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

April 13, 2017

Mr. Ken J. Peters
Senior Vice President and
 Chief Nuclear Officer
Attention: Regulatory Affairs
TEX Operations Company LLC
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-545, REVISION 3, "TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5 TESTING" (CAC NOS. MF7684 AND MF7685)

Dear Mr. Peters:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 168 to Facility Operating License No. NPF-87 and Amendment No. 168 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 (TSs) in response to your application dated April 27, 2016, as supplemented by letter dated June 30, 2016.

The amendment deletes TS 5.5.8, "Inservice Testing Program." A new defined term, "Inservice Testing Program," is added to TS Section 1.1, "Definitions." In addition, existing uses of the term "Inservice Testing Program" in the TSs are capitalized throughout to indicate that it is now a defined term. These changes are based on NRC-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing."

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

Sincerely,

Margaret M. Watford, 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. 168 to NPF-87
2. Amendment No. 168 to NPF-89
3. Safety Evaluation

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SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-545, REVISION 3, "TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5 TESTING" (CAC NOS. MF7684 AND MF7685) DATED APRIL 13, 2017

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NAME	MWatford	PBlechman	AKlein*	DAlley*
DATE	04/05/17	04/05/17	03/28/17	03/20/17
OFFICE	OGC NLO	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM	
NAME	BHarris	RPascarelli	MWatford	
DATE	04/07/17	04/13/17	04/13/17	

OFFICIAL AGENCY RECORD

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

COMANCHE PEAK POWER COMPANY LLC

AND TEX OPERATIONS COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. NPF-87

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

   A. The application for amendment by TEX Operations Company LLC (TEX OpCo) dated April 27, 2016, as supplemented by letter dated June 30, 2016, 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) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 168 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TEX 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 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

R J Pascarelli

Robert J. Pascarelli, 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: April 13, 2017

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

COMANCHE PEAK POWER COMPANY LLC

AND TEX OPERATIONS COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. NPF-89

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

   A. The application for amendment by TEX Operations Company LLC (TEX OpCo) dated April 27, 2016, as supplemented by letter dated June 30, 2016, 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. 168 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TEX 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 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

R. J. Pascarelli

Robert J. Pascarelli, 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: April 13, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 168

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 168

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 the Facility Operating License Nos. NPF-87 and NPF-89, 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.

Facility Operating License No. NPF-87

REMOVE	INSERT
3	3

Facility Operating License No. NPF-89

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
1.1-3	1.1-3
3.4-21	3.4-21
3.4-35	3.4-35
3.5-6	3.5-6
3.6-13	3.6-13
3.6-17	3.6-17
3.7-3	3.7-3
3.7-7	3.7-7
3.7-9	3.7-9
3.7-11	3.7-11
3.7-14	3.7-14
5.5-4	5.5-4
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7

REMOVE	INSERT
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	5.5-18
5.5-19	----

(3) TEX 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) TEX 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) TEX 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) TEX 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) Maximum Power Level

TEX 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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 168 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TEX OpCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TEX 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) TEX 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) TEX 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) TEX 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>

TEX 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. 168 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TEX OpCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

DELETED

1.1 Definitions (continued)

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil", or using the dose conversion factors from Table B-1 of Regulatory Guide 1.109, Revision 1, NRC, 1977.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the license program that fulfills the requirements of 10 CFR 50.55a(f).

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-------------------NOTES------------------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. --- Verify leakage from each RCS PIV is equivalent to $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ 2215 psig and $\leq$ 2255 psig.	In accordance with the INSERVICE TESTING PROGRAM, and in accordance with the Surveillance Frequency Control Program. <u>AND</u> Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, and if leakage testing has not been performed in the previous 9 months except for valves 8701A, 8701B, 8702A and 8702B <u>AND</u> Within 24 hours following check valve actuation due to flow through the valve

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.2.3	Verify ECCS piping is full of water.	Prior to entry into MODE 3
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position. <u>Valve Number</u> 8810A 8816A 8822A 8810B 8816B 8822B 8810C 8816C 8822C 8810D 8816D 8822D	In accordance with the Surveillance Frequency Control Program.
SR 3.5.2.8	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.6	Not used.	
SR 3.6.3.7	-----NOTE----- This surveillance is not required when the penetration flow path is isolated by a leak tested blank flange. ----- Perform leakage rate testing for containment purge, hydrogen purge and containment pressure relief valves with resilient seals.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.3.8	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.3.9	Not used.	
SR 3.6.3.10	Not used.	
SR 3.6.3.11	Not used.	
SR 3.6.3.12	Not used.	
SR 3.6.3.13	Not used.	

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.6.2	Not used.	
SR 3.6.6.3	Not used.	
SR 3.6.6.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.6.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.6.7	Not used.	
SR 3.6.6.8	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	--------------------------------NOTE-------------------------------- Only required to be performed in MODES 1 and 2. -- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift setting shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. -- Verify the isolation time of each MSIV is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.2	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. -- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more FIV or FCV bypass valves inoperable.	C.1 Close or isolate bypass valve. AND C.2 Verify bypass valve is closed or isolated.	72 hours Once per 7 days
D. Two valves in the same flowpath inoperable	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time of each FIV, FCV, and associated bypass valves is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.3.2	Verify each FIV, FCV, and associated bypass valves actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ARV.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.4.2	Verify one complete cycle of each ARV block valve.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	-------------------------------NOTE------------------------------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---	
	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.5.2	-------------------------------NOTE------------------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 532 psig in the steam generator. ---	
	Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	-------------------------------NOTE------------------------------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---	
	Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals (continued)

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 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program

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

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

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

    1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), 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 (continued)

5.5.9 Unit 1 Model D76 and Unit 2 Model D5 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. For Unit 1, 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. For Unit 2, 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 14.01 inches below the top of the tubesheet on the hot leg side to 14.01 inches below the top of the tubesheet on the cold leg side 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

5.5 Programs and Manuals

5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

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. For the Unit 2 model D5 steam generators (Alloy 600 thermally treated) after the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

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

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

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

3. For the Unit 1 model Delta-76 steam generators (Alloy 690 thermally treated) after the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every

5.5 Programs and Manuals

5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

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.

4. For Unit 1, 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 indications shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). For Unit 2, if crack indications are found in any SG tube from 14.01 inches below the top of the tubesheet on the hot leg side to 14.01 inches below the top of the tubesheet on the cold leg side, then the next inspection for each affected and potentially affected SG for the degradation mechanism

5.5 Programs and Manuals

5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

that caused the crack indications shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 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.

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

-----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.

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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 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

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

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%

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.

5.5 Programs and Manuals

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

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.

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:

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program (continued)

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.

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 Programs and Manuals (continued)

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:

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.

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

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

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 $\geq P_a$.

5.5.16 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.

5.5.22 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.

5.5 Programs and Manuals

5.5.22 Spent Fuel Storage Rack Neutron Absorber Monitoring Program (continued)

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 168 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 168 TO

FACILITY OPERATING LICENSE NO. NPF-89

COMANCHE PEAK POWER COMPANY LLC

AND TEX OPERATIONS COMPANY LLC

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

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated April 27, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16120A432), as supplemented by letter dated June 30, 2016 (ADAMS Accession No. ML16196A238), TEX Operations Company LLC (previously known as Luminant Generation Company LLC) (the licensee), requested changes to the technical specifications (TSs) for Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2 (CPNPP). Specifically, the licensee requested changes to the TSs consistent with Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015 (ADAMS Accession No. ML15294A555). The supplement dated June 30, 2016, 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 July 19, 2016 (81 FR 46963).

The licensee's proposed changes delete CPNPP TS 5.5.8, "Inservice Testing Program," and adds a new defined term, "INSERVICE TESTING PROGRAM," to the TSs. All existing references to the "Inservice Testing Program" in the CPNPP TS SRs are replaced with "INSERVICE TESTING PROGRAM" so that the SRs refer to the new definition in lieu of the deleted program. The licensee also proposed several variations from the TS changes described in TSTF-545, Revision 3, including TS format changes.

2.0 REGULATORY EVALUATION

2.1 Description of Inservice Testing Requirements and TSTF-545

An inservice test is a test to assess the operational readiness of a structure, system, or component after first electrical generation by nuclear heat. The American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) provides requirements for inservice testing of certain components in light-water nuclear power plants. The ASME OM Code identifies the components subject to the testing (i.e., pumps, valves, pressure relief devices, and dynamic restraints), responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating results, corrective actions, personnel qualification, and recordkeeping. Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(f), "Inservice testing requirements," requires that inservice testing of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME OM Code and applicable addenda. The facility's TSs also prescribe inservice testing requirements and frequencies for ASME Code Class 1, 2, and 3 components.

The regulation in 10 CFR 50.55a(f)(5)(ii) states, in part, "If a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program." TSTF-545, Revision 3, provides guidance to licensees on how to request license amendments that would eliminate conflicting requirements between 10 CFR 50.55a, "Codes and standards," and the TSs. TSTF-545, Revision 3, proposes elimination of the Inservice Testing Program from the Administrative Controls section of the TSs. The TSs contain surveillances that require testing or test intervals in accordance with the Inservice Testing Program. The elimination of the Inservice Testing Program from the TSs could cause uncertainty regarding the correct application of these SRs. Therefore, TSTF-545, Revision 3, also proposes adding a new definition, "INSERVICE TESTING PROGRAM," to the TSs, which would be defined as "the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." TSTF-545, Revision 3, proposes replacement of existing uses of the term, "Inservice Testing Program," with the defined term, as denoted by capitalized letters, throughout the TSs.

The NRC approved TSTF-545, Revision 3, by letter dated December 11, 2015 (ADAMS Package Accession No. ML15317A071), and published a notice of availability in the *Federal Register* (FR) on March 28, 2016 (81 FR 17208).

2.2 Proposed Technical Specifications Changes

The licensee requested to delete TS 5.5.8 from the Administrative Controls section of TSs and replace it with the word "Deleted." TS 5.5.8 currently states:

> 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.

SR 3.0.2 allows an extension of inservice testing intervals by up to 25 percent. If it is discovered that a surveillance associated with an inservice testing activity was not performed within the required interval, SR 3.0.3 allows the licensee to delay declaring the associated limiting condition for operation not met in order to perform the missed surveillance. The licensee did not request changes to SR 3.0.2 or SR 3.0.3.

The licensee requested to revise the Definitions section of TSs by adding the term, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The licensee also requested that all existing occurrences of "Inservice Testing Program" in TS SRs be replaced with "INSERVICE TESTING PROGRAM," so that the SRs refer to the new definition in lieu of the deleted program.

2.3 Regulatory Requirements and Guidance

The NRC staff considered the following regulatory requirements, guidance, and licensing information during its review of the proposed changes:

Technical Specifications

Paragraph 50.36(c) of 10 CFR requires TSs to include the following categories: (1) safety limits, limiting safety systems settings, and control settings; (2) limiting conditions for operation; (3) SRs; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. Section 50.36(c)(3) of 10 CFR states that "[s]urveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." Section 50.36(c)(5) of 10 CFR states 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."

The NRC staff's guidance for review of the TSs is in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Chapter 16, "Technical Specifications," Revision 3, dated March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the staff has prepared improved STSs for each of the LWR nuclear steam supply systems and associated balance-of-plant equipment systems. The licensee's proposed amendment is based on TSTF-545, Revision 3, which is an NRC-approved change to the improved STSs. The staff's review includes consideration of whether the proposed changes are consistent with TSTF-545, Revision 3. Special attention is given to TS provisions that depart from the improved STSs, as modified by NRC-approved TSTF travelers, to determine whether proposed differences are justified by uniqueness in plant design or other considerations so that 10 CFR 50.36 is met. In addition, the guidance states that comparing the change to previous STS can help clarify the TS intent.

Inservice Testing

Pursuant to 10 CFR 50.54, "Conditions of licenses," the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. These requirements include inservice testing of pumps and valves at nuclear power reactors in accordance with the ASME OM Code as specified in 10 CFR 50.55a(f). The regulations in 10 CFR 50.55a(f) state, in part:

> Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV [Boiler and Pressure Vessel] Code and ASME Code for Operation and Maintenance of Nuclear Power Plants as specified in this paragraph. Each operating license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions [referring to 10 CFR 50.55a(f)(1) through (f)(6)]. . . .

The ASME OM Code is a consensus standard, which is incorporated by reference into 10 CFR 50.55a. During the incorporation process, the NRC staff reviewed the ASME OM Code requirements for technical sufficiency and found that the ASME OM Code inservice testing program requirements were suitable for incorporation into the NRC's rules.

The regulation in 10 CFR 50.55(a)(f)(5)(ii) states, in part: "If a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program."

NUREG-1482, Revision 2, "Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants," Final Report, October 2013 (ADAMS Accession No. ML13295A020) provides guidance for the inservice testing of pumps and valves.

NUREG-0800, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," Revision 3, March 2007 (ADAMS Accession No. ML070720041), provides guidance and acceptance criteria for the NRC staff review of the inservice testing program for pumps and valves.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the guidance, regulations, and licensing information discussed in Section 2.3 of this safety evaluation. In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Among the considerations are whether the TSs, as amended, would provide the necessary administrative controls per 10 CFR 50.36(c)(5) (i.e., provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner). In making its determination as to whether to amend the license, the staff considered those regulatory requirements that are automatically conditions of the license through 10 CFR 50.54. Where the regulations already condition the license, there is no need for a duplicative requirement in the TSs; the regulations provide the necessary reasonable assurance of the health and safety of the public.

3.1 Deletion of the Inservice Testing Program from the TSs

TS 5.5.8 requires the licensee to have an inservice testing program that provides controls for inservice testing of ASME Code Class 1, 2, and 3 components (i.e., pumps and valves). Through 10 CFR 50.54, the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. These requirements include 10 CFR 50.55a(f), which specifies the requirements for the inservice testing of pumps and valves. Therefore, requiring the licensee to have an inservice testing program in TSs is duplicative of the license condition in 10 CFR 50.54. Thus, with the proposed TS changes, the licensee will still be required to maintain an inservice testing program in accordance with the ASME OM Code, as specified in 10 CFR 50.55a(f). For the reasons explained below, it is not necessary to have additional administrative controls in the TSs relating to the inservice testing program to assure operation of the facility in a safe manner.

Consideration of TS 5.5.8.a

The ASME OM Code requires testing to normally be performed within certain time periods. TS 5.5.8.a sets inservice testing frequencies more precisely than those specified in the ASME OM Code and applicable addenda (e.g., "at least once per 31 days" contrasted with "monthly"). However, the NRC staff determined that the more precise inservice testing frequencies are not necessary to assure operation of the facility in a safe manner.

Consideration of TS 5.5.8.b

TS 5.5.8.b allows the licensee to extend, by up to 25 percent, the interval between inservice testing activities, as required by TS 5.5.8.a and for other normal and accelerated frequencies specified as 2 years or less in the inservice testing program. Similar to TS 5.5.8.b, the NRC authorization of ASME Code Case OMN-20, "Inservice Test Frequency," by letter dated

January 19, 2016 (ADAMS Accession No. ML16011A073), also permits the licensee to extend the inservice testing intervals specified in the ASME OM Code by up to 25 percent.

The NRC staff determined that the TS 5.5.8.b allowance to extend inservice testing intervals is not needed to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that deletion of TS 5.5.8.b is acceptable. The deletion of TS 5.5.8.b does not impact the licensee's ability to extend inservice testing intervals using Code Case OMN-20, as authorized by the NRC.

Consideration of TS 5.5.8.c

TS 5.5.8.c allows the licensee to use SR 3.0.3 when it discovers that an SR associated with an inservice test was not performed within its specified frequency. SR 3.0.3 allows the licensee to delay declaring a limiting condition for operation not met in order to perform the missed surveillance. The use of SR 3.0.3 for inservice tests is limited to those inservice tests required by an SR. In accordance with 10 CFR 50.55a, the licensee may also request relief from the ASME OM Code requirements to address issues associated with a missed inservice test. Deletion of TS 5.5.8.c does not change any of these requirements, and SR 3.0.3 will continue to apply to those inservice tests required by SRs. Based on the above, the NRC staff determined that deletion of TS 5.5.8.c is acceptable.

Consideration of TS 5.5.8.d

TS 5.5.8.d states that nothing in the ASME OM Code shall be construed to supersede the requirements of any TS. However, the regulations in 10 CFR 50.55a(f)(5)(ii) address what to do if a revised inservice testing program for a facility conflicts with the TSs for the facility; they require the licensee to apply for an amendment to the TSs to conform the TSs to the revised program at least 6 months prior to the start of the period for which the provisions become applicable. Accordingly, there is no need for a TS stating how to address conflicts between the TSs and the inservice testing program because the regulations specify how conflicts must be resolved.

Conclusion Regarding Deletion of TS 5.5.8

The NRC staff determined that the requirements currently in TS 5.5.8 are not necessary to assure operation of the facility in a safe manner. Based on this evaluation, the staff concludes that deletion of TS 5.5.8 from the licensee's TSs is acceptable, because TS 5.5.8 is not required by 10 CFR 50.36(c)(5).

3.2 Definition of INSERVICE TESTING PROGRAM and Revision to SRs

The licensee proposes to revise the TS Definitions section to include the term, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The proposed definition of the INSERVICE TESTING PROGRAM is consistent with the definition in TSTF-545, Revision 3. The definition is acceptable to the NRC staff because it correctly refers to the inservice testing requirements in 10 CFR 50.55a(f).

The licensee requested that all existing references to the "Inservice Testing Program" in SRs be revised to "INSERVICE TESTING PROGRAM" to reference the new TS defined term in lieu of the deleted program. The proposed change is consistent with the intent of TSTF-545, Revision 3, to replace the current references in SRs with the new definition. The NRC staff verified that for each SR reference to the "Inservice Testing Program," the licensee proposed to change the reference to "INSERVICE TESTING PROGRAM." The proposed change does not alter how the SR testing is performed. However, the inservice testing frequencies could change because the TSs will no longer include the more precise test frequencies in TS 5.5.8.a. As discussed in Section 3.1 of this safety evaluation, the staff determined that the TSs do not need to include the more precise testing frequencies currently in TS 5.5.8.a. Based on its review, the staff determined that revising the SRs to refer to the new definition is acceptable because these SRs will continue to be performed in accordance with the requirements of 10 CFR 50.55a(f). The staff also determined that, with the proposed changes that allow less-precise testing frequencies, 10 CFR 50.36(c)(3) will continue to be met because the SRs will continue to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

3.3 Deviations from TSTF-545

In its application, the licensee identified the following deviations from TSTF-545, Revision 3:

1. TSTF-545, Revision 3, completely deletes TS 5.5.8 from the TSs and renumbers the subsequent TS programs. The licensee proposes to delete the content of TS 5.5.8, but retains the TS number, and adds the word "Deleted." The licensee did not propose to renumber the subsequent TS programs.

2. SR 3.7.4.1 and SR 3.7.4.2 are not included in TSTF-545, Revision 3. However, the licensee stated that these SRs are included since the CPNPP TS differ from the STS.

The NRC staff finds that the proposed deviations are editorial in nature and the licensee's proposed TS changes remain consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendments on March 13, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes SRs. 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 July 19, 2016 (81 FR 46963). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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