.d.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

(continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Filtration	2.0%	70%
Control Room Pressurization	2.0%	70%
Emergency Exhaust System	2.0%	70%

d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration	5.4" WG	2000 cfm, ± 200 cfm
Control Room Pressurization	5.4" WG	500 cfm, +500,- 50 cfm
Emergency Exhaust System	5.4" WG	9000 cfm, ± 900 cfm

(continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI 510-1975 and corrected to design nameplate voltage settings.

ESF Ventilation System	Wattage
Control Room Pressurization	15 ± 2 KW
Emergency Exhaust System	37 ± 3 KW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure, Revision 0". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures, Revision 2".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in the outdoor liquid radwaste tanks listed below that are not

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste System is less than the quantities determined in accordance with the Standard Review Plan, Section 15.7.3:

a. Reactor Makeup Water Storage Tank,
b. Refueling Water Storage Tank,
c. Condensate Storage Tank, and
d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a water and sediment content within limits for ASTM 2D fuel oil.

b. Other properties for ASTM 2D fuel oil are analyzed within 31 days following sampling and addition of new fuel oil to storage tanks; and

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 31 days based on applicable ASTM D-2276 standards.

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program

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

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

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

   1. a change in the TS incorporated in the license; or

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

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

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

(continued)

5.5.15 Safety Function Determination Program (SFDP) (continued)

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

   1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

(continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

3. The unit is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement during the Refuel 14 outage (fall of 2005).

4. The first Type A test performed after the October 26, 1999 Type A test shall be performed no later than October 25, 2014.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.1 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests;

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$;

b) For each door, leakage rate is $\leq 0.005\ L_a$ when pressurized to $\geq$ 10 psig.

e. The provisions of Technical Specification SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of Technical Specification SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.5.17 Control Room Envelope Habitability Program

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

a. The definition of the CRE, CRE boundary, control building envelope (CBE), and the CBE Boundary.

b. Requirements for maintaining the CRE and CBE boundaries in their design condition, including configuration control and preventive maintenance.

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

   The following exception is taken to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

   1. The Tracer Gas Test based on the Brookhaven National Laboratory Atmospheric Tracer Depletion (ATD) Method is used to determine the unfiltered air inleakage past the CRE and CBE boundaries. The ATD Method is described in AmerenUE letters dated December 15, 2004 (ULNRC-05104), June 6, 2006 (ULNRC-05298), July 16, 2007 (ULNRC-05427), and October 30, 2007 (ULNRC-05448).

d. Measurement, at designated locations, of the CRE pressure relative to the outside atmosphere during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

(continued)

5.5 Programs and Manuals

5.5.17 Control Room Envelope Habitability Program (continued)

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

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE and CBE unfiltered inleakage, and measuring CRE pressure and assessing CRE and CBE as required by paragraphs c and d, respectively.

5.5.18 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period.

The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Not used.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient limits in Specification 3.1.3,

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference Limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor, $F_Q(Z)$, F_Q^{RTP}, K(Z), W(Z) and F_Q Penalty Factors for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$, $F_{\Delta H}{}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3.2.2,

7. Shutdown Margin Limits for Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8,

8. Reactor Core Safety Limits Figure for Specification 2.1.1,

9. Overtemperature ΔT and Overpower ΔT Setpoint Parameters for Specification 3.3.1, and

10. Reactor Coolant System Pressure and Temperature DNB Limits for Specification 3.4.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY."

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION."

3. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE."

(continued)

5.6 Reporting Requirements

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT."

5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."

6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."

7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."

8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."

9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER DT AND THERMAL OVERTEMPERATURE DT TRIP FUNCTIONS."

10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."

11. WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."

12. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Cold Overpressure Mitigation System (COMS)."

b. The analytical methods used to determine the RCS pressure and temperature and COMS PORV limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves".

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used.

5.6.8 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.

5.6.9 Not used.

(continued)

5.6 Reporting Requirements (continued)

5.6.10 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 Specification 5.5.9, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. Degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

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

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

f. The number and percentage of tubes plugged to date, and 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.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601 (a) and (b) of 10 CFR Part 20:

5.7.1 <u>High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation:</u>

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment;

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits does rate and cumulative dose rate information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(continued)

5.7 High Radiation Area

5.7.1 <u>High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation:</u> (continued)

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:</u>

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the Shift Manager/Operating Supervisor or Radiation Protection Department Supervision, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

(Continued)

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:</u> (continued)

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably

(Continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individual qualified in radiation protection procedures or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area, such as PWR containment, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 215 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By application dated March 9, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15068A422), as supplemented by letters dated April 8, August 12, and December 10, 2015 (ADAMS Accession Nos. ML15098A575, ML15224B472, and ML15344A338, respectively), Union Electric Company (dba Ameren Missouri, the licensee) requested changes to Renewed Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1 (Callaway). The licensee is proposing to amend the Technical Specifications (TS) and adopt TS Task Force (TSTF) traveler TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," with some minor administrative differences.

The supplemental letters dated August 12 and December 10, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 9, 2015 (80 FR 32630).

In its application, the licensee proposed changes to TS Limiting Condition for Operation (LCO) 3.4.17, "Steam Generator (SG) Tube Integrity," as well as TS 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report." The licensee stated that the changes are needed to address implementation issues associated with the inspection periods, and address other administrative changes and clarifications.

In Section 2.1 of the Enclosure to the letter dated March 9, 2015, the licensee stated:

> Ameren Missouri has reviewed TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," as well as the model safety evaluation dated October 27, 2011 (76 FR 66763) provided as part of the Federal Register Notice for Availability. As described in the subsequent paragraphs, Ameren Missouri has concluded that the justifications

> presented in TSTF-510 and the model safety evaluation prepared by the NRC staff are applicable to Ameren Missouri and justify the requested amendment for incorporation of the applicable changes to the Callaway Plant Technical Specifications (TSs)."

The licensee's supplemental letter dated April 8, 2015, stated, in part, that:

> Correcting the noted inconsistencies and minor errors requires changes to be made to the text of the affected Technical Specifications or to what was proposed to be changed for those Technical Specifications, as originally presented in Attachments 1 and/or 3 of the original submittal. The changes include adding the phrase "and that may satisfy the applicable tube plugging criteria" to the third sentence in proposed TS 5.5.9.d.2, as well as removing the word "active" from TS 5.6.10.e. Additional items include the removal of several double spaces between words, as well as the use of a hyphenated form of "in-service" in TS 5.5.9.b.l, and making grammatical improvements to TS 5.6.10 by removing the second instance of "the" and putting "Steam Generator (SG) Program" in quotes. Correction of these inconsistencies is reflected in the attached.

The NRC staff's technical evaluation for the proposed changes is provided in Section 3.0 of this safety evaluation.

Technical Specifications Task Force (TSTF) Travelers, such as TSTF-510, evaluate changes to the Standard Technical Specifications (STSs). The STSs applicable to the Callaway is NUREG-1431, Revision 4, Volume 1, "Standard Technical Specifications Westinghouse Plants," April 2012 (ADAMS Accession No. ML12100A222). 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," and the availability of this TS improvement was announced in the *Federal Register* on May 6, 2005 (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 September 29, 2005 (ADAMS Accession No. ML052570054), in License Amendment No. 168, the NRC approved TSTF-449 for implementation in the Callaway TSs.

After the issuance of TSTF-449, TSTF-510 was developed to reflect the industry's early implementation experience with respect to TSTF-449. Technical Specification Task Force traveler TSTF-510 characterizes the changes as editorial corrections, changes, and clarifications intended to improve internal 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, "Reactor pressure coolant boundary"), "shall be designed with sufficient margin to assure that the design conditions ... are not exceeded ..." (GDC 15, "Reactor coolant system design"), "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, "Fracture prevention of reactor coolant pressure boundary"), shall be of "the highest quality standards practical" (GDC 30, "Quality of reactor coolant pressure boundary"), and "shall be designed to permit periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32, "Inspection of reactor coolant pressure boundary"). These GDC are referred to in TSTF-510. The licensee's Updated Final Safety Analysis Report (UFSAR) Section 3.1 provides an evaluation of the design bases of Callaway against the GDCs discussed above. The NRC 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) 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 stated that the Callaway TSs utilize different numbering than the STS on which

TSTF-510 was based. Specifically, the STS and corresponding Callaway TS numbering is as follows.

Callaway TS Numbering	STS NUREG-1431 Numbering
LCO 3.4.17, "Steam Generator (SG) Tube Integrity"	LCO 3.4.20, "Steam Generator (SG) Tube Integrity"
5.6.10, "Steam Generator Tube Specification Inspection Report"	5.6.7, "Steam Generator Tube Inspection Report"

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

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. The Callaway TS defines the SG Program in TS 5.5.9, while the reporting requirements relating to implementation of the SG Program are defined in TS 5.6.10.

Callaway TS 5.5.9 requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Specification 5.5.9.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected to confirm that the performance criteria are being met. SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.9.b. for structural and leakage integrity, consistent with the plant design and licensing basis. The applicable tube repair criteria specified in TS 5.5.9. c. 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 5.5.9.d. includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that: (1) may be present along the length of a tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet; and (2) may satisfy the applicable tube repair criteria.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

TS Table of Contents

In its letter dated December 10, 2015, the licensee proposed editorial changes to the TS table of contents (TOC) to reflect pagination changes associated with the incorporation of changes and/or additional text into the TS such that some of the page numbers listed on the table of contents were affected. This supplement is acceptable because this new TOC page is administrative update to reflect the change to the TS.

TS 3.4.17, Steam Generator (SG) Tube Integrity

Current TS LCO 3.4.17 states:

> SG tube integrity shall be maintained.
>
> AND
>
> All SG tubes satisfying the tube repair criteria shall be plugged in accordance with Steam Generator Program.

Revised TS LCO 3.4.17 would state:

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

Current TS LCO 3.4.17 Condition A states:

> One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.

Revised TS LCO 3.4.17 Condition A would state:

> One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.

Current SR 3.4.17.2 states:

> Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.

Revised SR 3.4.17.2 would state:

> Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.

TS 5.5.9, Steam Generator (SG) Program

Current introductory paragraph of TS 5.5.9 states, in part, that:

> 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 provisions:

Revised introductory paragraph of TS 5.5.9 would state, in part, that:

> 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:

Current TS 5.5.9.b.1 states, in part, that:

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

Revised TS 5.5.9.b.1 would state, in part, that:

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

Current TS 5.5.9.c states:

> Provisions for SG tube repair 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.

Revised TS 5.5.9.c would state:

> 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.

Current TS 5.5.9.d states:

> 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 repair 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. An assessment of degradation 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.

Revised TS 5.5.9.d would state:

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 locations.

Current TS 5.5.9.d.1 states:

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

Revised TS 5.5.9.d.1 would state:

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

Current TS 5.5.9.d.2 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.

Revised TS 5.5.9.d.2 would state:

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 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.

Current TS 5.5.9.d.3 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). 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.

Revised TS 5.5.9.d.3 would state:

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.

TS 5.6.10, Steam Generator Tube Inspection Report

Current introductory paragraph of TS 5.6.10 states, in part, that:

> 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 5.5.9, Steam Generator (SG) Program. The report shall include:

Revised introductory paragraph of TS 5.6.10 would state, in part, that:

> A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, "Steam Generator (SG) Program." The report shall include:

Current TS 5.6.10.b states:

> Active degradation mechanisms found;

Revised TS 5.6.10.b would state:

> Degradation mechanisms found;

Current TS 5.6.10.e states:

> Number of tubes plugged during the inspection outage for each active degradation mechanism;

Revised TS 5.6.10.e would state:

> Number of tubes plugged during the inspection outage for each degradation mechanism;

Current TS 5.6.10.f states:

> Total number and percentage of tubes plugged to date; and

Revised TS 5.6.10.f would state:

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

3.2 NRC Staff Evaluation

Each proposed change to the TSs is described individually below, followed by the NRC staff's assessment of the change. The affected changes are shown in 'bold' text.

Specification 5.5.9, "Steam Generator (SG) Program," currently states, in part, that:

> In addition, the Steam Generator Program shall include the following **provisions:**

The proposed change deletes the word "provisions," thus changing the sentence to:

> In addition, the Steam Generator Program shall include the following:

The NRC staff reviewed the licensee's proposed change to Specification 5.5.9, and determined that the word "provisions," in the introductory paragraph is duplicative, that the editorial change is corrective or minor in nature, and it changes no technical requirements; therefore, the change is acceptable.

Paragraph 5.5.9.b.1, "Structural integrity performance criterion," currently states, in part, that:

> All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown**), and** all anticipated transients included in the design specification) and design basis accidents.

The proposed change revises the sentence by hyphenating the word "inservice," and **moving the ")",** after cooldown, removing "**and,**" after cooldown, and replacing **")"** with a **comma** after specification as follows:

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

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 since each refers to separate and distinct parameters. Therefore, the NRC staff concludes that the change is acceptable.

Paragraph 5.5.9.c, "Provisions for SG tube repair criteria," Paragraph 5.5.9.d, "Provisions for SG tube inspections," TS LCO 3.4.17, "Steam Generator (SG) Tube Integrity," and SR 3.4.17.2, currently refer to term "**tube repair criteria.**"

The proposed change replaces all instances of the term "tube repair criteria," with "**tube plugging criteria.**"

The NRC staff concludes that the proposed change 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 5.5.9.c. As stated in the application, the licensee does not have an approved tube repair criteria, and thus plugging is the only available option if the criteria are exceeded. Since this option (tube plugging) has been approved by the NRC staff, the staff concludes that the change is acceptable.

Paragraph 5.5.9.d, "Provisions for SG tube inspection," currently states:

> An **assessment of degradation** shall be performed ....at what locations.

The proposed change modifies term "assessment of degradation" to "**degradation assessment**" to be consistent with the terminology used in industry program documents.

The proposed editorial change does not alter technical requirements; therefore, the NRC staff concludes that the change is acceptable.

Paragraph 5.5.9.d.1 currently states:

> Inspect 100% of the tubes in each SG during the first refueling outage following **SG replacement.**

The proposed change replaces the term "SG replacement" with "**SG installation**" to allow the SG Program to be applied to both existing plants and new plants.

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

Paragraph 5.5.9.d.2 (SGs with Alloy 690 thermally treated tubes)

TSTF-510 is written to accommodate plants with several variations of SG tubing material. In the licensee's application, the Callaway SGs employ a thermally treated Alloy 690 tubing design.

Paragraph 5.5.9.d.2 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.

The proposed change replaces paragraph 5.5.9.d.2 in its entirety 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 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 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.

Regarding paragraph 5.5.9.d.2, the proposed change relocates the currently stated 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," to the inspection periods specified in a through d of the revised paragraph 5.5.9.d.2, and clarifies existing inspection requirements for the sequential periods. The NRC staff concludes that 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 effective full power months (EFPM).

- 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 notes 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 5.5.9.d.2, which states that in addition to meeting the requirements of paragraph 5.5.9.d.2, 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 5.5.9.d.2 will continue to ensure both adequate inspection scopes and tube integrity for the reasons addressed below.

For each inspection period, paragraph 5.5.9.d.2 currently requires that at least 50 percent of the tubes be inspected by the refueling outage nearest to the mid-point of the inspection period and the remaining 50 percent by the refueling outage nearest the end of the inspection period. The NRC staff notes that, if there are not an equal number of inspections in the first half and second half of the inspection period, the average minimum sampling requirement may be markedly different for inspections in the first half of the inspection period, 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, therefore, acceptable.

The proposed third and fourth sentences of paragraph 5.5.9.d.2 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 5.5.9.d.2 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 5.5.9.d.2 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 5.5.9.d.2 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. [Callaway TS 5.5.9.d 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 needs to 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 concludes 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 5.5.9.d.2 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 5.5.9.d.2, which states, in part, that in addition to meeting the requirements of this paragraph, "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 5.5.9.d.2. In addition, these requirements are enhanced by the performance-based requirements in the subject paragraph 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 5.5.9.d.2, "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 5.5.9.d.2, "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 Callaway TSs, the NRC staff concludes 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.

Paragraph 5.5.9.d.3 currently states (first sentence):

> 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)**.

The proposed change revises the sentence by replacing the words "for each SG" with words for each affected and potentially affected SG" and "(whichever is less)," with "(whichever results in more frequent inspections)", as shown in bold text below:

> 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 proposed changes in paragraph 5.5.9.d.3 permit SG inspection intervals to extend over multiple fuel cycles for SGs with 690 thermally treated tubing, assuming that such intervals can be implemented while ensuring tube integrity is maintained in accordance with paragraph 5.5.9.b. 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 5.5.9.d.3 restricts the allowable interval to the next scheduled inspection to 24 EFPM or one refueling outage (whichever is less). The licensee stated 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 5.5.9.d.3. 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 5.5.9.d.3. The NRC staff concludes that the proposed change to paragraph 5.5.9.d.3 is acceptable because it requires that inspections be performed to ensure tube integrity consistent with scope of the suspected degradation mechanism.

Specification 5.6.10, “Steam Generator Tube Inspection Report,” 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 5.5.9, “Steam Generator (SG) Program.”

The proposed change deletes the word “**Active**” in Items b and e as follows:

> Item b: “Active degradation mechanisms found,” would be revised to state: “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 state: “The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator.”

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 concludes that these changes are 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.

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 Missouri State 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 the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 published in *Federal Register* on June 9, 2015 (80 FR 32630). 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

Date: February 2, 2016

February 2, 2016

Mr. Fadi Diya
Senior Vice President and
Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-510, REVISION 2 (CAC NO. MF5826)

Dear Mr. Diya:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 215 to Renewed Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 9, 2015, as supplemented by letters dated April 8, August 12, and December 10, 2015.

The amendment revises TS requirements regarding steam generator tube inspections and reporting as described in TS Task Force (TSTF) traveler TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," with some minor administrative differences.

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

Sincerely,
/RA/
L. John Klos, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 215 to NPF-30
2. Safety Evaluation

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ADAMS Accession No. ML15324A114

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DSS/STSB/BC
NAME	JKlos	JBurkhardt	RElliott
DATE	12/14/15	12/8/15	12/18/15
OFFICE	OGC – NLO	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME	BMizuno	RPascarelli	JKlos
DATE	1/13/16	02/01/16	02/02/16

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