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

June 10, 2024

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

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 187 AND 187, RESPECTIVELY RE: ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2023-LLA-0057)

Dear Ken Peters:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 187 to Facility Operating License (FOL) No. NPF-87 and Amendment No.187 to FOL No. NPF-89 for Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2 (Comanche Peak), respectively. The amendments consist of changes to the FOLs in response to your application dated April 19, 2023, as supplemented by letters dated June 8, 2023, and December 13, 2023.

The amendments revise the Comanche Peak FOL Nos. NPF-87 and NPF-89 to add a new license condition to allow for the implementation of Title 10 of the *Code of Federal Regulations* Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

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

Sincerely,

/RA/

Samson S. Lee, 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. 187 to NPF-87
2. Amendment No. 187 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. 187
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 April 19, 2023, as supplemented by letters dated June 8, 2023, and December 13, 2023, 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 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 as indicated in the attachment to this license amendment, and Renewed Facility Operating License No. NPF-87 is hereby amended to add paragraph 2.J to read as follows:

   J. Vistra OpCo is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except wind-generated missiles and seismic; the high winds safe shutdown equipment list for wind-generated missiles; and the alternative seismic approach as described in Vistra OpCo's submittal letter April 19, 2023, and all its subsequent associated supplements, as specified in License Amendment No. 187 dated June 10, 2024.

   Vistra OpCo will complete the High Winds Safe Shutdown Equipment List (HWSSEL) prior to performing any system categorization per 10 CFR 50.69.

   Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennivine K. Rankin

Digitally signed by Jennivine K. Rankin
Date: 2024.06.10 12:13:18 -04'00'

Jennivine K. Rankin, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License

Date of Issuance: June 10, 2024

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. 187
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 April 19, 2023, as supplemented by letters dated June 8, 2023, and December 13, 2023, 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 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 as indicated in the attachment to this license amendment, and Renewed Facility Operating License No. NPF-89 is hereby amended to add paragraph 2.J to read as follows:

   J. Vistra OpCo is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except wind-generated missiles and seismic; the high winds safe shutdown equipment list for wind-generated missiles; and the alternative seismic approach as described in Vistra OpCo's submittal letter April 19, 2023, and all its subsequent associated supplements, as specified in License Amendment No. 187 dated June 10, 2024.

      Vistra OpCo will complete the High Winds Safe Shutdown Equipment List (HWSSEL) prior to performing any system categorization per 10 CFR 50.69.

      Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

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

FOR THE NUCLEAR REGULATORY COMMISSION

Jennivine K. Rankin

Digitally signed by Jennivine K. Rankin
Date: 2024.06.10 12:13:46 -04'00'

Jennivine K. Rankin, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License

Date of Issuance: June 10, 2024

ATTACHMENT TO LICENSE AMENDMENT NO. 187

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 187

TO FACILITY OPERATING LICENSE NO. NPF-89

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

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of Facility Operating License Nos. NPF-87 and NPF-89 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
-8-	-8-
-9-	-9-

Facility Operating License No. NPF-89

REMOVE	INSERT
-8-	-8-
---	-9-

(3) CP PowerCo shall promptly notify the NRC of any attempts by subsurface mineral rights owners to exercise mineral rights, including any legal proceeding initiated by mineral rights owners against CP PowerCo.

G. Vistra OpCo shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report through Amendment 78 and as approved in the SER (NUREG-0797) and its supplements through SSER 24, subject to the following provision:

Vistra OpCo may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

H. Vistra OpCo shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans, previously approved by the Commission, and all amendments made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled: "Comanche Peak Steam Electric Station Physical Security Plan" with revisions submitted through May 15, 2006, with limited approvals as provided for in the Safety Evaluation by the Office of Nuclear Reactor Regulation dated December 5, 2000; "Comanche Peak Steam Electric Station Security Training and Qualification Plan" with revisions submitted through May 15, 2006; and "Comanche Peak Steam Electric Station Safeguards Contingency Plan" with revisions submitted through May 15, 2006. Vistra OpCo shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). Vistra OpCo's CSP was approved by License Amendment No. 155, as supplemented by a change approved by License Amendment 163.

I. CP PowerCo shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. Vistra OpCo is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except wind-generated missiles and seismic; the high winds safe shutdown equipment list for wind-generated missiles; and the alternative seismic approach as described in Vistra OpCo's submittal letter April 19, 2023, and all its subsequent associated supplements, as specified in License Amendment No. 187 dated June 10, 2024.

Vistra OpCo will complete the High Winds Safe Shutdown Equipment List (HWSSEL) prior to performing any system categorization per 10 CFR 50.69.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

K. This license is effective as of the date of issuance and shall expire at Midnight on February 8, 2030.

FOR THE NUCLEAR REGULATORY COMMISSION

original signed by:

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Appendix A – Technical Specifications (NUREG-1399)
2. Appendix B – Environmental Protection Plan
3. Appendix C – Deleted
4. Appendix D – Additional Conditions

Date of Issuance: April 17, 1990

H. Vistra OpCo shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans, previously approved by the Commission, and all amendments made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled: "Comanche Peak Steam Electric Station Physical Security Plan" with revisions submitted through May 15, 2006, with limited approvals as provided for in the Safety Evaluation by the Office of Nuclear Reactor Regulation dated December 5, 2000; "Comanche Peak Steam Electric Station Security Training and Qualification Plan" with revisions submitted through May 15, 2006; and "Comanche Peak Steam Electric Station Safeguards Contingency Plan" with revisions submitted through May 15, 2006. Vistra OpCo shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). Vistra OpCo's CSP was approved by License Amendment No. 155, as supplemented by a change approved by License Amendment 163.

I. CP PowerCo shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. Vistra OpCo is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)–1, RISC–2, RISC–3, and RISC–4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except wind-generated missiles and seismic; the high winds safe shutdown equipment list for wind-generated missiles; and the alternative seismic approach as described in Vistra OpCo's submittal letter April 19, 2023, and all its subsequent associated supplements, as specified in License Amendment No. 187 dated June 10, 2024.

Vistra OpCo will complete the High Winds Safe Shutdown Equipment List (HWSSEL) prior to performing any system categorization per 10 CFR 50.69.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

K. This license is effective as of the date of issuance and shall expire at Midnight on February 2, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Appendix A - Technical Specifications (NUREG-1468)
2. Appendix B - Environmental Protection Plan
3. Appendix C – Deleted
4. Appendix D – Additional Conditions

Date of Issuance: April 6, 1993

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 187 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 187 TO

FACILITY OPERATING LICENSE NO. NPF-89

VISTRA OPERATIONS COMPANY LLC

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

DOCKET NOS. 50-445 AND 50-446

1.0 PROPOSED LICENSE CONDITION

By letter dated April 19, 2023 (Reference [1]), as supplemented by letters dated June 8, 2023 (Reference [2]), and December 13, 2023 (Reference [3]), Vistra Operations Company LLC, (Vistra OpCo, the licensee) submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC, the Commission) for the Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2 (Comanche Peak or CPNPP). The licensee proposed the following license condition to the Comanche Peak Facility Operating Licenses to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

> Vistra OpCo is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE [Individual Plant Examination of External Events] Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS [American Society of Mechanical Engineers/American Nuclear Society] PRA Standard RA-Sa-2009 for other external hazards except wind-generated missiles and seismic; the high winds safe shutdown equipment list for wind-generated missiles; and the alternative seismic approach as described in Vistra OpCo's

submittal letter April 19, 2023, and all its subsequent associated supplements, as specified in License Amendment No. 187 dated June 10, 2024.

Vistra OpCo will complete the High Winds Safe Shutdown Equipment List (HWSSEL) prior to performing any system categorization per 10 CFR 50.69.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed process that includes several approaches and methods for categorizing SSCs according to their safety significance[1].

The NRC staff participated in a regulatory audit during November 14, 2023. The NRC staff performed the audit to ascertain the information needed to support its review of the application and develop requests for additional information (RAIs), based on audit questions transmitted by email October 10, 2023 (Reference [4]). On January 29, 2024, the NRC staff issued an audit summary (Reference [5]). The licensee responded to the audit with a supplemental letter dated December 13, 2023, providing additional information associated with NRC questions discussed in the audit.

The proposed amendments would use the methodology from the NRC staff approved LARs related to 10 CFR 50.69 from Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs) seismic approach (Reference [22]).

The proposed amendments would also use some of the methodologies from the NRC staff approved Comanche Peak LAR related to the adoption of Technical Specifications Task Force Traveler (TSTF) TSTF-505, Revision 2 (Reference [20]).

The supplemental letter dated December 13, 2023, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on August 8, 2023 (88 FR 53542).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design basis functions.

---

1 Regulatory Guide (RG) 1.201, Revision 1, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” May 2006, describes the SSC categorization process in its entirety as an overarching approach that includes multiple approaches and methods identified for a PRA hazard and non-PRA methods.

Section 50.69 of 10 CFR contains requirements regarding how a licensee categorizes SSCs using a risk-informed process; adjusts treatment requirements consistent with the relative significance of the SSC; and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four RISC categories.

SSC categorization does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that perform safety significant functions, existing treatment requirements are maintained or potentially enhanced. Conversely, for SSCs that perform low safety significant functions, such that they do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that performs safety significant functions.

2.2 Regulatory Guide

The NRC staff considered the following regulatory guidance during its review of the proposed changes:

- Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" (Reference [6])

- RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference [7])

- RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference [8])

- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (Reference [9])

- NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference [10])

NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference [11]), which is endorsed by RG 1.201 for trial use with clarifications and describes a process that the NRC staff considers acceptable for complying with 10 CFR 50.69. This process determines the safety significance of SSCs and categorizes them into one of four RISC categories defined in 10 CFR 50.69.

Sections 2 through 10 of NEI 00-04 describe the following steps/elements of the SSC categorization process for meeting the requirements of 10 CFR 50.69:

- Sections 3.2, “Use of Risk Information”; and 5.1, “Internal Events Assessment,” provide specific guidance corresponding to 10 CFR 50.69(c)(1)(i).
- Sections 3, “Assembly of Plant-Specific Inputs”; 4, “System Engineering Assessment”; 5, ”Component Safety Significance Assessment”; and 7, “Preliminary Engineering Categorization of Functions,” provide specific guidance corresponding to 10 CFR 50.69(c)(1)(ii).
- Section 6, “Defense-in-Depth Assessment,” provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iii).
- Section 8, “Risk Sensitivity Study,” provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iv).
- Section 2, “Overview of Categorization Process,” provides specific guidance corresponding to 10 CFR 50.69(c)(1)(v).
- Sections 9, “IDP [Integrated Decisionmaking Panel] Review and Approval”; and 10, “SSC Categorization,” provide specific guidance corresponding to 10 CFR 50.69(c)(2).

Additionally, section 11, “Program Documentation and Change Control,” of NEI 00-04 provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f). Section 12, “Periodic Review,” of NEI 00-04 provides guidance on the periodic review related to the requirements in 10 CFR 50.69(e), “Feedback and process adjustment.” Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current as-built, as-operated plant configuration and applicable plant and industry operational experience as required by 10 CFR 50.69 (c)(1)(ii).

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

An acceptable approach for making risk-informed decisions about proposed technical specification changes, including both permanent and temporary changes, is to show that the proposed licensing basis changes meet the five key principles stated in section C of RG 1.174, Revision 3. These key principles are:

Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption….

Principle 2: The proposed licensing basis change is consistent with the defense-in-depth philosophy.

Principle 3: The proposed licensing basis change maintains sufficient safety margins.

Principle 4: When the proposed licensing basis changes result in an increase in risk, the increase should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

Principle 5: The impact of the proposed licensing basis change should be monitored by using performance measures strategies.

3.2 Traditional Engineering Evaluation

The traditional engineering evaluation below addresses the first three key principles of RG 1.174, Revision 3 and are pertinent to: (1) compliance with current regulations, (2) evaluation of defense-in-depth, and (3) evaluation of safety margins.

Key Principle 1: Licensing Bases Change Meets the Current Regulations

Section 50.69(c) of 10 CFR requires licensees to use an integrated decision-making process to categorize safety-related and non-safety-related SSCs according to the safety significance of the functions they perform into one of the following four RISC categories, which are defined in 10 CFR 50.69(a), as follows:

RISC–1: Safety-related SSCs that perform safety significant functions[2]
RISC–2: Non-safety-related SSCs that perform safety significant functions
RISC–3: Safety-related SSCs that perform low safety significant functions
RISC–4: Non-safety-related SSCs that perform low safety significant functions

The SSCs are classified as having either high-safety-significant (HSS) functions (i.e., RISC-1 and RISC-2 categories) or low safety-significant (LSS) functions (i.e., RISC-3 and RISC-4 categories). For HSS SSCs, 10 CFR 50.69 maintains current regulatory requirements for special treatment (i.e., it does not remove any requirements from these SSCs). In addition, 10 CFR 50.69(d)(1) requires that '[t]he licensee or applicant shall ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance." For LSS SSCs, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2). For RISC-3 SSCs, licensees may replace certain special treatment requirements with an alternative treatment approach that meets 10 CFR 50.69(d)(2). For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements.

Section 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of this section by issuance of a license amendment if the Commission determines that the categorization process satisfies the requirements of 10 CFR 50.69(c).

---

2 NEI 00-04, Revision 0, uses the term "high-safety-significant" to refer to SSCs that perform safety-significant functions. The NRC understands HSS to have the same meaning as "safety-significant" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

As stated in 10 CFR 50.69(b), after the NRC approves an application for a license amendment, a licensee may voluntarily comply with 10 CFR 50.69, as an alternative to compliance with the following requirements for LSS SSCs:

(i) 10 CFR Part 21
(ii) a certain portion of 10 CFR 50.46a(b)
(iii) 10 CFR 50.49
(iv) 10 CFR 50.55(e)
(v) certain requirements of 10 CFR 50.55a
(vi) 10 CFR 50.65, except for paragraph (a)(4)
(vii) 10 CFR 50.72
(viii) 10 CFR 50.73
(ix) Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50
(x) certain requirements for containment leakage testing
(xi) certain requirements of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100

The NRC staff reviewed the licensee's SSC categorization process against the categorization process described in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and the acceptability of the licensee's PRA for use in the application of the 10 CFR 50.69 categorization process. The NRC staff's review, as documented in this safety evaluation (SE), used the framework provided in RG 1.174, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, with specific clarifications.

Section 2 of NEI 00-04, Revision 0, in part, states that the categorization process includes eight primary steps:

1. Assembly of Plant-Specific Inputs (Section 3 of NEI 00-04, Revision 0)
2. System Engineering Assessment (Section 4 of NEI 00-04, Revision 0)
3. Component Safety Significance Assessment (Section 5 of NEI 00-04, Revision 0)
4. Defense-In-Depth Assessment (Section 6 of NEI 00-04, Revision 0)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04, Revision 0)
6. Risk Sensitivity Study (Section 8 of NEI 00-04, Revision 0)
7. Integrated Decisionmaking Panel Review and Approval (Section 9 of NEI 00-04, Revision 0)
8. SSC Categorization (Section 10 of NEI 00-04, Revision 0)

In section 3.1.1, "Overall Categorization Process," of the enclosure to its LAR, the licensee stated that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. In sections 3.1.2, "Passive Categorization Process," and 3.2.4, "Other External Hazards," of the LAR enclosure, the licensee has proposed the use of the ANO-2 Risk-Informed Repair / Replacement passive categorization method, the Electric Power Research Institute (EPRI) Tier 1 Alternative Seismic Method, and a High Winds Safe Shutdown Equipment List (HWSSEL) of SSCs to achieve and maintain safe shutdown of the reactor assuming unavailability of offsite power as alternative methods to assess the applicable hazard contribution(s). The NRC notes that use of these alternative methods is a deviation from the NEI 00-04 guidance as endorsed. A more detailed NRC staff review of the alternative methods is provided in section 3.3.2 of this SE.

The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process that are delineated in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

The regulatory requirements in 10 CFR 50.69 and 10 CFR Part 50, Appendix B; the monitoring outlined in NEI 00-04, Revision 0, and clarifications in RG 1.201, Revision 1, ensures that the SSC categorization process is sufficient to assure that the SSC functions continue to be met and that any performance deficiencies will be identified, and appropriate corrective actions taken. The licensee's SSC categorization program includes the appropriate steps/elements prescribed in NEI 00-04, Revision 0, to assure that SSCs specified are appropriately categorized consistent with 10 CFR 50.69. The NRC staff performed a more detailed review of specific steps/elements of the licensee's SSC categorization process where necessary to confirm consistency with the NEI 00-04 guidance, as endorsed. Considering the above, the staff concludes that the proposed 10 CFR 50.69 program meets the first key principle for risk-informed decision making prescribed in RG 1.174, Revision 3.

<u>Key Principle 2: Licensing Basis Change is Consistent with the Defense-In-Depth Philosophy</u>

In RG 1.174, Revision 3, the NRC identified the following considerations used for evaluating how the licensing basis change is maintained for the defense-in-depth philosophy:

- Preserve a reasonable balance among the layers of defense.
- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
- Preserve adequate defense against potential CCFs [common-cause failures].
- Maintain multiple fission product barriers.
- Preserve sufficient defense against human errors.
- Continue to meet the intent of the plant's design criteria.

RG 1.201, Revision 1, endorses the guidance in section 6 of NEI 00-04, but notes that the containment isolation criteria in this section of the guidance, are separate and distinct from those set forth in 10 CFR 50.69(b)(1)(x). The criteria in 10 CFR 50.69(b)(1)(x) are to be used in determining which containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Option A, "Prescriptive Requirements," and Option B, "Performance-Based Requirements," of Appendix J to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The criteria provided in 10 CFR 50.69(b)(1)(x) are not to determine the proper RISC category for containment isolation valves or penetrations.

In section 3.1.1 of the enclosure to its LAR, the licensee clarified that it would require an SSC to be categorized as HSS based on the defense-in-depth assessment performed in accordance with NEI 00-04, Revision 0. In light of the observation above, the NRC staff concludes that the proposed change is consistent with the defense-in-depth philosophy and therefore satisfies the second key principle for risk-informed decision-making prescribed in RG 1.174, Revision 3. The NRC staff finds that the licensee's process is consistent with the NRC-endorsed guidance in NEI 00-04; therefore key principle 2 of risk-informed decision-making is met and fulfills the 10 CFR 50.69(c)(1)(iii) criterion that requires defense-in-depth to be maintained.

Key Principle 3: LB Change Maintains Sufficient Safety Margins

The engineering evaluation that will be conducted by the licensee under 10 CFR 50.69 for SSC categorization will assess the design function(s) and risk significance of the SSC to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the categorization of the SSC does not adversely affect any assumptions or inputs to the safety analysis; or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

The SSCs design basis function as described in the plants' licensing basis, including the Updated Final Safety Analysis Report and Technical Specifications Bases do not change and should continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. On this basis, the staff concludes that safety margins are maintained by the proposed methodology, and the third key safety principle of RG 1.174, Revision 3, is satisfied.

System Engineering Assessment (NEI 00-004, Revision 0, Section 4)

In section 2.2, "Reason for the Proposed Change," of the enclosure to its LAR, the licensee stated, in part, that "[t]he safety functions [in the categorization process] include the design basis functions, as well as functions credited for severe accidents (including external events)." In section 3.1.1 of the enclosure to its LAR, the licensee summarizes the different hazards and plant states for which functional and risk significant information will be collected. In section 3.1.1 of the LAR, the licensee confirmed that the SSC categorization process documentation will include, among other items, system functions, identified and categorized with the associated bases, and mapping of components to support function(s).

The NRC staff finds that the process described in the LAR is consistent with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii) and 10 CFR 50.69(c)(1)(iv).

3.3 Risk-Informed Assessment

Key Principle 4: Change in Risk is Consistent with the Safety Goals

The risk-informed considerations prescribed in NEI 00-04, Revision 0, endorsed by RG 1.201, Revision 1, address the fourth and fifth key principles of the NRC staff's guidance for risk-informed decision-making, pertaining to the assessment for change in risk and monitoring the impact of the licensing basis change.

A summary of how the licensee's SSC categorization process is consistent with the guidance and methodology prescribed in NEI 00-04, Revision 0, and RG 1.201, Revision 1, is provided in the sections below:

Assembly of Plant-Specific Inputs (NEI 00-04, Revision 0, Section 3)

The NRC staff acknowledges that elements of the categorization process, as discussed in section 3.2 of this SE (e.g., system selection, system boundary definition, identification of system functions, and mapping of components to functions), are not always performed in chronological order and may be performed in parallel. The licensee's risk categorization process uses PRAs to assess risks from the internal events (IEPRA) (including internal floods), and fire PRA (FPRA). For non-PRA methods that depart from the methodology prescribed in NEI 00-04, additional staff review is discussed in section 3.3.2 of this SE.

Paragraph 50.69(c)(1)(v) of 10 CFR requires that SSC categorization be performed for entire systems and structures, not for selected components within a system or structure. The NRC staff finds the process described in the LAR for collecting and organizing information at the system level for defining boundaries, functions, and components is consistent with NEI 00-04 as endorsed by the staff in RG 1.201, and therefore meets the requirements set forth in 10 CFR 50.69(c)(1)(v).

Component Safety Significance Assessment (NEI 00-04, Section 5)

This step in the licensee's categorization process assesses the safety significance of components using quantitative or qualitative risk information from a modeled PRA hazard, other hazards that can be screened, and non-PRA method(s). In NEI 00-04 guidance, component risk significance is assessed separately for the following hazard groups:

- internal events (including internal floods)
- internal fire events
- seismic events
- external hazards (e.g., high winds, external floods)
- other hazards
- shutdown events
- passive categorization

In sections 3.1, "Categorization Process Description ([10 CFR 50.69(b)(2)(i)])," and 3.2, "Technical Adequacy Evaluation ([10 CFR 50.69(b)(2)(ii)])," of the enclosure to its LAR, as supplemented by letter dated December 13, 2023, the licensee described that the Comanche Peak categorization (including passive categorization) process uses PRA to assess risks for the internal events (including internal floods) and internal fires. For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

- Seismic Hazard: Tier 1 alternate method provided in EPRI Report 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk Informed Categorization" (Reference [12]).

- High Winds Hazard: Use of High Winds Safe Shutdown Equipment List.

- External and Other Hazards, except for seismic and high winds: Screening analysis performed for IPEEE (Reference [13]) updated using criteria from Part 6 of the PRA Standard ASME/ANS RA-Sa–2009, "Addendum A to RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference [14]), as endorsed by the NRC.

- Shutdown Events: Safe Shutdown Risk Management program consistent with Nuclear Management and Resources Council (NUMARC) 91-06, “Guidelines for Industry Actions to Assess Shutdown Management” (Reference [15]).

- Passive Components: ANO-2 passive categorization methodology (Reference [16]).

The approaches and methods proposed by the licensee to address internal events, non-seismic external events (except for high winds hazard), other hazards, defense-in-depth, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0. The non-PRA method for the categorization for passive components is consistent with the ANO-2 methodology for passive components approved for risk-informed safety classification and treatment for repair/replacement activities in Class 2 and 3 moderate- and high-energy systems. The use of the ANO-2 methodology in the SSC categorization process is provided in section 3.3.2 of this SE. To address the seismic and high winds risk, the licensee proposed to use the alternative Tier 1 seismic method and HWSSEL method, respectively, not specified in NEI 00-04 guidance as endorsed by the NRC. An NRC staff review of the licensee’s use of these proposed approaches is provided in section 3.3.2 of this SE.

3.3.1 Scope of the PRA

The Comanche Peak PRA is comprised of a full-power, Level 1, IEPRA and FPRA, which evaluate the core damage frequency (CDF) and large early release frequency (LERF) risk metrics. The licensee discussed in section 3.3, “PRA Review Process Results ([10 CFR 50.69(b)(2)(iii)])” of the enclosure to its LAR that the IEPRA (includes internal floods) model has been assessed against RG 1.200, Revision 2. Furthermore, the LAR references the Comanche Peak TSTF-505 LAR by Vistra OpCo (Reference [17]) where further details on PRA technical acceptability are provided. As stated in the Comanche Peak TSTF-505 LAR, a finding closure review was conducted in 2019. Open findings were reviewed and closed using the NRC-accepted process documented in the NEI letter to the NRC “Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-out of Facts and Observations,” dated February 21, 2017 (Reference [18]).

In the LAR supplement dated June 8, 2023, the licensee stated there have been no IEPRA (includes internal floods) model updates since the TSTF- 505 submittal.

The NRC staff finds that the information provided in the LAR, as supplemented by the Comanche Peak TSTF-505 LAR, support the staff review of the IEPRA (includes internal floods) and FPRA for technical acceptability and therefore the requirements set forth in 10 CFR 50.69(b)(2)(iii) are met.

Aspects considered by the NRC staff to evaluate the scope of the PRA include: (1) peer-review history, process, and results, (2) credit for diverse and flexible mitigation capability (FLEX) in the PRA, and (3) assessment of PRA model assumptions and approximations. The staff’s review of these aspects of the PRA to assess for consistency with the applicable processes as endorsed by the NRC, where necessary, are provided below.

Internal Events PRA (includes internal floods) Peer-Review History

In section 3.3 of the enclosure to its LAR, the licensee references the Comanche Peak TSTF-505 LAR, stating that the internal events PRA model was subjected to a full-scope peer review in 2011. Subsequently in 2019, the licensee conducted an independent assessment for closure of the finding-level facts and observations (F&Os) and concluded all the IEPRA (includes internal floods) F&Os have been closed. In its LAR supplement dated June 8, 2023, the licensee stated that there have been no IEPRA (includes internal floods) model updates since the TSTF-505 submittal.

In section 3.3 of the enclosure to its LAR, the licensee references the Comanche Peak TSTF-505 LAR which indicates all F&Os were appropriately assessed by the independent assessment team, which resulted in the closure of all finding-level F&Os. In its LAR supplement dated June 8, 2023, the licensee acknowledged two supporting requirements (SRs) (i.e., IFEV-A6 and LE-C11)) that are only met at Capability Category (CC) I and provided assessments of their impact on the 10 CFR 50.69 application. For SR IFEV-A6, the licensee then stated that no internal flooding events have occurred at Comanche Peak based on its search, and therefore, not using event frequencies based on a Bayesian update with plant-specific data yields conservative internal flooding PRA risk results. The NRC staff also notes that the lack of internal flooding events provides no new information about plant-specific internal flooding vulnerabilities, and therefore the impact this F&O being met at only CC-I has minimal impact on 10 CFR 50.69 categorization.

Concerning SR LE-C11, the licensee stated in its LAR supplement dated June 8, 2023, that "[n]o credit was taken for continued operation of equipment after containment failure" and that not crediting such actions has minimal impact on 10 CFR 50.69 categorization because in these cases the core is damaged, and containment has failed. The NRC staff observed that crediting operator actions or/and other equipment after containment failure could potentially limit the release to the environment from the core to less than a large early release for certain scenarios and meaningfully skew LERF importance factors.

In its LAR supplement dated December 13, 2023, the licensee stated that Comanche Peak Level 2 PRA results are dominated by "by-pass" and steam generator tube ruptures. Containment over-pressure scenarios, to which the cited credit for continued operation after containment failure would apply, are minor contributors as evidenced by the insignificance of the containment spray system. The licensee stated that the concerns associated with "by-pass" and steam generator tube ruptures are governed by the availability of the auxiliary feedwater system, which is a safety significant system, and the size of the penetration opposed to the "limited" uncredited equipment in determining the Level 2 bin. Therefore, the licensee concluded that the CC-I modeling associated with SR LE-C11 will not play a major role in identifying significant components. None-the-less, the licensee stated that the "50.69 Program implementing procedures will include instruction to the categorization team to include an assessment of these effects in the information package for consideration by the Integrated Decision-making Panel." Therefore, the NRC staff finds that the resolution of SR LE-C11 at CC-I is sufficient for this application because (1) the uncredited equipment will not play an important role in identifying significant components and (2) the Comanche Peak 10 CFR 50.69 Program implementing procedures will include consideration of this modeling information using instruction to the categorization team in the information package for the Integrated Decision-making Panel (IDP).

The licensee stated, in its LAR supplement dated June 8, 2023, that there have been no IEPRA model updates since the TSTF-505 submittal.

Therefore, the NRC staff concludes that the Comanche Peak IEPRA (including internal floods) was appropriately peer reviewed, consistent with RG 1.200, Revision 2, and the F&O's have been adequately closed or dispositioned to assess the impact on the 10 CFR 50.69 risk-informed application.

Internal Fire PRA Peer-Review History

In section 3.3 of the enclosure to its LAR, the licensee references the Comanche Peak TSTF-505 LAR, which indicated that the licensee's fire PRA was subject to a full-scope industry peer review in 2016, consistent with RG 1.200, Revision 2. The finding-level F&Os from the 2016 full-scope peer review were considered fully resolved by the independent assessment review team in 2019. Therefore, in accordance with RG 1.200, Revision 2, no F&Os associated with the FPRA were provided in the LAR.

In its LAR supplement dated June 8, 2023, the licensee stated that there have been no FPRA model updates since the TSTF-505 submittal.

In section 3.2.2, "Fire Hazards," of the enclosure to its LAR, the licensee stated, in part, that "[t]he Internal Fire PRA model has been developed consistent with NUREG/CR-6850 [Reference [19]] and only utilizes methods previously accepted by the NRC." Since the last full-scope peer review of the licensee's FPRA, there have been several changes to NRC-accepted fire methods and studies, whose integration into the licensee's FPRA could potentially impact the 10 CFR 50.69 risk categorization results or risk acceptance guidelines for total CDF and total LERF.

In its LAR supplement dated June 8, 2023, the licensee stated, concerning impacts on FPRA associated with updated guidance on modeling methods, that the responses provided during the Comanche Peak TSTF-505 LAR review "can be used to support the 10 CFR 50.69 application." The NRC staff re-reviewed the dispositions provided in the Comanche Peak TSTF-505 SE (Reference [20]) associated with the new post-NUREG/CR-6850 guidance on FPRA method concerns. The NRC staff found the associated concerns to be resolved for the 10 CFR 50.69 application for the following reasons:

- The new guidance was explicitly considered.
- The licensee's approach was consistent with the new guidance.
- The licensee's approach was conservative compared to the new guidance.
- The resolution of the concern is the same for the 10 CFR 50.69 application even though the risk metrics used in the 10 CFR 50.69 application is different from TSTF-505 application, which involves change-in-risk.

The NRC staff has reviewed the FPRA peer review results, and the licensee's resolution of the results concludes that the Comanche Peak FPRA was appropriately peer-reviewed, consistent with RG 1.200, Revision 2, and the F&O's have been adequately dispositions to assess the impact on the risk-informed application.

<u>Credit for FLEX Equipment</u>

The NRC memorandum dated May 6, 2022, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Risk Assessments" (Reference [21]), provides the NRC staff's assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision making in accordance with the guidance of RG 1.200, Revision 2.

In its LAR supplement dated December 13, 2023, the licensee stated "[t]he CPNPP PRA models do not model any FLEX (portable) equipment or mitigating actions." Therefore, the NRC staff finds that no further review is necessary for crediting FLEX equipment.

Assessment of PRA model Assumptions and Approximations

Identification of Key Assumptions and Sources of Uncertainty

In section 3.2.7, "PRA Uncertainty Evaluations," of the enclosure to its LAR, the licensee indicated that the guidance in NUREG-1855, Revision 1 was used to identify, screen, and characterize those sources of model uncertainty and related assumptions in the base PRA that are relevant to this application. In its supplement dated June 8, 2023, the licensee further described its process for reviewing PRA uncertainties. The licensee indicated that all PRA models were reviewed to identify those uncertainties that would be significant for the 10 CFR 50.69 application. The NRC staff finds that the assessment performed to identify the key assumptions/sources of uncertainty is consistent with the guidance provided in NUREG-1855, Revision 1.

Treatment of the Key Assumptions and Sources of Uncertainty

NUREG-1855, Revision 1, provides guidance regarding how to address PRA uncertainties to assure the risk-informed decision is in the context of the application for the decision under consideration.

The licensee described how its assessment of PRA modeling uncertainties in its LAR, as supplemented, is consistent with the guidance in NUREG-1855, Revision 1. In its LAR supplement dated June 8, 2023, the licensee presented a list of candidate sources of uncertainty and their associated dispositions for the 10 CFR 50.69 application. The licensee concluded that there are no key sources of uncertainty for this application, and therefore no additional sensitivity studies are required to address Comanche Peak PRA model specific assumptions or sources of uncertainty for the 10 CFR 50.69 program.

In addition, the NRC staff recognizes that the licensee will perform routine PRA changes and updates to assure the PRA continually reflects the as-built, as-operated plant, in addition to changes made to the PRA to support the context of the analysis being performed (i.e., sensitivities). Sections 50.69(e) and (f) of 10 CFR stipulates the process for feedback and adjustment to assure configuration control is maintained for these routine changes and updates to the PRA(s).

PRA Acceptability Conclusions

Pursuant to 10 CFR 50.69(c)(1)(i), the categorization process must consider results and insights from a plant-specific PRA. The use of the IEPRA and FPRA to support SSC categorization is endorsed by RG 1.201, Revision 1. The PRAs must be acceptable to support the categorization process and must be subjected to a peer-review process assessed against a standard that is endorsed by the NRC. Revision 2 of RG 1.200 provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa-2009 PRA Standard using a peer-review process.

The licensee has subjected the IEPRA and FPRA to the peer-review processes and submitted the results of the peer review. The NRC staff reviewed the peer-review history (which included the results and findings), the licensee's resolution of peer-review findings, and the identification and disposition of key assumptions and sources of uncertainty. The staff concludes that (1) the licensee's IEPRA and FPRA are acceptable to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201, Revision 1, and (2) the key assumptions for the PRAs have been identified consistent with the guidance in RG 1.200, Revision 2, and NUREG-1855, as applicable, and addressed appropriately for this application.

The NRC staff finds the licensee provided the required information, and the IEPRA (including internal floods) and FPRA are acceptable and therefore meets the requirements set forth in 10 CFR 50.69(c)(1)(i) and 10 CFR 50.69(c)(1)(ii).

3.3.2 Evaluation of the Use of Non-PRA Methods in SSC Categorization

The licensee's categorization process uses the following non-PRA method(s), respectively:

- Seismic Hazard: Tier 1 alternate method provided in EPRI Report 3002017583.
- High Winds Hazard: Use of a HWSSEL.
- External and Other Hazards, except for seismic and high winds: Screening analysis performed for IPEEE updated using criteria from Part 6 of the ASME/ANS RA-Sa–2009 PRA Standard, as endorsed by the NRC.
- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06.
- Passive Components: ANO-2 passive categorization methodology.

The NRC staff's review of these methods is discussed below.

<u>Seismic Risk</u>

As part of its proposed process to categorize SSCs according to safety significance, the licensee proposed to use a non-PRA method to consider seismic hazards. The regulations in 10 CFR 50.69(c)(1)(ii) and 10 CFR 50.69(b)(2)(ii) permit the use of systematic evaluation techniques in the risk-informed categorization process.

The licensee provided a description of its proposed alternative seismic approach for considering seismic risk in the categorization process and described how the proposed alternative seismic approach would be used in the categorization process described in section 3.2.3, "Seismic Hazards," of the enclosure to its LAR, as supplemented.

Specifically, in its LAR supplement dated December 13, 2023, the licensee clarifies that its alternative seismic approach uses EPRI Report 3002017583 with some clarifications, which are discussed below in this SE in the subsection titled "Evaluation of the Implementation of Conclusions from the Case Studies." EPRI Report 3002017583 is a technical update that incorporated updates submitted to the NRC staff in a Calvert Cliffs 10 CFR 50.69 RAI response to EPRI Report 3002012988. Aside from those updates, the technical criteria in EPRI

Report 3002017583 are identical to EPRI Report 3002012988. The licensee clarified that, to support the Comanche Peak alternate seismic approach, it is incorporating by reference the supplements from NRC review and approval of the Calvert Cliff's use of this method:

- Calvert Cliffs LAR supplement dated May 10, 2019 (Reference [22]); contains additional information related to the alternate seismic approach including docketed information related to case study Plants A, C, and D.

- Calvert Cliffs RAI response dated July 1, 2019 (Reference [23]); further clarifies the information related to the alternate seismic approach (see Calvert Cliffs response to RAI 4).

- Calvert Cliffs RAI response dated July 19, 2019 (Reference [24]); provides responses to support the technical acceptability of the PRAs used for the Plant A, C, and D case studies, as well as the technical adequacy of certain details of the conduct of the case studies (see Calvert Cliffs responses to RAI questions 1, 2, and 3).

- Calvert Cliffs RAI response dated August 5, 2019 (Reference [25]); clarifies the response to RAI 3.

In part, the licensee based its plant-specific evaluation on the case studies performed in EPRI Report 3002017583 and indicated that the case studies are applicable to Comanche Peak and stated that the licensee's proposed alternative seismic approach would be used in the categorization process; and the measures for assuring the quality and level of detail for the licensee's proposed alternative seismic approach are adequate for the categorization of SSCs. Therefore, based on the above, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(ii) for the proposed alternative seismic approach are met because the licensee provided a description of the measures taken to assure the quality and level of detail of their proposed alternative seismic approach.

EPRI Report 3002017583 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs (SPRAs). The NRC staff's review confirmed that the case studies in EPRI Report 3002017583 used by the licensee to support its proposed alternative seismic approach, as well as the information in its supplement, provided sufficient plant-specific evaluation of the applicability and differences for Comanche Peak as compared to the amendment approved by the NRC for Calvert Cliffs on February 28, 2020 (Reference [26]). Accordingly, the use of EPRI Report 3002017583 and its cited case studies have been previously approved by the NRC (Reference [26]). The information presented in the LAR and its supplement provided a sufficient description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv) for the alternative seismic approach. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the proposed alternative seismic approach.

<u>Evaluation of the EPRI Report 3002017583 Case Studies</u>

In its supplement dated December 13, 2023 regarding section 3.2.3 of the LAR enclosure, the licensee stated that the plant specific case studies from other licensees in EPRI Report 3002017583 are incorporated by reference to support its proposed alternative seismic approach. The NRC staff reviewed and evaluated the technical acceptability of the PRAs used in case studies for Plants A, C, and D, in EPRI Report 3002017583, and the licensee's assertion of plant-specific applicability to the approach used in the Calvert Cliffs' amendments.

The NRC staff finds that the acceptability of PRAs used in the Plants A, C, and D, case studies in EPRI Report 3002017583, the mapping approach used in those case studies, and the conclusions on the determination of unique HSS SSCs from the case studies in the Calvert Cliffs' amendments are applicable to this licensee's proposed plant-specific alternative seismic approach.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In the LAR, the licensee stated, "[t]he Maximum GMRS [Ground Motion Response Spectrum] value for CPNPP in the 1-10 Hz [hertz] range meets the Tier 1 criterion of approximately 0.2g in Reference 5." Additionally, in its supplement dated December 13, 2023, the licensee provided a Comanche Peak safe shutdown earthquake (SSE)/GMRS comparison plot demonstrating this fact and showed that the GMRS is below the SSE between 1.0 and 10.0 Hz. This demonstrates that Comanche Peak qualifies as a Tier 1 plant under the criteria provided in EPRI Report 3002017583.

The NRC staff observes that the licensee's plant-specific evaluation is supported by its NRC 10 CFR 50.54(f) response dated March 27, 2014 (Reference [27]). The NRC staff reviewed the licensee's submittal and supplements and plant-specific evaluation and concludes that the proposed criteria in EPRI Report 3002017583 to determine the applicability and use of the proposed seismic Tier 1 approach is acceptable.

Evaluation of Applicability of Criteria for 10 CFR 50.69

In its LAR supplement dated December 13, 2023, regarding section 3.2.3 of the LAR enclosure, the licensee compared the Comanche Peak GMRS from the reevaluated seismic hazard developed and submitted by the licensee in response to Near-Term Task Force (NTTF) Recommendation 2.1 against the site's design basis SSE, as shown in figure 3-2 of the LAR supplement dated December 13, 2023, to demonstrate that the site meets the criteria for application of the proposed alternative seismic approach as a Tier 1 plant. The NRC staff's review confirmed the licensee's statements and the comparison of the GMRS from the reevaluated seismic hazard against the SSE. Based on its review, the NRC staff finds that the licensee's seismic hazard meets the criteria for the proposed alternative seismic approach.

In its LAR supplement dated December 13, 2023, the licensee stated that "[t]he small percentage contribution of seismic to total plant risk makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination." Section 2.2.2, "Technical Basis for Approach," of EPRI Report 3002017583, which identifies the expectation that low contribution of seismic risk to the total plant risk reduces the likelihood of a unique seismic condition that would cause an SSC to be designated HSS so the seismic CDF would also be expected to be low.

The NRC staff compared the seismic risk to total plant risk based on information in the Comanche Peak TSTF-505 LAR. The NRC staff noted that seismic CDF (SCDF) contribution to total plant CDF is low (i.e., about 3 percent for Unit 1 and about 4 percent for Unit 2). The NRC staff reviewed the seismic LERF (SLERF) estimate in the Comanche Peak TSTF-505 LAR and noted that SLERF is about 12 percent of the total LERF for Unit 1 and 16 percent for Unit 2 using the SLERF penalty as a conservative estimate of SLERF. Therefore, given the above, and considering the conservative SLERF estimate presented, the staff finds that the overall seismic risk is relatively low compared to total plant risk due to its low SCDF and SLERF.

Further, as noted in section 3.6.5, "Defense-in-Depth Assessment," of EPRI Report 3002017583, containment defense-in-depth assessment addresses containment failures and containment bypass situations. Section 3.6.6, "Civil Structures," of EPRI Report 3002017583, used for the licensee's proposed alternative seismic approach, recommends that if the licensee chooses to categorize civil structures housing HSS SSCs, the structures are considered as HSS. Therefore, based on its evaluation and review, the NRC staff concludes that the proposed alternative seismic approach, in conjunction with the other elements of the 10 CFR 50.69 categorization program, will appropriately determine the safety significance of any SSCs whose seismic-induced failures led directly to core damage and large early release and that the seismic risk contribution would not solely result in any additional SSC being categorized as HSS.

The NRC staff finds that the licensee's basis for applying the proposed alternative seismic approach to its site is acceptable because: (1) the reevaluated hazard meets the criteria for use of the proposed alternative seismic approach, (2) in conjunction with the other elements of the 10 CFR 50.69 categorization program, the approach will appropriately determine the safety significance of any SSCs whose seismic-induced failures would lead directly to core damage and large early release, and (3) the seismic risk contribution would not solely result in any additional SSC being categorized as HSS.

Evaluation of the Implementation of Conclusions from the Case Studies

The licensee stated in its LAR supplement dated December 13, 2023, that the proposed categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for Comanche Peak. The objective of the alternative seismic approach is to identify plant-specific seismic insights derived from the components in the system being categorized.

The NRC staff's review of the licensee's proposed alternative seismic approach determined that the approach used in the Calvert Cliff's amendments (Reference [26]) is applicable to this licensee's proposed alternative seismic approach and that the plant-specific evaluation on the implementation of the alternative seismic approach is acceptable. There are essentially no differences that exist between the Comanche Peak proposed alternative approach and the approach used in the NRC staff approved Calvert Cliffs 10 CFR 50.69 safety evaluation. In the supplement dated December 13, 2023, the licensee presented the differences between the supplements modifying the EPRI guidance used by Calvert Cliffs and the Comanche Peak approach. The NRC staff reviewed these differences and found them not to be technical deviations from the approach that NRC approved for Calvert Cliffs but rather to be (1) discussions to provide clarification and further information included in recent similar LAR submittals, (2) elimination of Calvert Cliffs specific discussion that do apply to Comanche Peak, and (3) elimination of discussion not corresponding to seismic evaluations. The NRC staff's review of the proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding statement of consideration, finds that the proposed alternative seismic approach includes the evaluations required by 10 CFR 50.69(c)(1)(ii), as well as 10 CFR 50.69(c)(1)(iv) because:

1. The proposed alternative seismic approach includes qualitative consideration of seismic events at several steps of the categorization process, including documentation of the information for presentation to the IDP as part of the integrated, systematic process for categorization.

2. The proposed alternative seismic approach presents system-specific seismic insights to the IDP for consideration as part of the IDP review process as each system is categorized, thereby providing the IDP a means to consider potential impacts of seismic events in the categorization process.

3. The insights presented to the IDP include potentially important seismically induced failure modes, as well as mitigation capabilities of SSCs during seismically induced design basis and severe accident events consistent with the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI Report 3002017583. The insights will use prior plant-specific seismic evaluations and, therefore, in conjunction with performance monitoring for the proposed alternative seismic approach, reasonably reflect the current plant configuration. Further, the recommendation for categorizing civil structures in the alternative seismic approach provides appropriate consideration of such failures from a seismic event.

4. The proposed alternative seismic approach presents the IDP with the basis for the proposed alternative seismic approach, including the low seismic hazard for the plant and the criteria for use of the proposed alternative seismic approach.

5. The proposed alternative seismic approach includes qualitative consideration and insights related to the impact of a seismic event on SSCs for each SSC that is categorized and does not limit the scope to SSCs from the case studies supporting this application.

<u>Consideration of Changes to Seismic Hazard</u>

The possibility exists for the seismic hazard at the site to increase such that the criteria for use of the proposed alternative seismic approach are challenged. The licensee stated in the supplement dated December 13, 2023, that the seismic hazard at the plant is subject to periodic reconsideration as new information becomes available through industry evaluations. The license also stated that in the unlikely event that the Comanche Peak seismic hazard changes to medium risk at some future time that the licensee will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization.

The NRC staff finds that the consideration of changes to the seismic hazard in the licensee's plant-specific proposed alternative seismic approach is the same as that approved in Amendment Nos. 332 and 310 for Calvert Cliff, Units 1 and 2, respectively. Consequently, the NRC staff finds that the consideration of changes to the seismic hazard at Comanche Peak that exceed the criteria for use of the proposed alternative seismic approach is acceptable for the proposed approach because: (1) the criteria for use of the proposed alternative seismic approach are clear and traceable, (2) the proposed alternative seismic approach includes periodic reconsideration of the seismic hazard as new information becomes available, (3) the proposed alternative seismic approach satisfies the requirements in 10 CFR 50.69 discussed above, and (4) the licensee has included a proposed license condition in its LAR to require NRC approval for a change to the specified seismic categorization approach.

Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach

In its LAR supplement dated December 13, 2023, regarding section 3.2.3 of the LAR enclosure, the licensee stated that its configuration control process ensures that changes to the plant, including a physical change and changes to documents, are evaluated to ensure that the qualitative determinations for the seismic hazard continue to remain in compliance with the requirements of 10 CFR 50.69.

Based on its review, the NRC staff found that consideration of the feedback and adjustment process in the licensee's proposed alternative seismic approach is acceptable. The NRC staff finds that:

1. The licensee's programs provide reasonable assurance that the existing seismic capacity of LSS components would not be significantly impacted, and

2. The monitoring and configuration control program ensures that potential degradation of the seismic capacity would be detected and addressed before significantly impacting the plant risk profile.

Therefore, the NRC staff finds that the potential impact of the seismic hazard on the categorization is maintained acceptably low, and the requirements in 10 CFR 50.69(e) are met for the proposed alternative seismic approach.

High Winds Hazard

In section 3.2.4 of the enclosure to its LAR, as supplemented in a letter dated December 13, 2023, the licensee refers to the extreme winds and tornado hazards evaluation that was performed in support of the Comanche Peak TSTF-505 LAR. The licensee stated that because the tornado hazard is not screened, an alternate process for 10 CFR 50.69 categorization is proposed. The proposed alternate process, as updated in the LAR supplement dated December 13, 2023, will use a HWSSEL of SSCs required for safe shutdown. The licensee stated that the HWSSEL method is conservative since it identifies SSCs that support the safe shutdown pathway to be required to be HSS and includes SSCs relied upon to maintain the tornado design criteria, such as doors, vents, dampers, and louvers if those types of SSCs are being categorized in a system categorization. In section 3.2.4 of its LAR, as supplemented, the licensee indicates when an SSC is categorized as HSS by this non-PRA method, it "may not be re-categorized by the IDP."

In its LAR supplement dated December 13, 2023, the licensee updated its initial LAR statement that a Tornado Safe Shutdown Equipment List (TSSEL) will be used in the categorization. The licensee explained that Comanche Peak does not yet have a TSSEL and therefore proposed to develop an explicit HWSSEL for use in the 10 CFR 50.69 categorization process. The licensee stated that the HWSSEL will encompass hazards associated with tornadoes and straight winds, which are referred together as high winds. The licensee provided the steps that will be undertaken to develop the HWSSEL starting with using the IEPRA to identify safe shutdown equipment resulting from initiating events that can be caused by tornadoes or straight winds including missiles. Then SSCs whose functionality would prevent core damage and large early release would be identified. The licensee also explained that this will be performed prior to performing any system categorization. To ensure this commitment is met, the licensee provided an amended proposed license condition in enclosure 2 of the LAR supplement dated December 13, 2023. This addition to the proposed license condition states: “Vistra OpCo will complete the High Winds Safe Shutdown Equipment List (HWSSEL) prior to performing any system categorization per 10 CFR 50.69.”

In its LAR supplement dated December 13, 2023, the licensee justified that the use of HWSSEL is a bounding approach, because it automatically classifies SSCs on the HWSSEL as HSS and it does not consider the probability that the SSC successfully provides a safe shutdown pathway.

The licensee further stated that the HWSSEL will be a document that is created for the 10 CFR 50.69 categorization. As part of periodic assessment of systems categorized, it will be confirmed that the equipment list has not been impacted by plant changes and still reflects the current as-built, as-operated plant. The licensee explained that changes to the plant design and operations are periodically reviewed for potential impact to the PRA model and the approved risk-informed applications as part of the Comanche Peak PRA configuration control process. The licensee stated that upon approval of the 10 CFR 50.69 application, this periodic review will include all aspects of the 10 CFR 50.69 program, including the HWSSEL input to the categorization. The NRC staff finds that the HWSSEL is an active document that will reflect the current as-built, as-operated plant and that changes to the plant will be evaluated to determine their impact to the equipment list and the categorization process.

The NRC staff agrees that the Comanche Peak HWSSEL will identify a list of SSCs that when assigned to be HSS will provide a conservative approach to meeting the guidance in NEI 00-04 because the SSCs are assumed to be HSS for high winds (which includes straight wind, tornadoes, and missiles) though they may not be risk significant. Updated LAR table 3-1, as presented in the supplement dated December 13, 2023, shows how the high winds will be treated in the 10 CFR 50.69 categorization process including the fact that when an SSC is categorized as HSS by this non-PRA method, it may not be recategorized by the IDP. The NRC staff finds that the identified SSCs included on the HWSSEL are acceptable for use in the SSC categorization process because HSS is assigned to all those SSCs that support the HWSSEL function.

The HWSSEL is a screening approach; therefore there are no importance measures used in determining safety significance related to the fire hazard and assessment for LSS SSCs can be limited. The NRC staff finds that, in the absence of importance measures to assess the safety significance (e.g., defense-in-depth, safety margin, etc.), risk categorization of SSCs not included on the HWSSEL will be performed consistent with the NEI 00-04 guidance, therefore assuring the SSC has been appropriately assigned candidate LSS.

Based upon the NRC staff’s review of the licensee’s approach to use the HWSSEL provided in section 3.2.4 of the LAR enclosure, as supplemented, the NRC staff finds the licensee’s approach to use the Comanche Peak HWSSEL to assess the risk for high winds, when integrated with the other steps/elements of the categorization process provided in the NEI 00-04 guidance as endorsed by the NRC, is acceptable for use in the 10 CFR 50.69 SSC categorization program.

External Hazards and Other Hazards (Non-Seismic)

This hazard category includes all non-seismic and high winds external hazards such as external floods, transportation, and nearby facility accidents, and other hazards.

In the safety evaluation report for the Comanche Peak IPEEE for Units 1 and 2, the staff states, in part, “other external events were assessed using a screening type approach,” as described in NUREG-1407 (Reference [28]). In its LAR supplement dated June 8, 2023, the licensee stated that all other external hazards (i.e., not seismic, fire hazards, or high winds) were screened from applicability to Comanche Peak per a plant-specific evaluation in accordance with Generic Letter 88-20 and updated using the criteria in ASME/ANS RA-Sa-2009 PRA Standard.

In its LAR supplement dated June 8, 2023, the license confirmed that Comanche Peak will subject the external hazards (excluding internal fires, seismic hazards, and high winds) to the process described by the flow chart in NEI 00-04, figure 5-6. NEI 00-04, figure 5-6 provides guidance to be used to determine SSC safety significance for other external hazards. The NRC staff finds that the licensee will assess the risk from all other external hazards consistent with figure 5-6 of NEI 00-04 as endorsed in RG 1.201, Revision 1.

In its LAR supplement dated June 8, 2023, the licensee also provided the results of the plant-specific evaluation that assessed updated IPEEE results using endorsed criteria in the ASME/ANS RA-Sa-2009 PRA Standard and current plant hazard information. The NRC staff notes that this plant-specific evaluation or its results were not peer reviewed against Part 6 of the ASME/ANS Ra-SA-2009 PRA Standard as endorsed in RG 1.200, Revision 2. However, it is consistent with section 5 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and is, therefore, acceptable for this application and meets the requirements of 10 CFR 50.69(c)(1)(ii).

In the Comanche Peak TSTF-505 SE, the NRC staff found that new information in the Comanche Peak Flood Hazard Reevaluation and Comanche Peak Flooding Focused Evaluation, which included reevaluation of local intense precipitation and probable maximum flood was appropriately considered by the licensee. In its LAR supplement dated December 13, 2023, the licensee stated that the flood reevaluation (Reference [29]), performed in response to the NRC 10 CFR 50.54(f) information request regarding NTTF Recommendation 2.1 on flooding, confirmed the plant’s original design and concluded that there are no doors or other SSCs credited for protection of safety-related components due to external flooding concerns.

In summary, the use of updated Comanche Peak IPEEE results and the external hazard screening criteria in the ASME/ANS RA-Sa-2009 PRA Standard described by the licensee in the supplement dated June 8, 2023, and the licensee’s assessment of the other external hazards (i.e., seismic hazards and high winds) such as external flood is consistent with section 5 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, providing reasonable assurance that the plant has no significant external hazards. The NRC staff concludes that the licensee’s treatment of other external hazards is acceptable and meets 10 CFR 50.69(c)(1)(ii).

Component Safety Significance Assessment for Passive Components

Passive components are not modeled in the PRA; therefore, a different assessment method is necessary to assess the safety significance of these components. Passive components are those components having only a pressure retaining function. This process also addresses the passive function of active components such as the pressure/liquid retention of the body of a motor-operated valve.

In section 3.1.2, “Passive Categorization Process,” of the enclosure to its LAR, the licensee proposed using a categorization method, for passive components not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1, for passive component categorization, but was approved by the NRC for ANO-2. The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for Class 2 and 3 pressure retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, “Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1” (Reference [30]). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

In section 3.1.2 of the enclosure to its LAR, the licensee stated, “[t]he passive categorization process is intended to apply the same risk-informed process accepted in the ANO-2-R&R-004 for the passive categorization of Class 2, 3, and non-class components.” Furthermore, the licensee stated in part, that “[a]ll ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety significant, HSS … and cannot be changed by the IDP.” The NRC staff finds the licensee’s proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process.

3.3.3 Risk Sensitivity Study (NEI 00-04, Section 8)

In section 3.1.1 of the enclosure to its LAR, the licensee stated that “[a]n unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04.” Section 3.2.7 of the LAR enclosure further confirms that a cumulative sensitivity study will be performed where the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of 3. The NRC staff finds the application of a factor of 3 for the sensitivities is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

In section 3.1.1 of the enclosure to its LAR, the licensee specifically noted that “the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence” and that “all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv).” This sensitivity study together with the periodic review process discussed in section 3.4 of this SE, assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee will perform the risk sensitivity study consistent with the guidance in section 8 of NEI 00-04, Revision 0, and, therefore, will assure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.3.4 Integrated Decision-Making

Appendix B of SRP Section 19.2 provides guidance and the NRC staff expectations for the licensee's integrated decision-making process. The appendix guidance states in part, that "[r]isk-informed applications are expected to require a process to integrate traditional engineering and probabilistic considerations to form the basis for acceptance." NEI 00-04 guidance identifies two steps in the categorization process: (1) Preliminary Engineering Categorization of Function and (2) IDP Review and Approval that are responsible for the integrated assessment of the traditional engineering analyses and the risk results from the PRA and non-PRA assessments that are performed to decide and approve the safety significance of the SSC for categorization. The staff review of the two steps to ensure the processes is well-defined, systematic, repeatable, and scrutable are provided as follows:

<u>Preliminary Engineering Categorization of Function (NEI 00-04, Section 7)</u>

All the information collected and evaluated in the licensee's engineering evaluations is provided to the IDP as described in section 7 of NEI 00-04, Revision 0. The IDP will make the final decision about the safety significance of SSCs based on guidelines in NEI 00-04, Revision 0, the information they receive, and their expertise.

In section 3.1.1 of the enclosure to its LAR, the licensee stated, in part, "... if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS." The licensee also stated that, "[o]nce a system function is identified as HSS, then all the components that support that function are preliminary HSS."

The NRC staff finds that the above description provided by the licensee for the preliminary categorization of functions is consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and is therefore acceptable.

<u>IDP Review and Approval (NEI 00-04, Sections 9 and 10)</u>

In section 3.1.1 of the enclosure to its LAR, the licensee stated that the IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA. Therefore, the IDP will comprise the required expertise.

The guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, provides confidence that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process as required by 10 CFR 50.69(c)(1)(ii). In section 3.1.1 of its LAR enclosure, the licensee discusses that "[a]t least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in modeling and updating of the plant-specific PRA." The licensee further stated that "[t]he IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy." The NRC staff finds that the licensee's IDP areas of expertise meet the requirements in 10 CFR 50.69(c)(2) and the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

As discussed in NEI 00-04, Revision 0, the only LSS SSC requirements that are relaxed for RISC-3 (LSS) SSCs are those related to treatment, not design or capability, and 10 CFR 50.69(d)(2)(i) requires the licensee to ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. Therefore, the NRC staff finds that the IDP for the Comanche Peak categorization process, is consistent with the endorsed guidance in NEI 00-04, Revision 0, and, therefore, fulfills 10 CFR 50.69(c)(1)(iv).

Considering the above NRC staff review for: (1) IEPRA, and FPRA acceptability, (2) PRA importance measures and integrated importance measure, (3) evaluation of the use of non-PRA methods, (4) risk sensitivity study, and (5) integrated decision making, the NRC staff has determined that the proposed change satisfies the fourth key principle for risk-informed decision making prescribed in RG 1.174, Revision 3.

Key Principle 5: Monitor the Impact of the Proposed Change

NEI 00-04, Revision 0, provides guidance that includes programmatic configuration control and a periodic review to ensure that the all aspects of the 10 CFR 50.69 program (i.e., includes traditional engineering analyses) and PRA models used to perform the risk assessment continue to reflect the as-built-as-operated plant and that plant modifications and updates to the PRA overtime are continually incorporated.

In its LAR supplement dated December 13, 2023, the licensee clarified that its performance monitoring process includes review of new information about seismic issues that could impact the application such as a change in seismic qualification, the potential for seismic interaction, the potential for failure of passive SSCs, the potential for structural or anchorage degradation, and relay vulnerabilities. Furthermore, the licensee also clarified that use of the HWSSEL will be subject to the PRA configuration control process and reflect the as-built, as-operated plant.

Programmatic Configuration Control (NEI 00-04, Sections 11 and 12)

Sections 11 and 12 of NEI 00-04, Revision 0, includes a discussion on periodic review and program documentation and change control. Maintaining change control and periodic review will also maintain confidence that all aspects of the 10 CFR 50.69 program and risk categorization for SSCs, continually reflect the Comanche Peak plant as-built, as-operated plant. A more detailed NRC staff review is provided as follows:

Program Documentation and Change Control (NEI 00-04, Section 11)

Section 50.69(f) of 10 CFR requires, in part, program documentation, change control, and records. In section 3.2.6 of the enclosure to its LAR, the licensee stated that it will implement a process that addresses the requirements in section 11 of NEI 00-04, Revision 0, pertaining to program documentation and change control records. In section 3.1.1 of its LAR enclosure, the licensee stated that the RISC categorization process documentation will include the following 10 elements:

- Program procedures used in the categorization.
- System functions, identified and categorized with the associated bases.
- Mapping of components to support function(s).
- PRA model results, including sensitivity studies.
- Hazards analyses, as applicable.
- Passive categorization results and bases.
- Categorization results including all associated bases and RISC classifications.
- Component critical attributes for HSS SSCs.
- Results of periodic reviews and SSC performance evaluations.
- IDP meeting minutes and qualification/training records for the IDP members.

The NRC staff also recognizes that for facilities licensed under 10 CFR Part 50, Appendix B Criterion VI, "Document Control," procedures are considered formal plant documents, which require that "[m]easures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality."

The NRC staff finds that the elements provided in section 3.1.1 of the LAR for the Comanche Peak 10 CFR 50.69 categorization process will be documented in formal licensee procedures consistent with section 11 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and therefore sufficient for meeting the 10 CFR 50.69(f) requirement for program documentation, change control and records.

<u>Periodic Review (NEI 00-04, Section 12)</u>

Section 50.69(e) of 10 CFR requires periodic updates to the licensee's PRA and SSC categorization must be performed. Changes over time to the PRA and to the SSC reliabilities are inevitable and such changes are recognized by the 10 CFR 50.69(e) requirement for periodic updates.

In section 3.5, "Feedback and Adjustment Process," of the enclosure to its LAR, the licensee described the process for maintaining and updating the Comanche Peak PRA models used for the 10 CFR 50.69 categorization process. In its LAR supplement dated December 13, 2023, the licensee revised section 3.5 of its LAR to reflect the requirement in 10 CFR 50.9(e)(1) by stating that "[s]cheduled periodic reviews, at least once every two refueling outages, will evaluate new risk insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance." In its LAR, as supplemented, the licensee also stated that "[i]f it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and categorization process will be updated."

Consistent with NEI 00-04, the licensee confirmed that the Comanche Peak risk management process ensures that the applicable PRA mode(s) used in this application continue to reflect the as-built and as-operated plant. The licensee's process includes provisions for: monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience); assessing the risk impact of unincorporated changes; and controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization.

The NRC staff finds the risk management process described by the licensee in the LAR is consistent with sections 11 and 12 of NEI 00-04, Revision 0, guidance as endorsed by the NRC. Furthermore, considering the above, the NRC staff has determined that the proposed change satisfies the fifth key principle for risk-informed decision-making prescribed in RG 1.174, Revision 3.

4.0 CHANGES TO THE OPERATING LICENSE

Based on the NRC staff's review of the licensee's LAR and its responses to the staff's RAIs, the staff identified specific actions, as described below that are identified as being necessary to support the NRC staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69, the guidance in RG 1.201, Revision 1, and NEI 00-04, Revision 0. Note: Additional actions (e.g., final procedures and proposed alternative treatment) need not, and have not been developed, submitted, or reviewed by the NRC staff for issuance of the SE, but will be completed before implementation of the program as specified in the 10 CFR 50.69 rule.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned upon the license condition. For the clarifications to the NEI 00-04, Revision 0, guidance and other changes that were described by the licensee, the NRC staff finds to be routine and systematically addressed through the configuration management and control and periodic update processes as described in section 3.6 of this SE.

The licensee proposed the following license condition to the facility operating licenses for Comanche Peak, Units 1 and 2. The proposed license condition would state:

> Vistra OpCo is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)–1, RISC–2, RISC–3, and RISC–4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except wind-generated missiles and seismic; the high winds safe shutdown equipment list for wind-generated missiles; and the alternative seismic approach as described in Vistra OpCo's submittal letter April 19, 2023, and all its subsequent associated supplements, as specified in License Amendment No 187 dated June 10, 2024.
>
> Vistra OpCo will complete the High Winds Safe Shutdown Equipment List (HWSSEL) prior to performing any system categorization per 10 CFR 50.69.

> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

The NRC staff finds that the proposed license condition is acceptable because (1) it adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC; and (2) the evaluation in section 3.3.2 of this SE finds the non-PRA methods for assessing risk for seismic, high winds, and passive components, which are deviations from NEI-00-04, to be acceptable.

The NRC staff notes that the guidance for implementing 10 CFR 50.69 provided by the Commission in the *Federal Register* notice published on November 22, 2004 (69 FR 68008, 68028-68029),[3] Section III.4.10.2, "Section 50.36 Technical Specifications," stated that the 10 CFR 50.69 rule does not include 10 CFR 50.36 in the list of special treatment requirements that may be replaced by the alternative 10 CFR 50.69 requirements for RISC-3 and RISC-4 SSCs when implementing a 10 CFR 50.69 license amendment. As a result, the NRC staff does not consider the technical specifications (including Improved Technical Specifications and the associated Technical Requirements Manual) to be part of the 10 CFR 50.69 rule. Therefore, the licensee needs to address proposed changes to its TS separately.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendments on April 9, 2024. The State official had no comments.

6.0 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. 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, as published in *Federal Register* on August 8, 2023 (88 FR 53542), and there has been no public comment on such finding. 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.

7.0 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

---

[3] Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" Final Rule, published in the *Federal Register* on November 22, 2004 (69 FR 68008, 68028-68029).

amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

[1] Lloyd, J., Vistra Operations Company LLC, letter to the U.S. Nuclear Regulatory Commission, “Application to Adopt 10 CFR 50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors’ LAR-001,” dated April 19, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23109A333).

[2] Lloyd, J., Vistra Operations Company LLC, letter to the U.S. Nuclear Regulatory Commission, “Response to Request for Supplemental Information for License Amendment Request to Adopt 10 CFR [50.69], ‘Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors’ (EPID L-2023-LLA-0057),” dated June 8, 2023 (ML23159A200).

[3] Lloyd, J., Vistra Operations Company LLC, letter to the U.S. Nuclear Regulatory Commission, “License Amendment Request to Adopt 10 CFR 50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors,’ Supplement 2,” dated December 13, 2023 (ML23347A206).

[4] Galvin, D., U.S. Nuclear Regulatory Commission, email to J. Hicks, Vistra Operations Company LLC (Luminant), “Comanche Peak Units 1 and 2 - Audit Questions - License Amendment Request to Adopt 10 CFR 50.69 ‘Risk Informed Categorization and Treatment of Structures, Systems and Compoenents for Nuclear Power Reactors’ (EPID L 2023 LLA 0057)” and attachment, dated October 10, 2023, (ML23283A242).

[5] Galvin, D., U.S. Nuclear Regulatory Commission, letter to K. Peters, Vistra Operations Company LLC, “Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2 – Summary of Regulatory Audit Regarding a License Amendment Request to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors (EPID L-2023-LLA-0057),” dated January 29, 2024 (ML24024A210).

[6] U.S. Nuclear Regulatory Commission, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” Regulatory Guide 1.201, Revision 1, dated May 2006 (ML061090627).

[7] U.S. Nuclear Regulatory Commission, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed Activities,” Regulatory Guide 1.200, Revision 2, dated March 2009 (ML090410014).

[8] U.S. Nuclear Regulatory Commission, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Regulatory Guide 1.174, Revision 3, dated January 2018 (ML 17317A256).

[9] U.S. Nuclear Regulatory Commission, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” Final Report, NUREG-1855, Revision 1, dated March 2017 (ML17062A466).

[10] U.S. Nuclear Regulatory Commission, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance,” NUREG-0800, Section 19.2, dated June 2007 (ML071700658).

[11] Nuclear Energy Institute, “10 CFR 50.69 SSC Categorization Guideline,” NEI 00-04, Revision 0, dated July 2005 (ML052910035).

[12] Electric Power Research Institute, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk Informed Categorization,” EPRI Technical Report 3002017583, dated February 2020 (ML21082A170).

[13] Terry, C. L., TU Electric, letter to the U.S. Nuclear Regulatory Commission, “Comanche Peak Steam Electric Station (CPSES) Docket Nos. 50-445 and 50-446, NRC Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events for Severe Accident Vulnerabilities,” dated June 27, 1995 (Package ML20085N284).

[14] American Society of Mechnical Engineers (ASME) and American Nuclear Society (ANS), “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” Addendum A to ASME/ANS RA-S-2008, PRA Standard ASME/ANS RA-Sa-2009, February 2009, New York, NY (Copyright).

[15] Nuclear Management and Resources Council, “Guidelines for Industry Actions to Assess Shutdown Management,” NUMARC 91-06, dated December 1991 (ML14365A203).

[16] Markley, M. T., U.S. Nuclear Regulatory Commission, letter to Vice President, Operations, Entergy Operations, Inc., “Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative AN02-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activites in Class 2 and 3, Moderate and High Energy Safety Systems,” dated April 22, 2009 (ML090930246).

[17] McCool, T. P., Vistra Operations, Company LLC, letter to U.S. Nuclear Regulatory Commission, “Application to Revise Technical Specifciations to Adopt Risk Informed Completion Times, TSTF-505, Revision 2, ‘Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b,’“ dated May 11, 2021 (ML21131A233).

[18] Anderson, V. K., Nuclear Energy Institute, letter to S. Rosenberg, U.S. Nuclear Regulatory Commission, “Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations,” dated February 21, 2017 (ML17086A431).

[19] Electric Power Research Institute and U.S Nuclear Regulatory Commission, “Fire PRA Methodology for Nuclear Power Facilities,” Volume 1: “Summary & Overview,” NUREG/CR-6850, EPRI 1011989, dated September 2005 (ML052580075).

[20] Galvin, D., U.S. Nuclear Regulatory Commission, letter to K. Peters, Vistra Operations Company LLC, “Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amndment Nos. 183 and 183 Regarding the Adoption of Technical Speicifcations Task Force Traveler TSTF-505, Revision 2 (EPID L-2021-LLA-0085),” dated August 22, 2022 (ML22192A007).

[21] Zoulis, A. M., U.S. Nuclear Regulatory Commission, memoradum to M. Franovich, U.S. Nuclear Regulatory Commission, “Updated Assessment of Industry Guidance for Crediting Mitgating Stategies in Probablistic Risk Assessment,” dated May 6, 2022 (ADAMS Accession No. ML22014A084).

[22] Barstow, J., Exelon Generation, letter to U.S. Nuclear Regulatory Commission Regarding Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Revised submittal to Application to Adopt 10 CFR 50.69, ‘Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,’“ dated May 10, 2019 (ML19130A180).

[23] Barstow, J., Exelon Generation, letter to U.S. Nuclear Regulatory Commission Regarding Calvert Cliffs Nuclear Power Plant, Units 1 and 2, “Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, ‘Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,’“ dated July 1, 2019 (ML19183A012).

[24] Barstow, J., Exelon Generation, letter to U.S. Nuclear Regulatory Commission Regarding Calvert Cliffs Nuclear Power Plant, Units 1 and 2, “Response to Request for Additional

Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,'" dated July 19, 2019 (ML19200A216).

[25] Barstow, J., Exelon Generation, letter to U.S. Nuclear Regulatory Commission Regarding Calvert Cliffs Nuclear Power Plant, Units 1 and 2, "Revised Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'" dated August 5, 2019 (ML19217A143).

[26] Marshall, M. L., U.S. Nuclear Regulatory Commission, letter to B. C. Hanson, Exelon Generation Company, LLC, "Calvert Cliffs Nuclear Power Plant, Units 1 and 2 - Issuance of Amendment Nos. 332 and 310 RE: Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors (EPID L-2018-LLA-0482)," dated February 28, 2020 (ML19330D909).

[27] Madden, F. W., Luminent Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Nuclear Power Plant, Docket Nos. 50-445 AND 50-446, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f), Regarding Recommendation 2.1 of the NearTerm Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 27, 2014 (ML14099A197).

[28] U.S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, dated June 1991 (ML063550238).

[29] Madden, F. W., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Nuclear Power Plant (CPNPP), Docket Nos. 50-445 and 50-446,, Submittal of Fukushima Lessons Learned - Flood Hazard Reevaluation Report Supplement 1 (TAC Nos. MF1099 and MF1100)," dated August 14, 2014 (ML14245A136).

[30] American Society of Mechanical Engineers, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," ASME Code Case, N-660, July 2002.

Principal Contributors: Ching Ng, NRR
Dan Widrevitz, NRR
De Wu, NRR
Diana Woodyatt, NRR
Gurjendra Bedi, NRR
Hanry Wagage, NRR
Keith Tetter, NRR
Mihaela Biro, NRR
Ming Li, NRR
Nageswara Karipineni, NRR
Stephen Cumblidge, NRR
Vijay K Goel, NRR

Date: June 10, 2024

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 187 AND 187, RESPECTIVELY RE: ADOPTION OF 10 CFR 50.69, “RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS” (EPID L-2023-LLA-0057) DATED JUNE 10, 2024

DISTRIBUTION:
PUBLIC
RidsACRS_MailCTR Resource
RidsNrrDorlLpl4 Resource
RidsNrrDexEeeb Resource
RidsNrrDexEicb Resource
RidsNrrDexEmib Resource
RidsNrrDnrlNphp Resource
RidsNrrDnrlNvib Resource
RidsNrrDraApla Resource
RidsNrrDraAplc Resource
RidsNrrDssScpb Resource
RidsNrrDssSnsb Resource
RidsNrrLAPBlechman Resource
RidsNrrPMComanchePeak Resource
RidsRgn4MailCenter Resource
CNg, NRR
DWidrevitz, NRR
DWu, NRR
DWoodyatt, NRR
GBedi, NRR
HWagage, NRR
KTetter, NRR
MBiro, NRR
MLi, NRR
NKaripineni, NRR
SCumblidge, NRR
VGoel, NRR

ADAMS Accession No.: ML24120A363 *concurrence by email NRR-058

OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA*	NRR/DRA/APLA/BC*	NRR/DRA/APLC/BC*
NAME	SLee	PBlechman	RPascarelli	SVasavada
DATE	4/29/2024	5/7/2024	4/5/2024	4/5/2024
OFFICE	NRR/DEX/EEEB/BC*	NRR/DEX/EICB/BC*	NRR/DEX/EMIB/BC*	NRR/DSS/SCPB/BC*
NAME	WMorton	FSacko	SBailey	BWittick
DATE	4/23/2024	4/12/2024	4/19/2024	4/14/2024
OFFICE	NRR/DSS/SNSB/BC*	NRR/DNRL/NPHP/BC*	NRR/DNRL/NVIB/BC*	OGC (NLO)*
NAME	PSahd	MMitchell	ABuford	MCarpentier
DATE	4/9/2024	4/10/2024	4/22/2024	5/24/2024
OFFICE	NRR/DORL/LPL4/BC*	NRR/DORL/LPL4/PM*		
NAME	JRankin	SLee		
DATE	6/10/2024	6/10/2024		

OFFICIAL RECORD COPY