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

February 24, 2022

Mr. Ken J. Peters
Senior Vice President and
Chief Nuclear Officer
Attention: Regulatory Affairs
Vistra Operations Company LLC
Comanche Peak Nuclear Power Plant
6322 N FM 56
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 182 AND 182 TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-577, REVISION 1, "REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS" (EPID L-2021-LLA-0134)

Dear Mr. Peters:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 182 to Facility Operating License No. NPF-87 and Amendment No. 182 to Facility Operating License No. NPF-89 for Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated July 27, 2021, as supplemented by letters dated August 31, 2021, October 25, 2021, January 25, 2022, and February 15, 2022.

The amendments adopt Technical Specifications Task Force (TSTF) Traveler TSTF–577, "Revised Frequencies for Steam Generator Tube Inspections," which is an approved change to the Standard Technical Specifications, into the Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2, Technical Specifications.

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

Sincerely,

/RA/

Dennis J. Galvin, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 182 to NPF-87
2. Amendment No. 182 to NPF-89
3. Safety Evaluation
4. Notices and Environmental Findings

cc: Listserv

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

COMANCHE PEAK POWER COMPANY LLC

AND VISTRA OPERATIONS COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 182
License No. NPF-87

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

   A. The application for amendment by Vistra Operations Company LLC (Vistra OpCo) dated July 27, 2021, as supplemented by letters dated August 31, 2021, October 25, 2021, January 25, 2022, and February 15, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this license 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 Facility Operating License No. NPF-87 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. 182 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Vistra OpCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity

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Date: 2022.02.24 16:35:58 -05'00'

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance: February 24, 2022

COMANCHE PEAK POWER COMPANY LLC

AND VISTRA OPERATIONS COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 182
License No. NPF-89

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

   A. The application for amendment by Vistra Operations Company LLC (Vistra OpCo) dated July 27, 2021, as supplemented by letters dated August 31, 2021, October 25, 2021, January 25, 2022, and February 15, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this license 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 Facility Operating License No. NPF-89 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A as revised through Amendment No. 182 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Vistra OpCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity

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Date: 2022.02.24 16:36:40 -05'00'

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance: February 24, 2022

ATTACHMENT TO LICENSE AMENDMENT NO. 182

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 182

TO FACILITY OPERATING LICENSE NO. NPF-89

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of Facility Operating License Nos. NPF-87 and NPF-89, and the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-87

REMOVE	INSERT
3	3

Facility Operating License No. NPF-89

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
5.5-4	5.5-4
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.5-11	5.5-11
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	5.5-15
5.5-16	5.5-16
5.5-17	5.5-17
5.5-18	---
5.5-19	---
5.6-5	5.6-5
5.6-6	5.6-6

(3) Vistra OpCo, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) Vistra OpCo, 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;

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

(6) Vistra OpCo, 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 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) <u>Maximum Power Level</u>

Vistra OpCo is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal through Cycle 13 and 3612 megawatts thermal starting with Cycle 14 in accordance with the conditions specified herein.

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

The Technical Specifications contained in Appendix A as revised through Amendment No. 182 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Vistra OpCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Vistra OpCo, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) Vistra OpCo, 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;

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

(6) Vistra OpCo, 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 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) <u>Maximum Power Level</u>

Vistra OpCo is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal through Cycle 11 and 3612 megawatts thermal starting with Cycle 12 in accordance with the conditions specified herein.

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

The Technical Specifications contained in Appendix A as revised through Amendment No. 182 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Vistra OpCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Antitrust Conditions</u>

DELETED

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

5.5.8 Deleted

5.5.9 Steam Generator (SG) Program

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

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

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

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

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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 per SG.

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

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

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

a. For Unit 2 only, tubes with service-induced flaws located greater than 14.01 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 14.01 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except for any portions of the tube that are exempt from inspection by alternate repair criteria, 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 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

2. For the Unit 1 model Delta-76 SGs (Alloy 690 thermally treated) after the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

   For the Unit 2 model D5 SGs (Alloy 600 thermally treated) after the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 54 effective full power months, which defines the inspection period. If none of the SG tubes have ever experienced cracking other than in regions that are exempt from inspection by alternate repair criteria and the SG inspection was performed with enhanced probes, the inspection period may be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of any type equivalent to or better than array probe technology. The enhanced probes shall be used from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except any portions of the tube that are exempt from inspection by alternate repair criteria. If there are regions where enhanced probes cannot be used, the tube inspection techniques shall be capable of detecting all forms of existing and potential degradation in that region.

3. If crack indications are found in any SG tube excluding any region that is exempt from inspection by alternate repair criteria, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indications shall be at the next refueling outage, but for Unit 2, the next inspection may be deferred to the following refueling outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. 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, Revision 2 and in accordance with Regulatory Guide 1.52, Revision 2, ANSI/ASME N509-1980, ANSI/ASME N510-1980, and ASTM D3803-1989.

---NOTE---
ANSI/ASME N510-1980, ANSI/ASME N509-1980, and ASTM D3803-1989 shall be used in place of ANSI 510-1975, ANSI/ASME N509-1976, and ASTM D3803-1979 respectively in complying with Regulatory Guide 1.52, Revision 2.
----

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% for Primary Plant Ventilation System - ESF Filtration units and < 0.05% for all other units when tested in accordance with Regulatory

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency filtration unit	8,000 CFM
Control Room Emergency pressurization unit	800 CFM
Primary Plant Ventilation System – ESF filtration unit	15,000 CFM

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% for Primary Plant Ventilation System - ESF Filtration units and < 0.05% for all other units when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency filtration unit	8,000 CFM
Control Room Emergency pressurization unit	800 CFM
Primary Plant Ventilation System - ESF filtration unit	15,000 CFM

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in 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 greater than or equal to the relative humidity specified below.

ESF Ventilation Systems	Penetration	RH
Control Room Emergency filtration unit	0.5%	70%
Control Room Emergency pressurization unit	0.5%	70%
Primary Plant Ventilation System – ESF filtration unit	2.5%	70%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below ± 10%

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency filtration unit	8.0 in WG	8000 CFM
Control Room Emergency pressurization unit	9.5 in WG	800 CFM
Primary Plant Ventilation System – ESF filtration unit.	8.5 in WG	15000 CFM

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/ASME N510-1980.

ESF Ventilation System	Wattage
Control Room Emergency pressurization unit	10 ± 1 kW
Primary Plant Ventilation System - ESF filtration unit	100 ± 5 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 Waste Processing System, the quantity of radioactivity contained in each Gas Decay Tank, 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, July 1981. 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, July 1981.

5.5 Programs and Manuals

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

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste Processing 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 Decay 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 all outdoor liquid radwaste tanks that are not 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 Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

d. 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 clear and bright appearance with proper color or a water and sediment content within limits.

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program (continued)

b. Within 31 days following addition of the new fuel oil to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

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

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) and exemptions thereto.

5.5.15 Safety Function Determination Program (SFDP)

a. 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:

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

1. 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;

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

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

4. Other appropriate limitations and remedial or compensatory actions.

b. 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:

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

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

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

c. 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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5 Programs and Manuals

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 NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2A, dated October 2008, 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 the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

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

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.3 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criteria 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 Type C tests and $\leq 0.75\ L_a$ for Type A tests;

   2. Air lock testing acceptance criteria are:

      i. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

      ii. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

e. The provision of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program, with the exception of the containment ventilation isolation valves.

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

5.5.17 Technical Requirements Manual (TRM)

The TRM contains selected requirements which do not meet the criteria for inclusion in the Technical Specification but are important to the operation of CPNPP. Much of the information in the TRM was relocated from the TS.

Changes to the TRM shall be made under appropriate administrative controls and reviews. Changes may be made to the TRM without prior NRC approval provided the changes do not require either a change to the TS or NRC approval pursuant to 10 CFR 50.59. TRM changes require approval of the Plant Manager.

5.5.18 Configuration Risk Management Program (CRMP)

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed Completion Time has been granted. The program shall include the following elements:

a. Provisions for the control and implementation of a Level 1, at-power, internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the LCO Action for preplanned activities.

c. Provisions for performing an assessment after entering the LCO Action for unplanned entry into the LCO Action.

d. Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Action.

e. Provisions for considering other applicable risk significant contributors such as Level 2 issues, and external events, qualitatively or quantitatively.

5.5 Programs and Manuals

5.5.19 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer for the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates.

5.5.20 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 Filtration System (CREFS), CRE occupants can control the reactor safety 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 and the CRE boundary.

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

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

   The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

   1. C. - Section 4.3.2 "Periodic CRH Assessment" from NEI 99-03 Revision 1 will be used as input to a site specific Self Assessment procedure.

   2. C.1.2 - No peer reviews are required to be performed.

5.5 Programs and Manuals

5.5.20 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, 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 18 month assessment of the CRE boundary.

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

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

5.5.21 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.

5.5 Programs and Manuals

5.5.22 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

The Region I storage cells in the CPNPP Spent Fuel Pool utilize the neutron absorbing material BORAL, which is credited in the Safety Analysis to ensure the limitations of Technical Specification 4.3.1.1 are maintained.

In order to ensure the reliability of the Neutron Poison material, a monitoring program is required to routinely confirm that the assumptions utilized in the criticality analysis remain valid and bounding. The Neutron Absorber Monitoring Program is established to monitor the integrity of neutron absorber test coupons periodically as described below.

A test coupon "tree" shall be maintained in each SFP. Each coupon tree originally contained 8 neutron absorber surveillance coupons. Detailed measurements were taken on each of these 16 coupons prior to installation, including weight, length, width, thickness at several measurement locations, and B-10 content (g/cm^2). These coupons shall be maintained in the SFP to ensure they are exposed to the same environmental conditions as the neutron absorbers installed in the Region I storage cells, until they are removed for analysis.

One test coupon from each SFP shall be periodically removed and analyzed for potential degradation, per the following schedule. The schedule is established to ensure adequate coupons are available for the planned life of the storage racks.

Year	Coupon Number	Year	Coupon Number
2013	1	2028	5
2015	2	2033	6
2018	3	2043	7
2023	4	2053	8

Further evaluation of the absorber materials, including an investigation into the degradation and potential impacts on the Criticality Safety Analysis, is required if:

- A decrease of more than 5% in B-10 content from the initial value is observed in any test coupon as determined by neutron attenuation.
- An increase in thickness at any point is greater than 25% of the initial thickness at that point.

5.6 Reporting Requirements

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

1. 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 the required actions 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 Steam Generator Tube Inspection Report

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

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

4. The number of tubes plugged during the inspection outage.

5.6 Reporting Requirements

5.6.9 Steam Generator Tube Inspection Report (continued)

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG;

f. The results of any SG secondary side inspections;

g. For Unit 2, the primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report;

h. For Unit 2, the calculated accident induced leakage rate from the portion of the tubes below 14.01 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 3.16 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined; and

i. For Unit 2, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 182 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-87

AND

AMENDMENT NO. 182 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-89

VISTRA OPERATIONS COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

Application (i.e., initial and supplements)	Safety Evaluation Date
• July 27, 2021, ADAMS Accession No. ML21208A023 • August 31, 2021, ADAMS Accession No. ML21243A232 • October 25, 2021, ADAMS Accession No. ML21298A260 • January 25, 2022, ADAMS Accession No. ML22025A412 • February 15, 2022, ADAMS Accession No. ML22046A198	February 24, 2022 Principal Contributors to Safety Evaluation: • Clinton Ashley

1.0 PROPOSED CHANGES

Vistra Operations Company LLC (the licensee) requested changes to the Technical Specifications (TSs) for Comanche Peak Nuclear Power Plant (Comanche Peak), Unit Nos. 1 and 2, in its license amendment request (application). In its application, as supplemented, the licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) process the proposed amendment under the Consolidated Line Item Improvement Process (CLIIP). The proposed changes would revise the "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program" and the "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report" TSs based on Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections," dated March 1, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21060B434), and the associated NRC staff safety evaluation (SE) of TSTF-577, dated April 14, 2021 (ADAMS Accession No. ML21098A188).

The tubes within an SG function as an integral part of the reactor coolant pressure boundary and, in addition, isolate fission products in the primary coolant from the secondary coolant and the environment. SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis.

Comanche Peak has two units. The Unit 1 SGs have Alloy 690 thermally treated (Alloy 690TT) tubes. The Unit 2 SGs have Alloy 600 thermally treated (Alloy 600TT) tubes.

1.1 Proposed TS Changes to Adopt TSTF-577

In accordance with NRC staff-approved TSTF-577, the licensee proposed changes that would revise Comanche Peak, Unit Nos. 1 and 2, TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," and TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report." Specifically, the licensee proposed the following changes to adopt TSTF-577:

TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program":

- TS 5.5.9 introductory paragraph and paragraph b.1 would be revised by replacing "steam generator" with "SG" in a few instances.

- TS 5.5.9.d would be revised by adding a phrase regarding portions of the tube that are exempt from inspection by alternate repair criteria that replaces Unit 2 information specifying distances from top of the tubesheet.

- TS 5.5.9.d.2 and TS 5.5.9.d.3 would be revised by combining into a new TS 5.5.9.d.2 (see additional discussion in SE Section 1.2.1), and the new TS 5.5.9.d.2 would be revised by:

    - deleting the requirement to base inspection frequency on the more restrictive metric between either the effective full power months (EFPM) or refueling outage and to use just the EFPM metric;

    - deleting the allowance to extend the inspection period by 3 months and by deleting the discussion of prorating inspections;

    - changing the Unit 1 requirement to inspect 100 percent of the tubes at periods of 144, 120, 96, and 72 EFPM to 96 EFPM; and

    - changing the Unit 2 requirement to inspect 100 percent of the tubes at periods of 120, 96, 90 (Cycle 19 only), and 72 EFPM to 54 EFPM. A 72 EFPM inspection period would be permitted if SG tubing has never experienced cracking (not including regions exempt from inspection by alternate repair criteria) and the SG inspection was performed with enhanced probes. A description of the enhanced probe inspection would be added.

- TS 5.5.9.d.4 would be revised by renumbering to new TS 5.5.9.d.3 (see additional discussion in SE Section 1.2.1), and the new TS 5.5.9.d.3 would be revised by:

    - adding a phrase regarding portions of the tube that are exempt from inspection by alternate repair criteria that replaces Unit 2 information specifying distances from top of the tubesheet.

- changing the next inspection after crack indications are found from “shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections)” to “shall be at the next refueling outage.”

- adding a phrase that permits deferring Unit 2 SG inspections after cracking indications are found if the 100 percent inspection was performed with enhanced probes.

TS 5.6.9, “Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report”:

- Existing reporting requirement b. would be renumbered as c. and be revised by editorial and punctuation changes.

- New reporting requirement b. would be added to require the nondestructive examination techniques utilized for tubes with increased degradation susceptibility be reported.

- Existing reporting requirement c. would be renumbered as c.1. and be revised by editorial and punctuation changes.

- Existing reporting requirement d. would be renumbered as c.2. and be revised to note that the location, orientation (if linear), measured size (if available), and voltage response do not need to be reported for tube wear indications at support structures that are less than 20 percent through-wall. However, the total number of tube wear indications at support structures that are less than 20 percent through-wall would be reported.

- New reporting requirement d. would be added to require an analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection relative to the applicable performance criteria, including the analysis, methodology, inputs, and results.

- Existing reporting requirement e. would be renumbered as c.4. and be revised by editorial and punctuation changes.

- Existing reporting requirement f. would be renumbered as e. and be revised by editorial and punctuation changes.

- New reporting requirement f. would be added to require the results of any SG secondary side inspections be reported.

- Existing reporting requirement g. would be renumbered as c.3. and be revised to add the requirements to report a description of the condition monitoring assessment, the margin to the tube integrity performance criteria, and a comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment. In addition, the requirement to report the results of tube pulls and in situ testing would be deleted.

- Existing reporting requirements h., i., and j. would be renumbered as new reporting requirement g., h., and i., respectively, that includes existing plant-specific reporting requirements.

1.2 Additional Proposed TS Changes

In addition to the changes proposed consistent with the traveler discussed in SE Section 1.1, the licensee proposed the following variations.

1.2.1 Editorial Variations

The licensee noted the following variations:

- The current Comanche Peak, Unit Nos. 1 and 2, TSs use different nomenclature than the Standard Technical Specifications (STSs).
    - The TS Section 5.5.9 program is currently titled, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program." The licensee proposed to rename TS 5.5.9 to "Steam Generator (SG) Program," for consistency with TSTF-577 and the STSs in NUREG-1431.[1]
    - The TS 5.6.9 report is currently titled, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report." The licensee proposed to rename TS 5.6.9 to "Steam Generator Tube Inspection Report," for consistency with TSTF-577 and the STSs in NUREG-1431.
- TS 5.5.9, paragraph d.2 contains the periodic inspection requirements for the Unit 2 SGs, and paragraph d.3 contains the periodic inspection requirements for the Unit 1 SGs. The revised periodic inspection requirements are contained in paragraph d.2 with the Unit 1 requirements preceding the Unit 2 requirements. Existing paragraph d.4 is renumbered d.3.
- TS 5.5.9, paragraph d.2 contains a one-time inspection interval extension for Unit 2 that is no longer needed.
- TS 5.5.9, paragraphs d and d.3 contain separate requirements for Unit 1 and Unit 2 due to difference in the SG tube alloy. Most of these differences are eliminated by TSTF-577, which includes the exception, "except for any portions of the tube that are exempt from inspection by alternate repair criteria." In the proposed renumbered paragraph d.3, the Alloy 600TT allowance to defer an inspection after detecting cracking is made applicable to only Unit 2.
- TS 5.5.9.d, sixth sentence currently contains, "the requirements below" and TSTF-577 contains, "the requirements of d.1, d.2, and d.3 below." The licensee proposed to retain the current wording.
- TS 5.6.9, first paragraph contains "Specification 5.5.9, Steam Generator (SG) Program" in which the title of TS 5.5.9 is not in quotation marks. In order to align with the STS, quotation marks are added to TS 5.5.9 title as follows: Specification 5.5.9, "Steam Generator (SG) Program."

[1] U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431, Revision 5, Volume 1, "Specifications," and Volume 2, "Bases," dated September 2021 (ADAMS Accession Nos. ML21259A155 and ML21259A159, respectively).

1.2.2 Other Variation

The licensee noted that the Comanche Peak, Unit Nos. 1 and 2, SG Program TS currently contains a provision for an alternate tube plugging criteria. The description of the alternate tube plugging criteria in the proposed change is equivalent to the descriptions in the current TS.

2.0 REGULATORY EVALUATION

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.36(c)(5), "Administrative controls," state that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in [10 CFR] 50.4." Technical Specification Section 5.0, "Administrative Controls," requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Programs established by the licensee, including the SG Program, are listed in the administrative controls section of the TSs to operate the facility in a safe manner.

The NRC staff's guidance for the review of TSs is in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Chapter 16.0, "Technical Specifications," Revision 3, dated March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared STSs for each of the LWR nuclear designs. Accordingly, the NRC staff's review includes consideration of whether the proposed changes are consistent with NUREG-1431, as modified by NRC-approved travelers.

TSTF-577 revised the STSs related to SG tube inspections and SG tube inspection reporting requirements. The NRC approved TSTF-577, under the CLIIP on April 14, 2021 (ADAMS Package Accession No. ML21099A086).

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes to Adopt TSTF-577

The NRC staff compared the licensee's proposed TS changes in Section 1.1 of this SE against the changes approved in TSTF-577. In accordance with SRP Chapter 16.0, the NRC staff determined that the STS changes approved in TSTF-577 are applicable because Comanche Peak, Unit Nos. 1 and 2 are pressurized water reactors (PWRs) and the NRC staff approved the TSTF-577 changes for PWRs. The NRC staff finds that the licensee's proposed changes to the Comanche Peak, Unit Nos. 1 and 2 TSs in Section 1.1 of this SE are consistent with those found acceptable in TSTF-577.

In the SE of TSTF-577, the NRC staff concluded that the TSTF-577 changes to STS 5.5.9, "Steam Generator (SG) Program," and STS 5.6.7, "Steam Generator Tube Inspection Report," were acceptable because, as discussed in Section 3.0 of that SE, they continued to ensure SG tube integrity and, therefore, protected the public health and safety. In particular, the structural integrity performance criterion and accident-induced leakage performance criterion (explained in STS 5.5.9.b, items 1 and 2, respectively) will continue to be met with the proposed revised SG inspection intervals (maximum allowable time between SG inspections) and inspection periods (maximum allowable time between 100 percent of SG tubes inspections). Additionally, the proposed changes to the reporting requirements will provide more detailed and consistent information to the NRC. Therefore, the NRC staff found that the proposed changes to the SG program and inspection reporting requirements were acceptable because they continued to meet the requirements of 10 CFR 50.36(c)(5) by providing administrative controls necessary to assure operation of the facility in a safe manner. For these same reasons, the NRC staff concludes that the corresponding proposed changes to the Comanche Peak, Unit Nos. 1 and 2 TSs in Section 1.1 of this SE continue to meet the requirements of 10 CFR 50.36(c)(5).

3.2 Additional Proposed TS Changes

3.2.1 Editorial Variations

The licensee noted that the current Comanche Peak, Unit Nos. 1 and 2 TSs use different nomenclature (titling) than the STSs and revised the nomenclature to align with the STSs. The NRC staff finds that the nomenclature changes are acceptable because they align with STSs and do not substantively alter TS requirements. In addition, the NRC noted that Comanche Peak, Unit Nos. 1 and 2 TSs have different numbering than STSs on which TSTF-577 was based. Specifically, the renamed “Steam Generator Tube Inspection Report” is numbered TS 5.6.9 in Comanche Peak, Unit Nos. 1 and 2 TSs rather than 5.6.7 as stated in the TSTF. The NRC staff finds that the different TS numbering is acceptable because it does not substantively alter TS requirements.

The licensee noted that the current Comanche Peak, Unit Nos. 1 and 2, TS 5.5.9, paragraphs d.2 and d.3 contain SG periodic inspection requirements for Unit 2 and Unit 1, respectively. The revised SG periodic inspection requirements for both units are now contained in paragraph d.2 with the Unit 1 requirements preceding the Unit 2 requirements. In addition, TS 5.5.9, paragraph d.4 is renumbered as d.3. The NRC staff finds the revised SG periodic inspection requirements being contained in d.2 and the proposed TS renumbering are acceptable because it aligns with the STSs and do not substantively alter TS requirements.

The licensee noted that the current Comanche Peak, Unit Nos. 1 and 2, TS 5.5.9, paragraph d.2 contains a one-time inspection interval extension for Unit 2. The extension permitted an inspection period of 54 EFPM. TSTF-577 provides a permanent 54 EFPM inspection period for plants with Alloy 600TT SG tubes. Comanche Peak, Unit No. 2 uses Alloy 600TT SG tubes. Therefore, because TSTF-577 provides a permanent 54 EFPM inspection period for plants with Alloy 600TT SG tubes, the licensee no longer needs the one-time extension for Unit 2. Based on the discussion above, the NRC staff finds that the deletion of the one-time extension is acceptable.

The licensee noted that the Comanche Peak, Unit Nos. 1 and 2, TS 5.5.9, paragraphs d and d.3 contain separate requirements for Unit 1 and Unit 2 due to differences in SG tube alloy. Most of these differences are eliminated by TSTF-577, which includes the exception, “except for any portions of the tube that are exempt from inspection by alternate repair criteria.” In the proposed renumbered paragraph d.3, the allowance to defer an inspection after detecting cracking is made applicable only to Unit 2. The NRC staff finds the proposed changes to Comanche Peak, Unit Nos. 1 and 2, TS 5.5.9, paragraphs d and d.3 are consistent with the staff-approved TSTF-577 requirements, appropriately identifying unit applicability due to differences in SG tube alloy, and therefore, are acceptable.

The licensee noted that the current Comanche Peak, Unit Nos. 1 and 2, TS 5.5.9.d, sixth sentence contains, “the requirements below “ and TSTF-577 contains, “the requirements of d.1, d.2, and d.3 below." The licensee proposed to maintain the current wording. The NRC staff finds maintaining the current wording is acceptable because it does not substantively alter TS requirements.

The licensee noted that the current Comanche Peak, Unit Nos. 1 and 2, TS 5.6.9, first paragraph contains "Specification 5.5.9, Steam Generator (SG) Program" in which the title of TS 5.5.9 is not in quotations marks. In order to align with the STS, quotation marks are added to TS 5.5.9 title as follows: Specification 5.5.9, "Steam Generator (SG) Program." The NRC staff finds that adding quotation marks acceptable because it is consistent with the STS and does not substantively alter TS requirements.

The NRC staff noted that there were a few punctuation changes (e.g., remove hyphen in "as-found" and inserting a comma between "inspection methods, and inspection intervals") made by the licensee to be consistent with the STS. The NRC staff finds that the punctuation changes acceptable because they are consistent with the STS and do not substantively alter TS requirements.

3.2.2 Other Variation

The licensee noted that the Comanche Peak, Unit Nos. 1 and 2, SG Program TS currently contains a provision for an alternate tube plugging criteria and the descriptions of the alternate tube plugging criteria in the proposed change are equivalent to the descriptions in the current TSs.

The current Comanche Peak, Unit Nos. 1 and 2 TS that address alternate tube plugging criteria (TS 5.5.9.c.1.a) reflects NRC-approved changes contained in Amendment No. 158 (ADAMS Accession No. ML12263A036). As part of the request to adopt TSTF-577, the licensee did not propose any changes to these criteria. Therefore, the NRC staff considers the proposed variation as information for awareness purposes, rather than a variation from the traveler or a change to the plant-specific TS.

3.3 TS Change Consistency

The NRC staff reviewed the proposed TS changes for technical clarity and consistency with the existing requirements for customary terminology and formatting. The NRC staff finds that the proposed changes are consistent with Chapter 16.0 of the SRP and are therefore acceptable.

4.0 <u>CONCLUSION</u>

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

NOTICES AND ENVIRONMENTAL FINDINGS

RELATED TO

AMENDMENT NO. 182 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-87

AND

AMENDMENT NO. 182 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-89

VISTRA OPERATIONS COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

Application (i.e., initial and supplements)	Safety Evaluation Date
• July 27, 2021, ADAMS Accession No. ML21208A023 • August 31, 2021, ADAMS Accession No. ML21243A232 • October 25, 2021, ADAMS Accession No. ML21298A260 • January 25, 2022, ADAMS Accession No. ML22025A412 • February 15, 2022, ADAMS Accession No. ML22046A198	February 24, 2022 [redacted]

1.0 INTRODUCTION

Vistra Operations Company LLC (the licensee) requested changes to the technical specifications (TSs) for Comanche Peak, Unit Nos. 1 and 2, in its license amendment request (application). In its application, as supplemented, the licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) process the proposed amendment under the Consolidated Line Item Improvement Process (CLIIP). The proposed changes would revise the "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program" and the "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report" TSs based on Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections," dated March 1, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21060B434), and the associated NRC staff safety evaluation of TSTF-577, dated April 14, 2021 (ADAMS Accession No. ML21098A188).

The supplements dated October 25, 2021, January 25, 2022, and February 15, 2022, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 2, 2021 (86 FR 60486).

2.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendment on November 16, 2021. The State official had no comments.

3.0 ENVIRONMENTAL CONSIDERATION

The amendments relate, in part, to changes in recordkeeping, reporting, or administrative procedures or requirements. The amendments also relate, in part, to changing requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on November 2, 2021 (86 FR 60486). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 182 AND 182 TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-577, REVISION 1, "REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS" (EPID L-2021-LLA-0134) DATED FEBRUARY 24, 2022

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RidsACRS_MailCTR Resource
RidsNrrDorlLpl4 Resource
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RidsNrrPMComanchePeak Resource
RidsNrrLAPBlechman Resource
RidsRgn4MailCenter Resource
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NAME	DGalvin	PBlechman	NJordan (MHamm for)
DATE	2/24/2022	2/24/2022	11/8/2021
OFFICE	NRR/DNRL/NCSG/BC*	NRR/DORL/LPL4/BC*	NRR/DORL/LPL4/PM
NAME	SBloom	JDixon-Herrity	DGalvin
DATE	11/16/2021	2/24/2022	2/24/2022

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