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

August 13, 2026

Mr. Cleveland Reasoner
Chief Executive Officer
and Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 248 RE: ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2025-LLA-0017)

Dear Mr. Reasoner:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 248 to Renewed Facility Operating License (RFOL) No. NPF-42 for the Wolf Creek Generating Station, Unit 1 (Wolf Creek). The amendment consists of changes to the RFOL in response to your application dated January 30, 2025, as supplemented by letters dated February 3, 2026, and June 25, 2026.

The amendment revises Wolf Creek RFOL No. NPF 42 to add a new license condition to allow for 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." The provisions in 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated, systematic, risk-informed process for categorizing SSCs according to their safety significance.

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

Sincerely,

/RA/

Samson S. Lee, Project Manager
Operating Reactor Licensing Branch 4
Division of Licensing Projects I
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Amendment No. 248 to NPF-42
2. Safety Evaluation

cc: Listserv

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 248
License No. NPF-42

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

   A. The application for amendment to the Wolf Creek Generating Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated January 30, 2025, as supplemented by letters dated February 3, 2026, and June 25, 2026, 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, as amended, 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 paragraph 2(C)16 of Renewed Facility Operating License No. NPF-42 is hereby amended to read as follows:

(16) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 248, are hereby incorporated into this license. Wolf Creek Nuclear Operating Corporation shall operate the facility in Accordance with the Additional Conditions.

In addition, the license is amended by changes as indicated in the attachment to this license amendment and Appendix D, “Additional Conditions,” to Renewed Facility Operating License No. NPF-42 is hereby amended to include a new license condition to read as follows:

WCNOC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC) 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 Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components; the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic and high winds; and the alternative seismic and high winds approaches described in WCNOC’s submittal letter 000779 dated January 30, 2025, as supplemented by letters 001182 and 001656 dated February 3, 2026, and June 25, 2026, respectively, as specified in License Amendment No. 248 dated August 13, 2026.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

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

FOR THE NUCLEAR REGULATORY COMMISSION

HIPOLITO GONZALEZ

Digitally signed by HIPOLITO GONZALEZ
Date: 2026.08.13 09:44:21 -04'00'

Hipólito González, Chief
Operating Reactor Licensing Branch 4
Division of Licensing Projects I
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF 42

Date of Issuance: August 13, 2026

ATTACHMENT TO LICENSE AMENDMENT NO. 248 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

Replace the following pages of Renewed Facility Operating License No. NPF-42 and Appendix D – Additional Conditions, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
-7-	-7-

Appendix D – Additional Conditions

REMOVE	INSERT
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(16) Additional conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 248, are hereby incorporated into this license. Wolf Creek Nuclear Operating Corporation shall operate the facility in Accordance with the Additional Conditions.

D. Exemptions from certain requirements of Appendix J to 10 CFR Part 50, and from a portion of the requirements of General Design Criterion 4 of Appendix A to 10 CFR Part 50, are described in the Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The set of combined plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: “Wolf Creek Security Plan, Training and Qualification Plan, and Safeguard Contingency Plan,” and was submitted on May 17, 2006.

The licensee 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). The licensee’s CSP was approved by License Amendment No. 197, as supplemented by changes approved by License Amendment No. 202, License Amendment No. 210, and License Amendment No. 217.

F. Deleted per Amendment No. 141.

G. The licensees 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.

H. The Updated Safety Analysis Report (USAR) supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the USAR required by 10 CFR 50.71(e)(4), as appropriate, following the issuance of this renewed operating license. Until that update is complete, WCNOC may make changes to the programs and activities described in the supplement without prior Commission approval, provided that WCNOC evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

Amendment Number	Additional Condition	Implementation Date
248	WCNOC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC) 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 Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components; the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic and high winds; and the alternative seismic and high winds approaches described in WCNOC's submittal letter 000779 dated January 30, 2025, as supplemented by letters 001182 and 001656 dated February 3, 2026, and June 25, 2026, respectively, as specified in License Amendment No. 248 dated August 13, 2026. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.	The amendment shall be implemented within 90 days from the date of issuance.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 248 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

1.0 INTRODUCTION AND PROPOSED LICENSE CONDITION

By letter dated January 30, 2025 (Reference 1), as supplemented by letters dated February 3, 2026 (Reference 2), and June 25, 2026 (Reference 3), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted a license amendment request (LAR) for Wolf Creek Generating Station, Unit 1 (Wolf Creek). The licensee proposed the following license condition to the Renewed Facility Operating License (RFOL) 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."

> WCNOC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC) 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 Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components; the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic and high winds; and the alternative seismic and high winds approaches described in WCNOC's submittal letter 000779 dated January 30, 2025, as supplemented by letters 001182 and 001656 dated February 3, 2026, and June 25, 2026, respectively, as specified in License Amendment No. 248 dated August 13, 2026.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

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]

A notice of consideration of proposed issuance of amendment with proposed no significant hazards consideration determination was published in the *Federal Register* on April 15, 2025 (90 FR 15732), and there has been no public comment on such determination. The supplemental letters dated February 3, 2026, and June 25, 2026, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in this notice.

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. For SSCs categorized as low safety significance (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety significance (HSS), requirements may not be changed.

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 are categorized as HSS, existing treatment requirements are maintained or potentially enhanced. Conversely, for SSCs categorized as LSS that 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 has HSS.

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[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 (Reference 4), 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.

2.2 Regulatory Guides

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," May 2006 (Reference 4).
- RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (Reference 5).
- RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018 (Reference 6).
- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2017 (Reference 7).
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (Reference 8).

NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 9), as 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:

- Section 3.2, "Use of Risk Information," and section 5.1, "Internal Events Assessment," provide specific guidance corresponding to 10 CFR 50.69(c)(1)(i).
- Section 3,"Assembly of Plant-Specific Inputs"; section 4, "System Engineering Assessment"; section 5,"Component Safety Significance Assessment"; and section 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).

- Section 9, “IDP [Integrated Decisionmaking Panel] review and Approval,” and section 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 (Reference 6). 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 change results 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

Paragraph 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 HSS functions (i.e., RISC-1 and RISC-2 categories) or 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). For RISC-2 components, existing treatment is evaluated and then maintained or supplemented as needed to provide reasonable assurance that the component can perform the safety-significant functions identified in the 10 CFR 50.69 categorization process. For LSS SSCs, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d). For RISC-3 SSCs, licensees can replace special treatment with an alternative treatment. For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements.

Paragraph 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). 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 portion of 10 CFR 50.46a(b)
(iii) 10 CFR 50.49
(iv) 10 CFR 50.55(e)
(v) specified 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) specified requirements for containment leakage testing
(xi) specified 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.

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[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.

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. IDP 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, “Categorization Process Description (10 CFR 50.69(b)(2)(i)),” of enclosure I to its LAR (Reference 1), the licensee stated that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0 (Reference 9), as endorsed in RG 1.201, Revision 1 (Reference 4). In section 3.2.3, “Seismic Hazards”; section 3.2.4, “High Winds Hazards”; and section 3.2.5, “Other External Hazards,” of enclosure I to its LAR, the licensee has proposed the use of the alternative seismic approach as described in Electric Power Research Institute (EPRI) Technical Report (TR) 3002017583, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,” February 2020 (Reference 10), a conservative high winds screening approach described in the LAR, and the Individual Plant Examination of External Events (IPEEE) screening method for external hazards (Reference 11) with subsequent screening using the process described in PRA standard ASME/ANS RA-Sa-2009, “Addenda to ASME/ANS RA-S-2008, Standard for Level1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” (Reference 12), respectively, as alternative methods to assess the applicable hazard contribution(s), respectively. In the supplement dated February 3, 2026 (Reference 2), the licensee proposed the use of the passive pressure-boundary categorization method described in EPRI TR 3002033536 Revision 1-A, “Enhances Risk-Informed Categorization Methodology for Pressure Boundary Components,” August 2025 (Reference 13). The NRC staff 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 0 (References 9 and 4, respectively).

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, ensure 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.

The NRC staff notes that the guidance for implementing 10 CFR 50.69 provided by the Commission in the *Federal Register* published on November 22, 2004 (69 FR 68008, 68028-68029), 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 technical specifications separately.

In light of the above, the NRC 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 (Reference 6).

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

In RG 1.174, Revision 3, the NRC staff 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, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” 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, “Overall Categorization Process,” of enclosure I to its LAR, the licensee clarified that it will require an SSC to be categorized as HSS based on the defense-in-depth assessment performed in accordance with NEI 00-04, Revision 0. Considering the 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: Licensing Basis 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 plant's licensing basis, including the Updated Final Safety Analysis Report (UFSAR) and technical specification 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's licensing basis. On this basis, the NRC 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 00-04, Revision 0, Section 4)

In section 2.2, Reason for the Proposed Change," of enclosure I to its LAR, the licensee stated, in part, that "[t]he safety functions include the design basis functions, as well as functions credited for severe accidents (including external events)." Section 3.1.1 of enclosure I 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, addresses the fourth and fifth key principles of the NRC staff's standards 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:

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 the 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 and 3.2 of enclosure I to its LAR, the licensee described that the Wolf Creek categorization process uses PRA modeled hazards to assess risks for internal events (including internal flood) and internal fires. For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

- Seismic Risks: Alternative Seismic Approach described in EPRI TR 3002017583 (Reference 10).

- High Winds: Conservative approach described in the LAR (Reference 1).

- External Hazards (excluding high winds): Screening analysis performed for IPEEE (Reference 11) updated using criteria from Part 6 of the 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" (the PRA Standard) (Reference 12), as endorsed by the NRC.

- Passive Pressure Boundary Components: EPRI Pressure Boundary Passive Categorization Method, EPRI TR 3002033536-A (Reference 13).

- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06 (Reference 14).

The approaches and methods proposed by the licensee to address internal events (including internal flooding), fire hazards, and defense-in-depth, are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0. The non-PRA methods for an alternative seismic approach, the conservative high winds approach, external hazards, passive pressure boundary categorization, and shutdown events are consistent with the methods described above.

3.3.1 Scope of the PRA

The Wolf Creek PRA is comprised of a full-power, Level 1, internal events PRA (including internal flooding) and fire PRA, which evaluates 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 enclosure I to its LAR that the internal events PRA including internal flooding model has been assessed against RG 1.200, Revision 2. Furthermore, in section 3.3 of its LAR, the licensee states that a finding closure review was conducted on the internal events PRA and fire PRA model using this process. 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 15), as accepted by NRC by letter dated May 3, 2017 (Reference 16).

The NRC staff finds that the information provided in the LAR to support the staff review of the internal events PRA (including internal flooding) and fire PRA for technical acceptability meets the requirements set forth in 10 CFR 50.69(b)(2)(iii).

Aspects considered by the NRC staff to evaluate the scope of the PRA include: (1) peer-review history and results, (2) the Appendix X, Independent Assessment process (Reference 15), (3) credit for the diverse and flexible coping strategy (FLEX) in the PRA, and (4) assessment of assumptions and approximations.

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

In section 3.3 of enclosure I to its LAR, the licensee stated that the internal events PRA model was subjected to a full-scope peer review in June 2019 (Reference 17). Subsequently, in November 2023, the licensee conducted an Independent Assessment for closure of the finding-level facts and observations (F&Os) and concluded all the internal events PRA (including internal flooding) F&Os have been closed (Reference 18). Final closure of F&Os was documented in November 2023 (Reference 19).

In section 3.2, “Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii)),” of enclosure I to its LAR, for the internal events PRA (including internal flooding), the licensee stated in part, “there are no PRA upgrades that have not been peer reviewed.” During the regulatory audit (Reference 20), the NRC staff verified that all F&Os were appropriately assessed by the Independent Assessment team to assure that no new methods or upgrades were inadvertently incorporated into the internal events PRA without a peer review in accordance with the ASME/ANS RA-Sa-2009 PRA standard as endorsed by the NRC (Reference 12).

Therefore, the NRC staff concludes that the Wolf Creek internal events PRA (including internal floods) was appropriately peer reviewed, consistent with RG 1.200, Revision 2 (Reference 5), and the F&Os have been adequately dispositioned to assess the impact on the risk-informed application.

Internal Fire PRA Peer-Review History

In section 3.3.2, “Fire PRA Model,” of enclosure I to its LAR, the licensee stated that the Wolf Creek fire PRA was subject to a full-scope industry peer review in January 2022 (Reference 21), using the process described in NEI 17-07 (Reference 22), and consistent with RG 1.200, Revision 2. Subsequently, a focused peer review was performed to address ASME/ANS RA-Sa-2009 PRA standard supporting requirements.

The finding-level F&Os from the January 2022 full-scope review and subsequent focused-scope peer review were considered fully resolved by an Independent Assessment review team in an independent assessment of F&Os closure of the Wolf Creek fire PRA published in November of 2022 (Reference 23). A subsequent special scope seismic-fire interaction fire PRA focused-scope peer review was conducted in November 2023, where no additional findings were identified (Reference 24). Therefore, in accordance with RG 1.200, Revision 2, no F&Os associated with the fire PRA were provided in attachment 3 of enclosure I to the LAR. Section 3.2.2, “Fire Hazards,” of enclosure I to its LAR, the licensee stated, in part, “[t]he internal Fire PRA model was developed consistent with NUREG/CR-6850 (Reference 25), RG 1.200 (Reference 5), and the 2009 ASME/ANS PRA Standard” (Reference 12).

The NRC staff has reviewed the fire PRA peer review results and the licensee's resolution of the results concludes that the Wolf Creek fire PRA was appropriately peer-reviewed, consistent with RG 1.200, Revision 2, and the F&Os have been adequately dispositioned to assess the impact on the risk-informed application.

External Events: Seismic PRA, High Winds PRA, and External Events Review History

The regulation in 10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For other risk hazards, such as seismic risk and external events, 10 CFR 50.69(b)(2) allows, and NEI 00-04 summarizes, the use of other methods for determining SSC functional important in the absence of a quantifiable PRA.

In section 3.2 of enclosure I to the LAR, the licensee stated that the proposed categorization process will use non-PRA models for seismic, high winds, external events (excluding high winds), and passive pressure boundary components, and shutdown events. The NRC staff’s review of the technical acceptability of these models for this application is discussed below.

Seismic PRA Peer-Review History

In section 3.2.3 of enclosure I to its LAR, the licensee proposes to use a risk-informed graded approach specified in EPRI TR 3002017583 (Reference 10). The seismic hazard approach was reviewed by NRC staff in section 3.3.2 of this SE.

High Winds PRA Peer-Review History

In section 3.2.4 of enclosure I to its LAR, the licensee proposed to use a conservative approach to consider high winds hazards in lieu of the licensee’s current high winds PRA, which has been peer-reviewed but is currently pending upgrade. The conservative approach to address high winds was reviewed by NRC staff in section 3.3.2 of this SE.

Other External Hazards

In section 3.2.5 of enclosure I to its LAR, the licensee states that external events, including external flooding, were initially screened and addressed per the IPEEE hazards assessment (Reference 11), and that external hazards were further screened for applicability to Wolf Creek using a plant-specific evaluation identified in Part 6, section 5.2 of the 2009 ASME/ANS PRA Standard (Reference 12). In section 3.2.5, the licensee stated that the Wolf Creek external hazards screening process was peer reviewed in December 2015 (Reference 26). The external hazards screening process was reviewed by NRC staff in section 3.3.2 of this SE.

<u>Appendix X, Independent Assessment Process for F&O Closure</u>

Based on its review regarding the information in section 3.3 of enclosure I to the LAR, the NRC staff concluded that all F&Os were appropriately assessed by the Independent Assessment team to assure that no new methods or upgrades were inadvertently incorporated into the internal events PRA without a peer review in accordance with the 2009 ASME/ANS PRA standard (Reference 12) as endorsed by the NRC. Therefore, the NRC staff finds that the internal events PRA and fire PRA were appropriately peer reviewed consistent with RG 1.200, Revision 2 and meet the requirements set forth in 10 CFR 50.69(c)(1)(i).

Credit for Diverse and Flexible Mitigation Strategies (FLEX) Equipment

The NRC memorandum dated May 6, 2022, “Updated Assessment of the Industry Guidance for Crediting Mitigating Strategies in Probabilistic Risk Assessments” (Reference 27), provides the NRC staff’s assessment of incorporating the 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 section 3.2.9, “Modeling of FLEX,” of enclosure I to its LAR, the licensee stated that “No credit is taken for FLEX strategies within any WCNOC PRA model.” Therefore, the NRC staff finds that the licensee’s internal events PRA and fire PRA do not credit FLEX equipment for the SSC categorization process; therefore, no new methods or upgrades were inadvertently incorporated into the internal events PRA or fire PRA without a peer review consistent with the 2009 ASME/ANS PRA standard (Reference 12), as endorsed by the NRC.

Assessment of Assumptions and Approximations

Identification of Key Assumptions and Sources of Uncertainty

In the supplement to the LAR, dated February 3, 2026 (Reference 2), the licensee confirmed that uncertainty analyses were developed for the internal events PRA (including internal flood) and the fire PRA, in accordance with industry guidance, including NEI 00-04 (Reference 9); EPRI TR-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments” (Reference 28); EPRI-1026511, “Practical Guidance on the Use of Probabilistic Risk Assessments in Risk-Informed Application with a Focus on the Treatment of Uncertainty” (Reference 29); and NUREG-1855, Revision 1 (Reference 7). Uncertainty analyses were performed to identify, screen, and characterize those sources of model uncertainty and related assumptions in the base PRA that are relevant to this application. The analyses were performed for both generic and plant-specific sources of uncertainty. The supplement to the LAR dated February 3, 2026, identified three generic sources of uncertainty and 28 potential plant-specific key sources of uncertainty. 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. In accordance with NEI 00-04, the results of the sensitivity study are given to the (IDP for consideration in the final risk characterization for components initially classified as LSS that may be reclassified to HSS. In the supplement to the LAR dated February 3, 2026, the licensee included a list of both generic and plant-specific potential sources of uncertainty and disposition of each potential source of uncertainty.

The licensee confirmed that sensitivity studies for the generic sources of uncertainty (i.e., basis for human error probabilities, treatment of human failure event dependencies, and intra-system common-cause events), will be performed consistent with table 5-2 of the NEI 00-04 guidance. For each plant-specific source of uncertainty associated with the internal events PRA (including internal flooding) and fire PRA, the licensee performed sensitivity studies according to the process described in NEI 00-04 and summarized the results of the sensitivity study in a table in the supplement to the LAR dated February 3, 2026. The results of these sensitivity studies showed that for most sources of uncertainty, there was an insignificant change to the PRA results, and therefore no change to the licensee's 10 CFR 50.69 program is necessary. For several sources of uncertainty, the associated function is explicitly modeled in the licensee's internal events PRA and fire PRA, and further sensitivity studies will be conducted during the 10 CFR 50.69 categorization process for associated components. Therefore, the processes included in the 10 CFR 50.69 program are sufficient to address key assumptions and sources of uncertainty.

The NRC staff finds that potential key sources of uncertainty either do not have significant impact on the internal events PRA (including internal flooding) or fire PRA models of record, or that those uncertainties will be addressed via sensitivity studies conducted during the 10 CFR 50.69 categorization process at the time of the risk analysis being performed.

In addition, the NRC staff recognizes that the licensee will perform routine PRA changes and updates to ensure 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). Paragraphs 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 Importance Measures and Integrated Importance Measures

The scope of modeled hazards for Wolf Creek includes the internal events PRA (including internal flooding) and fire PRA. The NRC staff finds that the licensee's use and treatment of importance measures is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. A more detailed NRC staff review of the alternate methods for assessing the risk for seismic hazard, high winds, and other external hazards is provided in section 3.3.2 of this SE.

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 internal events PRA (including internal flooding) and fire PRA 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 2009 ASME/ANS PRA standard (Reference 12) using a peer-review process.

The licensee has subjected the internal events PRA (including internal flooding) and fire PRA to the peer-review processes and submitted the results of the peer review. The NRC staff reviewed the peer-review history, the licensee's resolution of peer-review findings, and the identification and disposition of key assumptions and sources of uncertainty. The NRC staff concludes that (1) the licensee's internal events PRA (including internal flooding) and fire PRA 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 internal events PRA (including internal floods) and fire PRA, are acceptable and therefore meet the requirements set forth in paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR.

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:

- EPRI Alternate Approach for Seismic Risk (Reference 10)
- Plant-specific conservative approach for high winds (Reference 1)
- Screening analysis for other external hazards (References 11 and 12)
- Low-Power and Shutdown Risk (Reference 14)
- EPRI Passive Components: EPRI passive pressure boundary component method described in EPRI TR 3002033536 Revision 1-A (Reference 13).

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 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 in section 3.2, of enclosure I to its LAR. In section 3.2, the licensee stated that the proposed approach for Tier 2 sites is documented in EPRI Report 3002017583 (Reference 10), with the EPRI markups provided in attachment 2 of letters dated October 16, 2020, and January 21, 2021 (References 30 and 31, respectively). The NRC staff notes that use of the alternative seismic approach is a method not endorsed by the NRC in NEI 00-04. Therefore, a detailed NRC staff review of the licensee’s proposed plant-specific alternative seismic approach is provided below.

The licensee based its plant-specific evaluation, in part, on the test cases described in EPRI Report 3002017583 (Reference 10) to determine the extent and type of unique HSS SSCs from seismic PRAs. The licensee stated that the test cases are applicable to Wolf Creek and demonstrated that there are very few, if any, SSCs that would be designated HSS for seismic unique purposes. Furthermore, the test cases identified that the unique seismic insights were typically associated with seismically correlated failures and led to unique HSS SSCs The licensee further stated that the categorization team will evaluate correlated seismic failures and seismic interactions between SSCs as described in EPRI Report 3002017583 and the referenced markups. The licensee stated that the determination of seismic insights will make use of the full power internal events PRA model supplemented by focused seismic walkdowns, and the licensee described the steps of the process used to determine the importance of SSCs for mitigating seismic events on a system basis. Based on the above, the NRC staff’s independent review finds that the requirements in 10 CFR 50.69(b)(2)(ii) for the proposed plant-specific alternative seismic approach are met because the licensee sufficiently described the measures taken to assure the quality and level of detail of their proposed alternative seismic approach to evaluate the plant for the categorization of SSCs.

In section 3.2 of enclosure I to its LAR, the licensee cited a precedent for its proposed alternative seismic approach. The precedent, the LaSalle alternative seismic approach, as described in Exelon's letter dated January 31, 2020 (Reference 32), and all its subsequent associated supplements, as specified in License Amendment Nos. 249 and 235 for LaSalle County Station, Unit Nos. 1 and 2 (LaSalle) dated May 27, 2021 (Reference 33), includes a combination of qualitative and quantitative considerations of the mitigation capabilities as well as seismic failure modes of SSCs in the categorization process. These considerations are based on plant-specific walkdowns for the SSCs undergoing categorization, quantification of the impact of seismic failure of SSCs subject to correlated or interaction failures, and insights obtained from prior seismic evaluations performed at LaSalle. The licensee identified differences between Wolf Creek and the NRC staff's approval of the precedent as documented in the LaSalle SE. The licensee described that, except for the site-specific LaSalle information (e.g., seismic capacity discussions, etc.), the licensee would follow the same alternative seismic approach in its proposed 10 CFR 50.69 categorization process as the approved plant-specific LaSalle 10 CFR 50.69 license amendment.

The NRC staff notes that the LaSalle alternative seismic approach was supported by EPRI Report 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," which is not publicly available. In section 3.2 of the enclosure to the LAR, the licensee stated that EPRI Report 3002017583 (Reference 10) is an update to EPRI Report 3002012988 and the technical criteria in EPRI Report 3002017583 are unchanged from its predecessor, EPRI Report 3002012988.

Based on information provided by the licensee in the LAR enclosure, the NRC staff understands that the technical criteria in EPRI Report 3002017583 are unchanged from its predecessor EPRI Report 3002012988, and that the test cases are applicable to Wolf Creek and are, therefore, justified for use in the licensee's proposed plant-specific alternative seismic approach. The NRC staff confirmed that the test cases in EPRI Report 3002017583 used by the licensee to support its proposed alternative seismic approach provided a sufficient plant-specific evaluation of the applicability and differences for Wolf Creek as compared to the plant-specific approach approved by the NRC for LaSalle. The NRC staff's independent review finds that the information presented in the licensee's LAR and licensee's analysis for items referenced in EPRI Report 3002017583 provides 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 Wolf Creek plant-specific alternative seismic approach. The NRC staff determined that there is reasonable confidence that evaluated LSS safety-related SSCs would have sufficient safety margins maintained, and that any potential increases in CDF and LERF resulting from the changes in SSC treatment are small. Based on the above, the NRC staff's independent review finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the proposed plant-specific alternative seismic approach.

Based on the information provided in the licensee's submittal, the NRC staff's independent review finds that the licensee provided acceptable bridging analysis for the Wolf Creek plant-specific alternative seismic approach, in part, based on analysis supporting approval of LaSalle's plant-specific alternative seismic approach because (1) the differences between the licensee's proposed alternative seismic approach and the alternative seismic approach previously approved for LaSalle are addressed; (2) there are no differences in the technical criteria used in EPRI Report 3002017583 and its predecessor, EPRI Report 3002012988, for use in this application; and (3) all references needed to support the NRC staff's finding on the proposed alternate seismic approach have been cited by the LAR.

Evaluation of Technical Acceptability of the PRAs Used for Test Cases Supporting the Proposed Alternative Seismic Approach

In section 3.2 of enclosure I to its LAR, the licensee discussed the precedent, including the test cases (also referred to as trial studies in EPRI Report 3002017583 (Reference 10)), mapping approach, and conclusions on the determination of unique HSS SSCs from the test cases, which were used by the licensee to support its proposed plant-specific alternative seismic approach. The licensee stated that Wolf Creek is using test case information from EPRI Report 3002017583, which demonstrated that seismic risk is adequately addressed for Tier 2 sites. The licensee stated that Tier 2 seismic demand sites have a lower likelihood of seismically induced failures and less challenges to plant systems than trial study plants. The NRC staff's independent evaluation of the Tier 2 criteria for Wolf Creek is provided in the following section. The NRC staff reviewed and evaluated the technical acceptability of the PRAs used in the test cases for Plants A, C, and D in EPRI Report 3002017583, as well as the applicability of these test cases to Wolf Creek. The NRC staff also evaluated the peer review process, resolution of peer review findings, and key assumptions and sources of uncertainty for Plants A, C, and D.

Based on the above, the NRC staff's independent review finds the technical acceptability of PRAs used for the Plant A, C, and D test cases in EPRI Report 3002017583; the mapping approach used in those test cases; and the conclusions on the determination of unique HSS SSCs from the test cases in the precedent are applicable to Wolf Creek's proposed plant-specific alternative seismic approach. Therefore, the NRC staff concludes that the Plant A, C, and D PRAs were technically acceptable and applicable for use in support of the licensee's proposed alternative seismic approach and the mapping of SSCs between the SPRA, the full power internal events PRA, and, as applicable, the FPRA for the Plant A, C, and D test cases. The NRC staff's independent review further finds that the licensee's plant-specific evaluation is technically justified to support conclusions on the determination of unique HSS SSCs from seismic PRAs in the Plant A, C, and D test cases in EPRI Report 3002017583, and that the licensee's proposed plant-specific alternative seismic approach is applicable to Wolf Creek.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In section 3.2 of enclosure I to its LAR, the licensee provided the basis for Wolf Creek being a Tier 2 plant. The licensee referred to the following criteria provided in EPRI Report 3002017583 (Reference 10):

- Tier 1: Plants where the ground motion response spectrum (GMRS) peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the safe shutdown earthquake (SSE) between 1.0 Hz and 10 Hz.

- Tier 2: Plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3. At these sites, the unique seismic categorization insights are expected to be limited.

- Tier 3: Plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform a seismic PRA to respond to the 10 CFR 50.54(f) letter regarding the Fukushima Dai-Ichi Accident (Reference 34).

The licensee explained that the Tier 2 criterion is based on a comparison of the GMRS to SSE. The licensee compared the GMRS to the SSE and explained that the GMRS to SSE comparison is above the Tier 1 threshold under criteria in the EPRI report but is not high enough for the NRC to require the plant to perform an seismic PRA to respond to Near-Term Task Force (NTTF) Recommendation 2.1. This determination on the licensee's plant-specific evaluation is supported by its response to NTTF Recommendation 2.1 (Reference 35). The NRC staff reviewed the plant-specific evaluation and concludes that the proposed Tier 2 criteria to determine the applicability and use of the proposed alternative seismic approach is acceptable. Based on its review of the GMRS to SSE comparison, the NRC staff finds that the licensee's seismic hazard meets the criteria for its proposed alternative seismic approach.

Evaluation of the Implementation of Conclusions from the Case Studies

The categorization conclusions supporting the request for Wolf Creek indicated that seismic specific failure modes resulted in HSS categorization uniquely from seismic PRAs. Seismic specific failure modes, such as correlated failures, relay chatter, and passive component structural failure modes, can influence the categorization process. The NRC staff reviewed the proposed alternative seismic approach for Wolf Creek to evaluate whether the categorization-related conclusions were assessed appropriately and provisions for implementation were sufficient.

In section 3.2 of enclosure I to its LAR, the licensee described the implementation of the proposed alternative seismic approach. For HSS SSCs uniquely identified by the PRA models but having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, the licensee stated that it will address these SSCs using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA. For components that are HSS due to the fire PRA, but not HSS due to the full power internal events PRA, the licensee stated that the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these as additional qualitative inputs to the IDP.

The licensee described that the categorization team will evaluate plant-specific seismic insights. The licensee stated that the objective of the seismic reviews is to identify plant-specific seismic insights that might include potentially important impacts such as:

- Impact of relay chatter.
- Implications related to potential seismic interactions such as with block walls.
- Seismic failures of passive SSCs such as tanks and heat exchangers.
- Any known structural or anchorage issues with a particular SSC.
- Components implicitly part of PRA-modeled functions (including relays).

Wolf Creek stated that it will follow the same categorization process that the NRC approved as precedent for LaSalle and will incorporate the LaSalle request for additional information (RAI) responses (References 30 and 31). Wolf Creek described its exceptions to the precedent, including LaSalle site-specific information, as evaluated above. The NRC staff reviewed the licensee's proposed alternative seismic approach and determined that the NRC-approved precedent is applicable to the proposed alternative seismic approach and the plant-specific evaluation on the implementation of the alternative seismic approach is acceptable. Based on its independent review of the licensee's proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding Statement of Consideration (Reference 35), the NRC staff finds that the proposed alternative seismic approach provides reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(ii) as well as 10 CFR 50.69(c)(1)(iv) because:

- 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.

- 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 means to consider potential impacts of seismic events in the categorization process.

- 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 seismic PRAs 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.

- The proposed alternative seismic approach includes qualitative considerations 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 test cases supporting this application.

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

An important input to the NRC staff's evaluation of the proposed alternative seismic approach is the current knowledge of the seismic hazard at Wolf Creek. The possibility exists for the seismic hazard at the site to increase or decrease such that the criteria for use of the proposed alternative seismic approach are challenged. In such a situation, the categorization process may be impacted from a seismic risk perspective either solely due to the seismic risk or by the integrated importance measure determination.

In section 3.2 of enclosure I to its LAR, the licensee stated, in part:

> If the [Wolf Creek] seismic hazard changes from medium risk (i.e., Tier 2) at some future time and the feedback process determines that a process different from the proposed alternative seismic approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69, prior NRC approval, pursuant to 10 CFR 50.90, will be requested.

The licensee also stated that it will follow its categorization review and adjustment process to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e) and the EPRI Report 3002017583 SSC categorization criteria for the updated tier, and that this includes use of the licensee's corrective action process. The licensee described the processes it will follow if the seismic hazard is reduced such that it meets the criteria for Tier 1 in EPRI Report 3002017583, or is increased to a degree that an seismic PRA becomes necessary to demonstrate adequate seismic safety.

The NRC staff's independent review finds that the consideration of changes to the seismic hazard in the licensee's proposed alternative seismic approach is consistent with the applicable regulations and the precedent approved by NRC staff for LaSalle. Therefore, the NRC staff's evaluation of the proposed changes to Wolf Creek's seismic hazard against the requirements in 10 CFR 50.69(e)(1), 10 CFR 50.69(e)(3), and 10 CFR 50.69(d)(2)(ii), as well as the resulting conclusion on consideration of changes to the seismic hazard for the NRC approved precedent is applicable to this licensee's proposed alternative seismic approach. Consequently, the NRC staff finds that the consideration of changes to the seismic hazard at Wolf Creek is acceptable 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 as discussed above, and (4) the license condition discussed in section 4.0 of this SE requires prior NRC approval if the licensee changes any of the approaches used in the categorization process, including the seismic method.

<u>Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach</u>

In section 3.5, "Feedback and Adjustment Process," of enclosure I to its LAR, the licensee described its feedback and adjustment (i.e., performance monitoring) process. The licensee described its performance monitoring process, configuration control process, and problem identification and resolution process. In the LAR, the licensee stated, in part:

> To more specifically address the feedback and adjustment (i.e., performance monitoring) process as it pertains to the proposed alternative seismic method for Tier 2 sites …, implementation of the [Wolf Creek] design control and corrective action programs provide assurance that the inputs for the qualitative determinations for seismic continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e).

Furthermore, the licensee then goes on to state, in part:

> Scheduled periodic reviews will be completed at least once every two complete refueling cycles and will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization), design changes, operational changes, and SSC performance. If 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 the categorization process will be updated.

The NRC staff's independent review finds that consideration of the feedback and adjustment process in the licensee's proposed alternative seismic approach is acceptable because (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 seismic capacity would be detected and addressed before significantly impacting the plant's risk profile. Therefore, the NRC staff's independent review finds that the potential impact of the seismic hazard on the categorization of RISC-3 SSCs is maintained acceptably low and the requirements in 10 CFR 50.69(c)(1)(iv) are met for the proposed alternative seismic approach.

High Winds:

The licensee proposed to use a conservative approach to consider high winds hazards in lieu of the licensee's current high winds PRA, which is pending upgrade. The licensee provided a description of its proposed conservative approach for high winds in section 3.2.4, "High Winds Hazards," and section 3.3.3, "High Winds PRA Model," of enclosure I to its LAR. The licensee stated in section 3.3 of enclosure I to its LAR that its high winds PRA contains sufficient information and risk insights to support a qualitative, but conservative, approach to assess component risk significance due to high winds. In section 3.2.4, the licensee stated that a formal high wind fragility analysis has been performed, which has evaluated plant equipment and provided a component level screening for identifying those components susceptible to high wind hazards and those that are not. The licensee also stated that the high winds PRA is sufficiently developed to identify components that will be included in its scope.

In the proposed approach, a component is categorized high wind HSS if a component is not screened "high wind impact screened" with justification that the component is within a seismic category 1 structure per the high wind fragility analysis and is included in the high wind plant response model and quantification. The licensee explained that this is a conservative approach since the quantified PRA results might indicate certain components in the high winds PRA scope could have had importance measures meeting LSS criteria, but the proposed approach would categorize them as HSS. The other components would be categorized as LSS.

The NRC staff reviewed the licensee's evaluation provided in section 3.3 of enclosure I to the LAR and finds that the licensee's qualitative approach to establishing a high wind qualitative component level active risk significance acceptable because the licensee has identified the components that will be in the scope of the high winds PRA and the proposed approach would classify more components as HSS than using the importance measures from the high winds PRA.

External Hazards and Other Hazards (Non-Seismic)

In section 3.2 of enclosure I to its LAR, the licensee explained that it had evaluated other external hazards using the criteria in the 2009 ASME/ANS PRA standard (Reference 12). In table 4-4, "Hazard Dispositions," of attachment 4, "External Hazards Screening," to enclosure I to its LAR, the licensee provided the evaluation results and stated that all other external hazards that could potentially affect the Wolf Creek site were screened out based on the criteria specified in supporting requirements EXT-B1 or EXT-C1 of the 2009 ASME/ANS PRA standard.

The NRC staff's independent review finds that the licensee's categorization process will evaluate the safety significance of SSCs for other external hazards consistent with the guidance provided in ASME/ANS RA-Sa–2009 PRA standard. The NRC staff finds that the licensee's treatment of other external hazards is acceptable and meets the requirements of 10 CFR 50.69(c)(1)(ii).

Shutdown Risk

Consistent with the guidance in NEI 00-04, Revision 0, the licensee proposed using the shutdown safety assessment based on NUMARC 91-06 (References 9 and 14, respectively). NUMARC 91-06 provides considerations for maintaining defense in depth for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment-primary/secondary. NUMARC 91-06 also specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

The use of NUMARC 91-06, described by the licensee in the submittal, is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1. The approach uses an integrated and systematic process to identify HSS components, consistent with the shutdown evaluation process. Therefore, the NRC staff finds that the licensee's use of NUMARC 91-06 is acceptable, and meets the requirements set forth in 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 the supplement to the LAR dated February 3, 2026, the licensee proposed utilizing the categorization method described in EPRI TR 3002033536, for passive components (Reference 13). EPRI TR 3002033536 was published as Revision 1-A after the NRC staff had issued the safety evaluation for EPRI TR 3002025288 (Reference 37).

The licensee stated that it will follow the five phases presented in section 4 of EPRI Technical Report 3002033536 to perform the enhanced passive categorization. These phases consist of:

Phase 1: Prerequisites
Phase 2: Predetermined HSS Passive SSCs
Phase 3: Design and plant specific search for HSS passive SSCs
Phase 4: Sensitivity study and IDP review and concurrence
Phase 5: Performance monitoring

As EPRI TR 3002033536 was reviewed by the NRC staff, this review of the LAR is limited to determining if the licensee meets the prerequisites described in Phase 1 of the EPRI TR 3002033536. This review did not include the implementation of EPRI TR 3002033536 Phases 2-5 as Phases 2-5 are subject to inspection at a later date.

Phase 1 of the EPRI TR 3002033536 categorization methodology for pressure boundary components requires a set of four prerequisites and a predetermined set of HSS systems/subsystems coupled with a plant specific review for potentially risk-significant or high consequence outliers. In order to use EPRI TR 3002033536, the licensee must meet these prerequisites:

- Prerequisite 1 addresses PRA Technical Adequacy in which the licensee must have a robust internal events PRA, including internal flooding, that addresses failure of all pressure boundary components.

- Prerequisite 2 Integrity Management indicates that applicants shall identify the programs for addressing pressure boundary integrity management, specifically, localized corrosion, flow-accelerated corrosion, and erosion, and provide justification to demonstrate the robustness of those programs.

- Prerequisite 3 addresses protective measures for internal flooding events. The prerequisite states that protective measures for internal flooding should not be categorized as LSS unless additional evaluations have been conducted to show that loss of these measures or a subset of them will not invalidate the HSS determination.

- Prerequisite 4 ensures the PRA reflects the as-built/as-operated plant by including a configuration control program, a feedback and adjustment process, and meeting periodicity requirements for updating the PRA.

Evaluation of Prerequisites:

In section 3.3 of enclosure I to its LAR, the licensee states that the Wolf Creek PRA was assessed against the requirements of RG 1.200. F&O closure reviews were conducted and all findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, confirming that the licensee's PRA models are of sufficient quality and level of detail to support the categorization process. Therefore, the NRC staff finds that the licensee meets Prerequisite 1.

Prerequisite 2 addresses Integrity Management and requires the licensee to have robust programs to address Localized Corrosion, Flow Accelerated Corrosion (FAC), and Erosion. The NRC staff conducted an audit of the FAC programs as part of the 2008 Safety Evaluation for the Wolf Creek license renewal (Reference 38) and found it to be consistent with the Generic Aging Lessons Learned (GALL) Aging Management Program (AMP) XI.M17, but not specifically to the prerequisites as described in the EPRI technical report. The NRC staff performed an audit of the Integrity Management programs cited for Prerequisite 2, as part of this submittal review, and confirmed that the Localized Corrosion, FAC, and Erosion programs at Wolf Creek are consistent with the prerequisites. The licensee supplemented the submittal by letter dated June 25, 2026 (Reference 3), affirming that the Corrosion, FAC, and Erosion programs meet the prerequisite requirements of the EPRI technical report and are consequently appropriately robust as described in the EPRI technical report. Therefore, the NRC staff finds that the licensee meets Prerequisite 2.

To meet Prerequisite 3, the licensee indicated, in attachment 2 of its LAR, that internal flooding components will be categorized according to the licensee's Internal Flood Model of Record 09 (IFPRA MOR 09). This PRA was independently peer-reviewed using the process outlined in RG 1.200, and all finding-level F&Os were closed. Therefore, an internal flooding component will not be categorized as LSS unless that component undergoes additional evaluation by the process described in IFPRA MOR 09. Therefore, the NRC staff finds that the licensee meets Prerequisite 3.

In section 3.2.7, “PRA Maintenance and Updates,” of enclosure I to its LAR, the licensee described the process for ensuring that the applicable PRAs continue to reflect the as-built as-operated plant. Potential areas affecting PRA models will be monitored, and the impact of plant changes will be assessed at least once every two refueling outages. The feedback and adjustment process is described in section 3.5 of enclosure I to the LAR, which the NRC staff finds acceptable. The licensee also states that a complete periodic review is scheduled at least once every two complete refueling cycles. Therefore, the NRC staff finds that Prerequisite 4 is met.

The NRC staff noted that the licensee has met all four prerequisites and did not propose any deviation from the EPRI Technical Report 3002033536 (Rev. 1-A). Furthermore, as discussed in the NRC staff safety evaluation dated July 15, 2025, of EPRI TR 3002025288 (Reference 37), Criterion 1 of the EPRI technical report indicated that all ASME Code Class 1 SSCs are described as HSS. The EPRI technical report then provides some guidance regarding which Class 1 SSCs may be eligible to be reclassified as something other than Class 1 outside of the 10 CFR 50.69 process.

The NRC staff also noted that while EPRI TR 3002025288 is not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1 for passive component categorization, it was approved by the NRC staff in its safety evaluation dated July 15, 2025. In the NRC staff safety evaluation of EPRI TR 3002025288, the NRC staff concludes that categorization of passive, pressure-retaining components in accordance with the methodology in chapter 4 of the approved version of EPRI TR 3002025288 is acceptable for use as part of a licensee’s passive component categorization process. Therefore, the NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process for Wolf Creek.

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

Section 3.1.1 of enclosure I to its LAR, the licensee states that an unreliability factor of three will be used for the sensitivity studies described in section 8, “Risk Sensitivity Study,” of NEI 00-04, Revision 0. Section 3.2.8, “PRA Uncertainty Evaluations,” of enclosure I to its LAR, the licensee 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 three. The NRC staff finds the application of a factor of three for the sensitivities consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

In section 3.1.1 of enclosure I to its LAR, the licensee specifically noted that RG 1.201 states “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 (Reference 8) must be followed to achieve reasonable confidence in the evaluations required by [10 CFR] 50.69(c)(1)(iv).” This sensitivity study together with the periodic review process discussed in section 3.3.4 of this safety evaluation, assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors 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).

In light of the above the NRC staff review for: (1) internal events PRA (including internal flooding) and fire PRA acceptability, (2) PRA importance measures and integrated importance measure, (3) evaluation of the use of non-PRA methods, (4) risk sensitivity study, and (4) integrated decision making, the 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 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 updates to the PRA over time are continually incorporated.

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

Sections 11 and 12 of NEI 00-04, Revision 0, includes 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 Wolf Creek as-built, as-operated plant.

The NRC staff recognizes that for facilities licensed under 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criterion VI, for Document Control, procedures are considered formal plant documents that require "[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 3.2.7 of enclosure I to the LAR, for the Wolf Creek 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. Therefore, 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 LAR and the licensee's responses to the NRC staff's RAIs, the NRC staff identified specific actions, as described below that are necessary to support the NRC staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69 and 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 provided below. 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 3.3.4 of this SE.

The licensee proposed the following amendment to the RFOLs for Wolf Creek. The proposed license condition states:

> WCNOC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC) 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 Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components; the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic and high winds; and the alternative seismic and high winds approaches described in WCNOC's submittal letter 000779 dated January 30, 2025, as supplemented by letters 001182 and 001656 dated February 3, 2026, and June 25, 2026, respectively, as specified in License Amendment No. 248 dated August 13, 2026.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

The NRC staff finds that the proposed license condition and referenced implementation items are acceptable because they adequately implement 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC. The NRC staff, through an onsite audit or during future inspections, may choose to examine the closure of the implementation items with the expectation that any variations discovered during this review, or concerns with regard to adequate completion of the implementation item, will be tracked and dispositioned appropriately in accordance with the requirements of 10 CFR 50.69(f) and 10 CFR Part 50, Appendix B, Criterion VI, and could be subject to NRC enforcement action(s).

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment on June 23, 2025. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

This action relates to authorizations under 10 CFR Part 50 with respect to installation or use of a facility component. The NRC staff has determined that any ground disturbance is limited to previously disturbed areas. Additionally, the NRC staff has determined that the action involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, no significant increase in individual or cumulative public or occupational radiation exposure, and no significant increase in the potential for or consequences from radiological accidents. Finally, the NRC staff has determined that a categorical exclusion applies and that special circumstances under 10 CFR 51.22, "Categorical exclusions," are not present that would preclude reliance on the categorical exclusion. Accordingly, this action meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(d)(8). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment need be prepared for this action in connection with the action.

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

8.0 REFERENCES

1. Boyce, M. T., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: License Amendment Request to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," dated January 30, 2025 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML25030A384).

2. Bayer, R. J., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Supplement to License Amendment Request to Adopt 10 CFR 50.69, dated February 3, 2026 (ML26034C129).

3. Bayer, R. J., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Supplement to License Amendment Request to Adopt 10 CFR 50.69, Response to Audit Question NPHP-01, dated June 25, 2026 (ML26177A269).

4. 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, May 2006 (ML061090627).

5. U.S. Nuclear Regulatory Commission, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 2, March 2009 (ML090410014).

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

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20. Lee, S. S., U.S. Nuclear Regulatory Commission, letter to C. Reasoner, Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Unit 1 – Audit Summary in Support of 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-2025-LLA-0017),” dated June 4, 2026 (ML26064A092).

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Date: August 13, 2026

SUBJECT: WOLF CREEK GENERATING STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 248 RE: ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2025-LLA-0017) DATED AUGUST 13, 2026

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