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

November 30, 2018

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 AMENDMENTS REGARDING REVISION TO TECHNICAL SPECIFICATIONS FOR ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (EPID L-2018-LLA-0081)

Dear Mr. Peters:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 171 to Facility Operating License No. NPF-87 and Amendment No. 171 to Facility Operating License No. NPF-89 for Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2 (CPNPP), respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 29, 2018.

The amendments revise CPNPP TS 3.3.2, Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation," by adding a footnote to the Applicable Mode Column for MODE 2. Specifically, this change modifies the MODE 2 applicability of Function g, "Trip of all Main Feedwater Pumps," in CPNPP Table 3.3.2-1, so that auxiliary feedwater (AFW) actuation is only required to be OPERABLE when one or more main feedwater pump(s) are supplying feedwater to the steam generators. This proposed change limits the potential for a low power, overcooling transient due to inadvertent AFW actuation when it is neither required by plant conditions nor desired for stable plant control.

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,

MOBanion

Margaret W. O'Banion, 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. 171 to NPF-87
2. Amendment No. 171 to NPF-89
3. Safety Evaluation

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. 171
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 March 29, 2018, 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. 171 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

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: November 30, 2018

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. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
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 March 29, 2018, 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. 171 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

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: November 30, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 171

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 171

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
3.3-33	3.3-33

(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) Maximum Power Level

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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 171 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. 171 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

Table 3.3.2-1 (page 5 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[(a)]
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1, 2, 3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Not Used.					
c. SG Water Level Low-Low	1, 2, 3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥37.5% of narrow range span (Unit 1) [(q)(r)] ≥34.9% of narrow range span (Unit 2) [(q)(r)]
d. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
e. Loss of Offsite Power	1, 2, 3	1 per train	F	SR 3.3.2.7 SR 3.3.2.9 SR 3.3.2.10	NA
f. Not Used.					
g. Trip of all Main Feedwater Pumps	1, 2[(d)]	2 per AFW pump	J	SR 3.3.2.8	NA
h. Not Used.					

(a) The Allowable Value defines the limiting safety system except for functions 5b and 6c (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

(d) When one or more Main Feedwater Pump(s) are supplying feedwater to steam generators.

(q) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(r) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint, the methodology used to determine the as-found tolerance and the methodology used to determine the as-left tolerance shall be specified in the Technical Specification Bases.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 171 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 171 TO

FACILITY OPERATING LICENSE NO. NPF-89

COMANCHE PEAK POWER COMPANY LLC

AND VISTRA OPERATIONS COMPANY LLC

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

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By license amendment request (LAR) dated March 29, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18102A516), Vistra Operations Company LLC (Vistra OpCo, the licensee) requested changes to the Technical Specifications (TSs) for Comanche Peak Nuclear Power Plant (CPNPP), Unit Nos. 1 and 2.

The amendments would revise CPNPP TS 3.3.2, Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation," by adding a footnote to modify the MODE 2 applicability of Function g, "Trip of all Main Feedwater Pumps," so that auxiliary feedwater (AFW) actuation is only required to be OPERABLE when one or more main feedwater (MFW) pump(s) are supplying feedwater to the steam generators. This proposed change limits the potential for a low power, overcooling transient due to inadvertent AFW actuation when it is neither required by plant conditions nor desired for stable plant control.

On July 9, 2018, a public teleconference was held between the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff and representatives of Vistra OpCo regarding the application dated March 29, 2018, for the purpose of seeking clarification on the proposed change, including changes to or introduction of new operator actions. For additional details, a summary of the public meeting dated August 2, 2018, is located under ADAMS Accession No. ML18194A777.

2.0 REGULATORY EVALUATION

The three subsections below describe: (1) the system being changed, (2) the proposed changes, and (3) the regulatory requirements and guidance applicable to the changes.

2.1 System Description

The feedwater system automatically provides feedwater to the steam generators during steady state and transient conditions. The AFW system is designed to provide a secondary side heat sink for the reactor in the event that the normal feedwater system is not available, for example, during plant cooldown, and startup operations. The AFW system has two motor-driven pumps and one turbine-driven pump. The normal source of water for the AFW system is the condensate storage tank. Three AFW trains are required to be OPERABLE in MODES 1, 2, and 3[1] in accordance with CPNPP TS 3.7.5, "Auxiliary Feedwater (AFW) System."

In Section 3.0, "Technical Evaluation," of the application, the licensee states that the design basis events (DBEs) that impose AFW safety function requirements are: (1) loss of all alternating current (AC) power to plant auxiliaries, (2) loss of normal feedwater, (3) steam generator fault in either the feedwater or steam lines, and (4) small break loss-of-coolant accidents. These DBEs assume AFW starts automatically on a low-low steam generator water level, station blackout, or safety injection. The licensee further states that the anticipatory AFW actuation function on a trip of all MFW pumps, which corresponds to Function 6.g of Table 3.3.2-1, is not credited in the accident analysis.

A trip of all MFW pumps is an indication of a loss of normal feedwater and the subsequent need for some method of heat removal to bring the reactor back to no load temperature and pressure. Each turbine-driven MFW pump is equipped with two pressure switches (one in Train "A" and one in Train "B") on the oil line for the speed control system. A trip signal from both MFW pumps anticipatory trip circuit would actuate the motor-driven AFW pumps to ensure that at least one steam generator is available with a water supply to act as the heat sink for the reactor.

2.2 Proposed TS Changes

The proposed changes would add a new footnote "d" to Function 6.g of TS Table 3.3.2-1. Footnote "d" modifies MODE 2 applicability so that the AFW actuation is required "[w]hen one or more Main Feedwater Pump(s) are supplying feedwater to steam generators."

Section 2.1, "System Design and Operation," of the LAR dated March 29, 2018, describes the current operation for entering MODE 2, then transitioning to MODE 1 at CPNPP. Currently, when entering MODE 2, the AFW system is in service to control and maintain steam generator water level through motor-driven AFW pumps. At approximately 2 percent rated thermal power (RTP), a MFW pump is reset and placed into service. When the flow from that MFW pump is sufficient to maintain steam generator water level, the AFW pumps are placed in standby. Currently, during the process of placing one MFW pump in service, the non-operating MFW pump is tripped and the pressure switches, which actuate the motor-driven AFW automatic start on loss of both MFW pumps, are isolated and vented (depressurized), thereby placing the anticipatory AFW automatic start circuit in a half trip condition (one-out-of-two inputs

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[1] MODE 1 is defined as Power Operations, greater than 5 percent rated thermal power. MODE 2 is defined as Startup, less than or equal to 5 percent rated thermal power. MODE 3 is defined as Hot Standby with reactor coolant temperature greater and equal to 350 degrees Fahrenheit.

to the start logic satisfied). If the operating MFW pump was to trip during this period, an AFW automatic start would send a start signal to both motor-driven AFW pumps and cause the flow control valves to the steam generators to "trip-to-auto," which would open them to 100 percent demand (full open). This would result in the potential for a low power, overcooling transient. Assuming the operating MFW pump did not trip during this period, after entering MODE 1, the second MFW pump is reset and placed into service (approximately 50 percent RTP), and the anticipatory AFW automatic start circuit is no longer in half trip condition (i.e., logic is restored to two-out-of-two).

2.3 Applicable Regulatory Requirements and Guidance

The NRC's requirements related to the content of the TSs are contained in Section 50.36, "Technical specifications," to Title 10 of the *Code of Federal Regulations* (10 CFR). The regulations in 10 CFR 50.36(c) require that the TSs include limiting conditions for operation. As specified in 10 CFR 50.36(c)(2)(i), "[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility."

The regulations at 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," include requirements that are related to the acceptable performance of the emergency core cooling systems (ECCS) designs.

Section 3.1, "Conformance with Nuclear Regulatory Commission (NRC) General Design Criteria," of the CPNPP Final Safety Analysis Report (FSAR), describes an evaluation of the design bases of the CPNPP as measured against the NRC general design criteria (GDCs) for nuclear power plants in Appendix A of 10 CFR Part 50. The following 10 CFR Part 50, Appendix A GDCs are listed in the CPNPP FSAR and are related to this LAR.

- GDC 10 – Reactor Design

  The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- GDC 15 – Reactor Coolant System Design

  The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- GDC 20 – Protection System Functions

  The protection systems shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, and to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- GDC 22 – Protection System Independence

  The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation shall be used to the extent practical to prevent loss of the protection function.

- GDC 29 – Protection Against Anticipated Operational Occurrences

  The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

The NRC staff also applied the review guidance in Section 18.0, Revision 3, "Human Factors Engineering" of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," dated December 2016 (ADAMS Accession No. ML16125A114), and NUREG-1764, Revision 1, "Guidance for the Review of Changes to Human Actions," dated September 2007 (ADAMS Accession No. ML072640413), to determine the appropriate level of human factors engineering review necessary for the requested changes.

3.0 TECHNICAL EVALUATION

3.1 Compliance with GDC 10, 15, and 10 CFR 50.46 Requirements

Chapter 15, "Accident Analysis," of the CPNPP FSAR discusses the licensee's analyses of the DBEs. The analyses consider the AFW safety function for consequence mitigation to meet the requirements of the following:

- GDC 10, which requires the plant to be designed not to exceed the specified acceptable fuel design limits (SAFDLs) during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs);

- GDC 15, which requires the plant to be designed to assure that the reactor coolant pressure boundary (RCPB) integrity is maintained during any condition of normal operation, including the effects of AOOs; and

- Section 50.46 of 10 CFR, which requires the acceptable performance of the ECCS designs for the DBEs.

The applicable DBEs include a loss of all AC power to plant auxiliaries, a loss of normal feedwater, main steam line breaks, MFW line breaks, steam generator tube ruptures and small-break loss-of-coolant accidents. The AFW actuation signals credited in the existing DBE analyses are low-low steam generator water level, safety injection, or loss of offsite power. None of the analyses for the DBEs assume AFW actuation based on the trip of both MFW pumps; therefore, the NRC staff finds that the proposed changes related to the AFW actuation on the trip of both MFW pumps would not affect the results of the existing DBE analyses. The existing analyses remain valid and continue to meet the requirements of GDC 10 for SAFDLs, GDC 15 for RCPB integrity, and 10 CFR 50.46 for the acceptable performance of the ECCS designs.

3.2 Compliance with GDC 20, 22, and 29 Requirements

The actuation of the motor-driven AFW pumps is an anticipatory actuation and is not quantitatively credited in the CPNPP FSAR, Chapter 15 analyses. It is only the turbine-driven AFW pump that is quantitatively (i.e., outcomes are calculated) credited. Anticipatory trips are qualitatively credited in that they provide defense-in-depth and additional margin. For this proposed change, the anticipatory actuation of the motor-driven AFW pumps on the trip of both MFW pumps is not needed before the MFW pumps are supplying water to the steam generator (i.e., motor-driven AFW pumps are already running and being controlled by an operator).

Per GDC 20, the automatic initiation of an appropriate protective system must assure SAFDLs are not to be exceeded as a result of AOOs. For this proposed TS change, the automatic start of the turbine-driven AFW pump on low-low steam generator water level assures SAFDLs are maintained when normal feedwater flow is lost.

Per GDC 22, the protection system shall be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function. For this proposed TS change, the credited actuation function corresponding to the steam generator low-low water level will actuate the turbine-driven AFW pumps to prevent loss of the heat sink protective function.

Per GDC 29, the engineered safety feature actuation system must maintain functionality in the event of AOOs (when the protective function is needed). The proposed TS change would remove the auto-start function from the anticipatory motor-driven AFW actuation circuit, while the motor-driven AFW pumps are feeding the steam generators. In the event of an AOO, the steam generator low-low water level for the turbine-driven AFW pumps will actuate.

The NRC staff finds that the proposed change to the operability requirements of the anticipatory actuation of the motor-driven AFW pumps does not affect compliance with the GDC 20, 22, and 29. Thus, the staff determined that the requirements of GDC 20, 22, and 29 will continue to be met.

3.3 Human Factors Review

The NRC staff used the guidance in Section 18.0, Revision 3, of NUREG-0800 and NUREG-1764, Revision 1, to determine the appropriate level of human factors engineering review necessary for the requested changes. NRC staff review of the proposed changes and CPNPP FSAR, Chapter 15 analyses determined that the LAR does not introduce new tasks or changes to risk-important operator actions. Although the timing of operator actions during plant startup to enable the AFW system to auto-start upon a loss of both MFW pumps is affected by the proposed change, the changes enhance the ability of the operators to safely control the plant due to the reduced likelihood of an inadvertent overcooling event during startup. Additionally, the skills and abilities required for appropriate operator action are similar to those required for existing tasks performed frequently by operators (e.g., alarm response, use of abnormal operating procedures). The staff determined that there are no significant changes to risk-important operator actions introduced in this LAR; therefore, no detailed human factors evaluation is required.

3.4 Compliance with 10 CFR 50.36 Requirements

In the LAR dated March 29, 2018, the licensee requests that the anticipatory AFW automatic start function (on trip of all MFW pumps) not be required to be OPERABLE prior to the MFW pump establishing sufficient feed flow to maintain steam generator water level (i.e., only OPERABLE when the event it protects against is possible, meaning only OPERABLE when the loss of MFW supplying the steam generators is possible).

Section 3 of the LAR describes that, if actuated, the automatic start function sends a start signal to both motor-driven AFW pumps and causes the flow control valves to the steam generators to "trip-to-auto," which opens them to 100 percent demand. This 100 percent demand flow may not be prudent, when the motor-driven AFW pumps are already feeding the steam generators with flow controlled by the operator, because it could cause a low power, overcooling transient. It is preferable for automatic start of the motor-driven AFW pumps to be OPERABLE only when the event it protects against (i.e., trip of all MFW pumps) is possible. Since anticipatory AFW automatic start function is always required to be OPERABLE when the event it protects against is possible, the NRC staff determined the proposed TS changes continue to meet the requirements of 10 CFR 50.36(c)(2)(i).

3.5 NRC Staff Conclusion

The NRC staff concludes that the proposed changes that add new footnote "d" to Function 6.g. (i.e., actuation of motor-driven AFW pumps on trip of all MFW pumps) of Table 3.3.2-1 is acceptable because it reduces the likelihood of inadvertent overcooling during startup. The proposed changes prevent the automatic start of all AFW pumps with flow control valves opening to 100 percent demand (full open) while the flow is being controlled by the operator prior to the MFW pumps supplying water to the steam generators during MODE 2. Additionally, either operator control or the automatic low-low steam generator water level signal would start the AFW pumps if needed to ensure an adequate secondary side heat sink. In addition, the staff finds that the proposed changes affect the timing of operator actions, which enhances the ability of the operator to safety control the plant. The staff finds that the proposed TS changes meet the regulatory requirements described in 10 CFR 50.36, 10 CFR 50.46, and GDCs 10, 15, 20, 22, and 29. Therefore, the staff concludes that the proposed 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 November 26, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to 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 public comment on such finding published in the *Federal Register* on June 5, 2018 (83 FR 26107). 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.

Principal Contributors: H. Akhavannik, NRR
N. Carte, NRR
S. Sun, NRR
M. Montecalvo, NRR

Date: November 30, 2018

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING REVISION TO TECHNICAL SPECIFICATIONS FOR ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (EPID L-2018-LLA-0081) NOVEMBER 30, 2018

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OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DE/EICB/BC(A)*	NRR/DSS/STSB/BC
NAME	MO'Banion	PBlechman	MWaters (RAlvarado for)	VCusumano
DATE	11/01/18	11/01/18	11/02/18	11/06/18
OFFICE	NRR/DSS/SRXB/BC*	NRR/DSS/SCPB/BC*	NRR/DRA/APOB/BC*	OGC
NAME	JWhitman (RBeaton for)	SAnderson	CFong	MYoung
DATE	07/24/18	10/31/18	11/27/18	11/28/18
OFFICE	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM		
NAME	RPascarelli	MO'Banion		
DATE	11/30/18	11/30/18		

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