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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Ref 10 CFR.90
10 CFR.91

Subject: Comanche Peak Nuclear Power Plant (CPNPP)
Docket Nos. 50-445 and 50-446
LICENSE AMENDMENT REQUEST (LAR) 23-004 TECHNICAL SPECIFICATIONS (TS) 3.9.3, "NUCLEAR INSTRUMENTATION"

Dear Sir or Madam:

Pursuant to 10 CFR 50.90 and 10 CFR 50.91, Vistra Operations Company LLC (Vistra OpCo) hereby requests an amendment to the Technical Specifications for Comanche Peak Nuclear Power Plant Units 1 and 2 (CPNPP).

The proposed amendment deletes TS 3.9.3, "Nuclear Instrumentation," and relocates the content to the Technical Requirements Manual.

The enclosure provides a description and assessment of the proposed changes. Attachment 1 provides the existing TS pages marked to show the proposed change. Attachment 2 provides revised (clean) TS pages. Attachment 3 provides the existing TS Bases pages marked to show the proposed change for information only.

Vistra OpCo has determined that the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92(c), and there are no significant environmental impacts associated with the change. The CPNPP Station Operations Review Committee (SORC) has reviewed the proposed license amendment. In accordance with 10 CFR 50.91(b)(l), a copy of the proposed license amendment is being forwarded to the State of Texas.

NRC staff review and approval of the proposed license amendment is requested within one year of the NRC acceptance date. Once approved, the amendment shall be implemented within 90 days.

This communication contains no new regulatory commitments regarding CPNPP Units 1 and 2.

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Should you have any questions, please contact Kris Brigman at (254) 266-3237 or Kristopher.Brigman@luminant.com.

I state under penalty of perjury that the foregoing is true and correct.

Executed on November 20, 2023.

Sincerely,

null Jay (Nov 17, 2023 07:49 CST)

Jay J. Lloyd

Enclosure: LICENSE AMENDMENT REQUEST (LAR) 23-004 TECHNICAL SPECIFICATION 3.9.3, "NUCLEAR INSTRUMENTATION"

Attachments:
1. PROPOSED TECHNICAL SPECIFICATION CHANGES (MARKUP)
2. PROPOSED TECHNICAL SPECIFICATION CHANGES (CLEAN)
3. PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (MARKUP - FOR INFORMATION ONLY)

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1.0 SUMMARY DESCRIPTION

License Amendment Request (LAR) 23-004 proposes to relocate Technical Specification (TS) 3.9.3, "Nuclear Instrumentation" for Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 (Reference 6.1) to the Technical Requirements Manual (TRM) (Reference 6.2). As described in TS 5.5.17, the TRM contains selected requirements which do not meet the criteria for inclusion in the Technical Specifications but are important to the operation of CPNPP. Changes to the TRM are controlled by 10 CFR 50.59, and require the approval of the Plant Manager.

Vistra Operations Company (Vistra OpCo) is proposing this change to the CPNPP TS on the basis of the Nuclear Regulatory Commission's (NRC) "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132), dated July 22, 1993 (Reference 6.3), and the evaluation contained in WCAP-11618, "Methodically Engineered, Restructured, and Improved Technical Specifications, MERITS Program – Phase II Task 5, Criteria Application" (Reference 6.4). Based on that evaluation, TS 3.9.3 does not satisfy any of the criteria of 10CFR50.36(c)(2)(ii) and can be relocated out of the TS to a licensee-controlled document.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

TS 3.9.3 addresses nuclear instrumentation requirements for Mode 6. The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. Any two of the four source range neutron flux monitors can be used to satisfy the LCO.

The installed Westinghouse source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). The installed source range neutron flux monitors are boron trifluoride (BF_3) detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second and cover six decades of neutron flux. The detectors provide continuous visual indication in the control room.

A separate Gamma-Metrics Neutron Flux Monitoring System (NFMS) monitors the neutron flux from the source range through 200% Rated Thermal Power (RTP) during all modes of plant operation. This system utilizes two separate Safety Category I (Class 1E) fission chamber neutron detectors for all ranges of neutron flux indication.

2.2 Current Technical Specification Requirements

LCO 3.9.3 requires that two source range neutron flux monitors be operable in Mode 6 to ensure that redundant monitoring capability is available to detect changes in core reactivity.

2.3 Reason for Proposed Change

LCO 3.9.3 does not satisfy the criteria for inclusion in the TS per 10CFR50.36(c)(2)(ii) as discussed in Section 3.0, therefore TS 3.9.3 is proposed to be relocated to the TRM, consistent with TS 5.5.17.

2.4 Description of Proposed Change

The changes requested by this amendment application are discussed below:

- TS 3.9.3 will be deleted in its entirety and relocated to the TRM.
- Page 3.9-4 will state "TS 3.9.3 deleted."
- Page 3.9-5 will state "This page intentionally left blank."

3.0 TECHNICAL EVALUATION

The NRC's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Reference 6.3) provided a specific set of four objective criteria to determine which of the design conditions and associated surveillances should be included in the TSs as limiting conditions for operations (LCOs). The Final Policy Statement noted the application of these criteria, which were subsequently included in a revision to 10 CFR 50.36, would result in some requirements in TSs to no longer be included in the TSs.

<u>TS 3.9.3 Bases</u>

Currently, the TS 3.9.3 Bases states the following relevant information.

In the Applicable Safety Analyses Section:

"Two OPERABLE source range neutron flux monitors are required to provide a visual signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident. (Ref. 2)

The source range neutron flux monitors satisfy Criterion 3 of 10CFR50.36(c)(2)(ii)."

The phrase "*or an improperly loaded fuel assembly*" was deleted from the CPNPP TS 3.9.3 Bases utilizing the guidance contained in Technical Specifications Task Force Traveler TSTF-555-T, "Clarify the Nuclear Instrumentation Bases Regarding the Detection of an Improperly Loaded Fuel Assembly," (Reference 6.12). The change corrected an error in the

TS Bases. The Bases stated that the source range neutron flux monitors can detect an improperly loaded fuel assembly; however, this statement is neither consistent with the licensing basis nor supported by operating experience or reactor physics. There is no licensing basis analysis assumption to detect misloaded fuel assemblies in Mode 6 with the source range detectors. The Bases change was made in accordance with Specification 5.5.14, “Technical Specification (TS) Bases Control Program.” This LAR is not requesting NRC review of the TS Bases correction NOR review of TSTF-555.

The Bases states that the source range neutron flux monitors satisfy Criterion 3 based on a boron dilution accident. However, as discussed in UFSAR Section 15.4.6 below, this statement is not correct, and will be deleted from the Bases after it is relocated to the TRM in accordance with the TS Bases Control Program.

In the Applicability Section:

> *“In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In other MODES, the NIS source range monitors are governed by LCO 3.3.1.”*

The capability to monitor the core during refueling activities will be maintained after TS 3.9.3 is relocated to the TRM. Relocating TS 3.9.3 to the TRM does not eliminate the source range neutron flux monitors and maintains the capability to monitor core changes in reactivity. However, this capability does not satisfy any of the criteria of 10CFR50.36(c)(2)(ii).

<u>UFSAR Section 15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT</u>

In Mode 6, the shutdown margin is maintained by the Reactor Coolant System (RCS) boron concentration which is controlled by TS 3.9.1, “Boron Concentration.” A significant reduction in shutdown margin can occur due to a boron dilution event, and the source range instrumentation controlled by TS 3.9.3 can provide an indication to the control room operator of a loss of shutdown margin due to a boron dilution event. However, plants which preclude a boron dilution event by administrative controls of the sources of unborated water which connect to the RCS do not analyze a boron dilution event, and the source range instrumentation is therefore not required to provide this mitigating function.

Subsection 15.4.6.2 of the UFSAR states:

> “*<u>Dilution During Refueling</u>*
>
> *An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls which isolate the RCS from*

the potential source of unborated water. Either valve 1,2CS-8455 or valves 1,2CS-8560, 1,2FCV-111B, 1,2CS-8441, 1,2CS-8453 and 1,2CS-8439 in the CVCS will be locked closed during refueling operations. These valves block all flow paths that could allow significant rates of unborated makeup water to reach the RCS. Any makeup which is required during refueling will be borated water supplied from the RWST."

At CPNPP, the control of unborated water sources is provided by TS 3.9.2, "Unborated Water Source Isolation Valves." The Applicable Safety Analyses section of the Bases for TS 3.9.2 states:

"By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6."

This evaluation is consistent with WCAP-11618 (Reference 6.4). The WCAP TS Screening Form for STS 3/4.9.2, "Instrumentation," Section (3) Discussion on page 3-205 states:

*"For those plants which use administrative control to preclude boron dilution accidents, the source range neutron flux monitors are not part of the primary success path which function to mitigate a DBA [Design Basis Accident] or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier**. The source range neutron flux monitors do not satisfy criterion 3 for these plants.**"*

Subsequent to the publishing of the WCAP in 1987, NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 0 (Reference 6.5) was issued in 1992 which included LCO 3.9.2, "Unborated Water Source Isolation Valves." This LCO incorporated the administrative controls credited in WCAP-11618 into the TS requirements applicable in Mode 6. The Applicable Safety Analyses Section of the Bases for LCO 3.9.2 in NUREG-1431 contains the same statement regarding not requiring a boron dilution analysis in Mode 6 that is discussed above in the Comanche Peak Bases for TS 3.9.2. Therefore, the source range instrumentation addressed by LCO 3.9.3 does not provide any required mitigation of an analyzed accident, since the boron dilution event is precluded by LCO 3.9.2 and is therefore not analyzed.

In Revision 2 of NUREG-1431 (Reference 6.6), a Reviewer's Note was added to LCO 3.9.2 which states:

"This Technical Specification is not required for units that have analyzed a boron dilution event in MODE 6. It is required for those units that have not analyzed a boron dilution event in MODE 6. For units which have not analyzed a boron dilution event in MODE 6, the isolation of all unborated water sources is required to preclude this event from occurring."

LCO 3.9.2 and the Bases in Revision 5 of NUREG-1431 (Reference 6.7) is the same as it existed in Revision 2 of NUREG-1431.

<u>UFSAR Section 15.4.7 "Inadvertent Loading and Operation of a Fuel Assembly into an Improper Position"</u>

Subsection 15.4.7.2 of the UFSAR states:

"The Inadvertent Loading Event comprises core misloading scenarios such as the loading of one or more fuel assemblies into improper positions, the loading of a fuel rod during manufacture with one or more pellets of the wrong enrichment, the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment, or events involving burnable absorbers. All of these misloading scenarios potentially result in a core reactivity distribution that differs from the intended core reactivity distribution. As a result, the core power distribution and peaking factors may differ from predictions. Specifically, misloading errors can lead to increased local power peaking at the location of the misloading if the misloading results in a local reactivity increase relative to the intended pattern.

Fuel misloads are prevented by the manufacturing controls employed to build the fuel and the core loading controls used to assemble the core. The manufacturing controls include checks on fuel rod weight to confirm the uranium loading in the fuel rod, active and passive gamma scans of individual fuel rods to confirm fuel enrichments, pellet stack lengths, pellet types, and the absence of pellet gaps during fuel manufacturing, and bar coding of each fuel rod to confirm its proper placement in the fuel assembly.

To reduce the probability of core loading errors during fuel loading, each fuel assembly and core component is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the fuel identification numbers are checked before each assembly is moved into the core. Identification numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed. The correct type of insert is also confirmed for each location. These procedures make the likelihood of core misloadings very small.

…Should misloadings occur, the incore system of movable flux detectors, which is used to verify power distributions during startup and throughout the operating cycle, is capable of revealing enrichment errors or misloadings which would cause the kind of substantial power distribution perturbation that would be necessary to induce large numbers of fuel rod failures. In addition, thermocouples and excore detectors can provide additional indications of power distribution anomalies. This instrumentation,

along with the startup testing performed each cycle, make the detection of severe misloadings highly likely."

Subsection 15.4.7.3 states:

"The incore moveable detector system is used to search for potential fuel misloads at the start of each operating cycle. Following fuel loading and low power physics testing, an initial core power distribution measurement is made. The core power level of this initial flux map is typically between ~30% and ~50% of rated thermal power. This initial power distribution measurement is used to confirm that the measured power distribution is consistent with the predicted power distribution. Observed flux map deviations in excess of the flux map review criteria (see Table 1) would prompt an investigation of a possible core anomaly…"

CPNPP FSAR Section 15.4.7 is consistent with WCAP-16676-NP misloaded assembly analysis methodology (Reference 6.11), which provides additional clarifying details regarding the original methodology but does not modify or alter the method relative to the original Westinghouse methodology approved by the NRC for Comanche Peak. This LAR is not requesting NRC review of WCAP-16676-NP. From the WCAP,

"Fuel misloads involving swaps of two fuel assemblies or a single assembly misload are considered; misloads involving more than two assemblies are considered less credible since multiple errors would have to occur. Also, detection of such misloads would be more likely since more assemblies are involved, giving more opportunities for the incore moveable detector system to detect the loading errors."

Based on the above discussions, administrative controls and the moveable incore detector system are credited in the licensing basis analysis of record for the detection of a fuel misload in Mode 1. The source range instrumentation addressed by LCO 3.9.3 in Mode 6 is not part of a primary success path to mitigate a fuel misload. Therefore, the source range instrumentation addressed by LCO 3.9.3 does not provide any required mitigation of a fuel misload accident.

<u>UFSAR Section 15.7.4 "Design Basis Fuel Handling Accidents"</u>

Subsection 15.7.4.1 of the UFSAR states:

"The accident is defined as dropping of a spent fuel assembly in the Containment Building or spent fuel pool fuel storage area floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations."

The source range neutron flux monitors are not part of the primary success path which function to mitigate a dropped fuel assembly. Therefore, the source range instrumentation addressed by LCO 3.9.3 does not provide any required mitigation of a fuel handling accident.

Conclusions

Based upon the above information and application of the criteria in 10 CFR 50.36(c)(2)(ii), the following determination was made with regards to CPNPP TS 3.9.3 “Nuclear Instrumentation” for the source range instrumentation in Mode 6. (Note that clarifications of the four criteria documented below were taken from Reference 6.3.)

<u>Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.</u>

This criterion addresses instrumentation specifically installed to detect excessive RCS leakage. The source range instrumentation addressed by TS 3.9.3 in Mode 6 is not used to monitor degradation of the reactor coolant pressure boundary. Thus, TS 3.9.3 “Nuclear Instrumentation” does not satisfy Criterion 1 for retention in the TS.

<u>Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.</u>

The purpose of this criterion is to capture those process variables that have initial values assumed in the design basis accident and transient analyses, and which are monitored and controlled during power operation. This criterion also includes active design features and operating restrictions needed to preclude unanalyzed accidents and transients.

The source range instrumentation addressed by TS 3.9.3 in Mode 6 does not involve process variables that have initial values assumed in the design basis accident and transient analyses; nor does the source range instrumentation provide any design feature or operating restriction needed to preclude unanalyzed accidents and transients. Therefore, TS 3.9.3 “Nuclear Instrumentation” does not satisfy Criterion 2 for retention in the TS.

<u>Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.</u>

The purpose of this criterion is to capture only those structures, systems, and components that are part of the primary success path of the safety sequence analyses. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function.

The relevant design basis accidents in Mode 6 for CPNPP are a Fuel Handling Accident and a Fuel Misload. The source range instrumentation does not mitigate these events.

Therefore, TS 3.9.3 “Nuclear Instrumentation” does not satisfy Criterion 3 for retention in the TS.

<u>Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.</u>

The purpose of this criterion is to capture only those structures, systems, and components that operating experience or probabilistic risk assessment has shown to be significant to public health and safety consistent with the Commission’s Safety Goal and Severe Accident Policies.

A review of industry operating experience during refueling operations did not identify any examples where the failure of source range instrumentation during refueling operations had a significant adverse effect on public health and safety, nor any event where the source range instrumentation provided mitigation for an event which otherwise might have had a significant adverse effect on public health and safety. This review included a search of the Institute of Nuclear Power Operations (INPO) IRIS operating experience database (Reference 6.8) as well as Curtiss-Wright Scientech NRS database of NRC documents (Reference 6.9). NUREG-1449, “Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States,” (Reference 6.10) was also reviewed for operating experience insights related to refueling operations and the safety significance of the source range instrumentation. Section 2.1.7 discussed reactivity addition industry events; all events for pressurized water reactors (PWRs) in this category were related to boron dilution. There was no discussion of the source range nuclear instrumentation as having any role in these events.

CPNPP does not have a shutdown Probabilistic Risk Assessment (PRA) model for Mode 6. A survey of other Westinghouse NSSS plants did not identify any Mode 6 plant PRA models. A survey of publicly available PRA studies did not identify events occurring during refueling as significant, nor was the failure of the source range instrumentation during refueling identified as being significant to public health and safety. The PRA summary report for the Merits Program (contained in Section 4 of WCAP-11618), did not identify Nuclear Instrumentation as a significant risk contributor on a dominant risk sequence. NUREG-1449 (Reference 6.10) was also reviewed for PRA insights regarding Mode 6, and the safety significance of the source range instrumentation. Section 4.3 discusses the Seabrook PRA for Shutdown Operation which included Mode 6. The summary of the findings did not identify any Mode 6 accident scenarios as significant to the PRA results. Section 4.8 discusses the NRC Shutdown PRA for Surry performed by Brookhaven National Laboratory which included Mode 6. The summary of the findings stated that the core damage frequency for plant operating state R10 (refueling) was two orders of magnitude lower than the dominant operating state. There was no discussion of the source range instrumentation as a significant contributor to PRA results in Mode 6.

Therefore, TS 3.9.3 “Nuclear Instrumentation” does not satisfy Criterion 4 for retention in the TS.

Based on the above evaluations and references, TS 3.9.3 “Nuclear Instrumentation” does not meet any of the four criteria of 10 CFR 50.36(c)(2)(ii) for inclusion in the TS and may be relocated to a licensee-controlled document.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements / Guidance

The proposed change relocates TS 3.9.3, “Nuclear Instrumentation,” to the TRM. No other changes are being made other than relocation.

The NRC provided guidance for the contents of TS in its “Final Policy Statement on Technical Specifications Improvement for Nuclear Power Reactors” (58 FR 39132, July 22, 1993). (Reference 6.3) In particular, the NRC concluded that certain LCOs could be relocated from the TS to licensee-controlled documents and identified criteria to be used to determine the LCOs to be included in the TS. The NRC incorporated revisions to 10 CFR 50.36 to codify and incorporate these criteria.

Section 50.36(c)(2)(ii) of Title 10 of the Code of Federal Regulations contains the requirements for the LCOs that must be in TS. This regulation provides the four criteria that can be used to determine the LCOs that must be included in the TS. A TS LCO must be established for each item meeting one or more of the four criteria.

As discussed in Section 3.0, TS 3.9.3, “Nuclear Instrumentation,” does not meet any of the four criteria of 10 CFR 50.36 (c)(2)(ii) and can be relocated from the TS to a licensee-controlled document.

4.2 Precedent

There are numerous precedent license amendments where the criteria of 10 CFR 50.36(c)(2)(ii) have been applied to a TS LCO as a basis to relocate the requirement to a licensee-controlled document.

- Salem LAR S22-07 to relocate TS requirements for reactor head vents to TRM, Submitted August 31, 2022 and approved by NRC March 13, 2023 EPID L-2022-LLA-0133.
- Waterford LAR W3F1-2021-0004 to relocate Chemical Detection Systems TS to TRM, Submitted April 5, 2021 and approved by NRC April 29, 2022 EPID L-2021-LLA-0061.

4.3 No Significant Hazards Consideration Determination

Vistra OpCo has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

The proposed change is the administrative relocation of a TS LCO that is applicable during Mode 6 to a licensee-controlled document. No actual change to any requirement is being made, so there is no impact to any aspect of Mode 6 activities. The proposed change is administrative in nature and does not change the level of programmatic and procedural control necessary to ensure operation of the facility in a safe manner.

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The change does not involve physical alterations of any plant structures, systems, or components, and is not associated with any accident initiator in Mode 6; therefore, there is no effect on the probability of accidents previously evaluated. The source range instrumentation does not mitigate the consequences of any accident evaluated in Mode 6; therefore, there is no effect on the consequences of accidents previously evaluated.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The change does not impact the design, configuration, or method of operation of the plant during refueling. The proposed change will not alter the manner in which equipment is operated, nor will the functional demands on credited equipment be changed. The proposed changes do not impact the interaction of any systems whose failure or malfunction can initiate an accident during refueling. The proposed changes do not create any new failure modes.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   There is no adverse impact on equipment design or operation and there are no changes being made to the TS required safety limits or safety system settings that would adversely affect plant safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, Vistra OpCo concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATIONS

Vistra OpCo has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

6.0 REFERENCES

6.1 Comanche Peak Unit 1 and Unit 2 Facility Operating Licenses NPF-87 and NPF-89 (ADAMS Accession Numbers ML053180521, ML053180525).

6.2 Technical Requirements Manual (TRM) for Comanche Peak Steam Electric Station Units 1 and 2.

6.3 USNRC, "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132), July 22, 1993.

6.4 WCAP-11618, “Methodically Engineered, Restructured, and Improved Technical Specifications, MERITS Program – Phase II Task 5, Criteria Application,” November 1987 (ADAMS Accession Number 8711240311).

6.5 NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” Revision 0, September 1992 (ADAMS Accession Number ML13196A330).

6.6 NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” Revision 2, April 2001 (ADAMS Accession Number ML011840211 and ML011840219).

6.7 NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” Revision 5, September 2021 (ADAMS Accession Number ML21259A155).

6.8 Institute of Nuclear Power Operations (INPO) IRIS Database (https://iris.inpo.org).

6.9 Curtiss-Wright Scientech NRS Database (https://rs.scientech.com).

6.10 USNRC, NUREG-1449, “Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States,” September 1993 (ADAMS Accession Number ML063470582).

6.11 WCAP- 16676-NP “Analysis Update for the Inadvertent Loading Event” March 2009.

6.12 TSTF-555-T Rev. 0 Technical Specifications Task Force Improved Standard Technical Specifications Change Traveler; “Clarify the Nuclear Instrumentation Bases Regarding the Detection of an Improperly Loaded Fuel Assembly”

Attachment 1

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARKUP)

3.9 REFUELING OPERATIONS

3.9.3 ~~Nuclear Instrumentation~~ DELETED

~~LCO 3.9.3~~ ~~Two source range neutron flux monitors shall be OPERABLE.~~

~~APPLICABILITY:~~ ~~MODE 6.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One required source range neutron flux monitor inoperable.~~	~~A.1 Suspend CORE ALTERATIONS.~~ ~~AND~~ ~~A.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.~~	~~Immediately~~ ~~Immediately~~
~~B. Two required source range neutron flux monitors inoperable.~~	~~B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.~~ ~~AND~~ ~~B.2 Perform SR 3.9.1.1.~~	~~Immediately~~ ~~Once per 12 hours~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.9.3.1~~	~~Perform CHANNEL CHECK.~~	~~In accordance with the Surveillance Frequency Control Program.~~
~~SR 3.9.3.2~~	~~NOTE~~ ~~Neutron detectors are excluded from CHANNEL CALIBRATION.~~ ~~Perform CHANNEL CALIBRATION.~~	~~In accordance with the Surveillance Frequency Control Program.~~

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Attachment 2

REVISED TECHNICAL SPECIFICATION CHANGES (CLEAN)

3.9 REFUELING OPERATIONS

3.9.3 DELETED

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Attachment 3

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (MARKUP - FOR

INFORMATION ONLY)

B 3.9 REFUELING OPERATIONS

B 3.9.3 ~~Nuclear Instrumentation~~ DELETED

~~BASES~~

~~BACKGROUND~~

~~The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. Any two of four source range neutron flux monitors may be used.~~

~~The installed Westinghouse BF_3 source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). The installed source range neutron flux monitors are BF_3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps). The detectors also provide continuous visual indication in the control room. The NIS is designed in accordance with the criteria presented in Reference 1. Each portion of the Westinghouse source range neutron flux monitors has two trains and each is assigned to an independent Class 1E electrical train. These trains are physically and electrically separated in accordance with applicable IEEE Standards.~~

~~A separate Gamma-Metrics Neutron Flux Monitoring System (NFMS) is installed to satisfy the requirements of Regulatory Guide 1.97, "Instrumentation For Light-Watered-Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident." The Gamma-Metrics NFMS monitors neutron flux from the source range through 200% Rated Thermal Power (RTP) during all Modes of plant operation. This system utilizes two separate Safety Category I (Class 1E) fission chamber neutron detectors for all ranges of neutron flux indication. Each portion of the Gamma-Metrics instrumentation has two trains and each is assigned to a separate Class 1E electrical train. These trains are physically and electrically separated in accordance with applicable IEEE Standards.~~

~~The source range neutron flux monitors do not provide a Reactor Protection System function in Mode 6.~~

~~APPLICABLE SAFETY ANALYSES~~

~~Two OPERABLE source range neutron flux monitors are required to provide a visual signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident. (Ref. 2)~~

~~The source range neutron flux monitors satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).~~

~~(continued)~~

~~BASES (continued)~~

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~~LCO~~	~~This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication in the control room.~~
~~APPLICABILITY~~	~~In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In other MODES, the NIS source range monitors are governed by LCO 3.3.1.~~
~~ACTIONS~~	~~A.1 and A.2~~
	~~With only one required source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position. Addition to the RCS of borated water with a concentration greater than or equal to the minimum required RWST concentration shall not be considered to be a positive reactivity change (Ref 3).~~
	~~B.1~~
	~~With no required source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.~~

~~(continued)~~

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~~ACTIONS (continued)~~

~~B.2~~

~~With no required source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and boron concentration changes inconsistent with Required Action A.2 are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.~~

~~The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.9.3.1~~

~~SR 3.9.3.1 is the performance of a CHANNEL CHECK of required channels, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~SR 3.9.3.2~~

~~SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. For the source range neutron detectors, performance data is obtained and evaluated. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~(continued)~~

~~BASES (continued)~~

~~REFERENCES~~	~~1.~~	~~10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.~~
	~~2.~~	~~FSAR, Section [15.2.4].~~
	~~3.~~	~~NRC letter (W. Reckley to N. Carns) dated November 22, 1993 "Wolf Creek Generating Station - Positive Reactivity Addition; Technical Specification Bases Changes".~~

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