Michael T. Boyce
Vice President Engineering

January 30, 2025
000779

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

Subject: 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"

Commissioners and Staff:

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Wolf Creek Nuclear Operating Corporation (WCNOC) is submitting a request for an amendment to the renewed facility operating license (FOL), NPF-42, for the Wolf Creek Generating Station (WCGS). The proposed amendment would modify the renewed FOL to allow for the implementation of the provisions of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors."

Enclosure I provides the evaluation and justification for the proposed license amendment. Additionally, Enclosure I contains seven attachments that discuss the following in support of the license amendment:

- Categorization prerequisites, PRA models used in categorization, disposition and resolution of open peer review findings and self-assessment open items, external hazards screening, progressive screening approach for addressing external hazards, disposition of key assumptions/sources of uncertainty, and the markup of the FOL showing the proposed change.

WCNOC requests approval of the proposed license amendment within one year from the date of this submittal with implementation within 90 days following NRC approval. Additionally, WCNOC requests that the NRC audit associated with this submittal be combined with the NRC audit planned for the TSTF-505 submittal (ML24352A438).

WCNOC has determined that this amendment application does not involve a significant hazards consideration as determined per 10 CFR 50.92, "Issuance of amendment." The amendment application was reviewed by the WCNOC Plant Safety Review Committee. In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," section (b)(1), a copy of this application, with attachments and enclosures, is being provided to the designated Kansas State official.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-8831 x8687, or Dustin T. Hamman at (620) 364-4204.

Sincerely,

Michael T. Boyce

MTB/jkt

Enclosure I – Evaluation of Proposed Change

cc: A. N. Agrawal (NRC), w/a, w/e
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WCNOC Licensing Correspondence ET 25-000779, w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Michael T. Boyce, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By____________________
Michael T. Boyce
Vice President Engineering

SUBSCRIBED and sworn to before me this 30 day of January , 2025.

JENA MICHELLE STUCKER
Notary Public - State of Kansas
My Appointment Expires 1-11-2027

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Notary Public

Expiration Date 1-11-2027

EVALUATION OF PROPOSED CHANGE

Subject: Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

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1.0 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.” The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

2.0 DETAILED DESCRIPTION

2.1 Current Regulatory Requirements

The Nuclear Regulatory Commission (NRC) has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a “deterministic” approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. The structures, systems and components (SSCs) necessary to defend against DBEs are defined as “safety-related,” and these SSCs are the subject of certain regulatory requirements, herein referred to as “special treatments,” designed to ensure that they are of high quality and high reliability and have the capability to perform during postulated design basis conditions. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between “treatment” and “special treatment” is the degree of NRC specification as to what must be implemented for particular SSCs or for particular conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: “safety-related,” “important to safety,” or “basic component.” The terms “safety-related” and “basic component” are defined in the regulations, while “important to safety,” used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

2.2 Reason for the Proposed Change

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, probabilistic risk assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner.

To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of LSS, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of HSS, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The rule contains requirements on 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 risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by Nuclear Energy Institute (NEI) 00-04, “10 CFR 50.69 SSC Categorization Guideline” [1], which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to make adjustments to the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule 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, the rule 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 enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow Wolf Creek Nuclear Operating Corporation (WCNOC) to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 Description of the Proposed Change

WCNOC proposes the addition of the following condition to Appendix D of the renewed facility operating license (FOL), NPF-42, for the Wolf Creek Generating Station (WCGS) to document the NRC's approval of the use of 10 CFR 50.69:

> *WCNOC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC) RISC-1, RISC-2, RISC3, and RISC-4 structures, systems, and components (SSCs) using: probabilistic risk assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2, Class 3, and non-class SSCs and their associated supports; 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 ET 25-000779 dated January 30, 2025; as specified in License Amendment No. [XXX] dated [DATE].*

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

3.0 TECHNICAL EVALUATION

10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation. This request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:

(i) A description of the process for categorization of RISC-1, RISC-2, RISC-3 and RISC-4 SSCs. (See Section 3.1).

(ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs. (See Section 3.2).

(iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i). (See Section 3.3).

(iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions). (See Section 3.4).

Each of these submittal requirements are addressed in the following sections.

Though routine maintenance updates have been applied, the NRC has previously reviewed and accepted the technical adequacy of the WCGS internal events PRA model identified in this application for Amendment No. 227 for adopting Technical Specification Task Force (TSTF)-425, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b" [25] and the "Seismic Hazard and Screening Report for Wolf Creek" [55]. The risk analyses described in this application utilize PRA models (model versions may vary) that have been described and used in the other applications that required NRC review and approval. Therefore, WCNOC requests that the NRC utilize the review of the PRA technical adequacy for those applications when performing the review for compliance with 10 CFR 50.69(b)(2)(ii) and 10 CFR 50.69(b)(2)(iii) in this application.

3.1 Categorization Process Description (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

WCNOC will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" [2]. NEI 00-04 Section 1.5 states "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04, as endorsed by RG 1.201 [2], with the following exceptions:

- the evaluation of the impact of seismic risks, which will use the Electric Power Research Institute (EPRI) 3002017583 [30] approach for seismic Tier 2 sites; and
- the evaluation of the impact of high winds risks, which will use a conservative and bounding screening approach to assess active component risks due to high winds.

The process includes additional steps to address seismic and high winds considerations, which satisfies the requirements in § 50.69(c)(1)(iv). RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by § 50.69(c)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. The order in which the categorization process steps are completed is flexible, and may be performed in parallel as long as all components are completed. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as LSS by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety-related active components/functions categorized as LSS by all other elements.

1. PRA-based assessments (e.g., the internal events, internal flooding, and fire PRAs)
2. Non-PRA approaches (e.g., other external events screening and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04
4. The defense-in-depth (DID) assessment
5. The passive categorization methodology

Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., HSS or LSS) that is presented to the Integrated Decision-Making Panel (IDP)). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1 below. The safety significance determination of each element, identified above, is independent of each other and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final RISC category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04, Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS, however the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201.

Table 3-1 summarizes these IDP limitations in NEI 00-04. The steps of the process are performed at either the function level, component level, or both, and is summarized in the Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

Table 3-1: Categorization Evaluation Summary

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case - Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Hazards Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment - Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire and Other External Hazards -	Component	Not Allowed	No
	Seismic – Alternative Tier 2 Approach	Function/Component	Allowed[1]	No
	High Winds – Alternative Screening Approach	Component	Not Allowed	No
	Shutdown - Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage - Section 6.1	Function/Component	Not Allowed	Yes
	Containment - Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[2]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

Notes:

[1] IDP consideration of seismic insights can also result in an LSS to HSS determination.

[2] The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration, however the final assessments of the seven considerations are the direct responsibility of the IDP.

The seven considerations are addressed preliminarily by the 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step (e.g., PRA). Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that system function consideration. If the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated *justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.*

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal Events PRA or Integral PRA Assessment) or DID evaluation will be initially treated as HSS. However, NEI 00-04, Section 10.2, allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with an HSS function, but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g., passive, non-PRA-modeled hazards - see Table 3-1 above). Except for seismic, these components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Components having seismic functions may be HSS or LSS based on the IDP's consideration of the seismic insights applicable to the system being categorized. Therefore, if an HSS component is mapped to an LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS, based on Table 3-1 above, or may remain LSS. For the seismic hazard, given that WCGS is a seismic Tier 2 (moderate seismic hazard) plant as defined in Reference [30], seismic considerations are not required to drive an HSS determination at the component level, but the IDP will consider available seismic information pertinent to the components being categorized and can, at its discretion, determine that a component should be HSS based on that information.

The following are clarifications to be applied to the NEI 00-04 categorization process:

- The IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design engineering, system engineering, safety analysis, and PRA. At least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in the modeling and updating of the plant-specific PRA.

- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant-specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the DID philosophy and requirements to maintain this philosophy.

- The decision criteria for the IDP for categorizing SSCs as safety significant or LSS pursuant to § 50.69(f)(1) will be documented in WCNOC procedures.

- Decisions of the IDP will be arrived at by consensus.

- Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding HSS and LSS. If the majority of IDP members do not agree on an LSS designation, then the item or component would default to HSS.

- Passive characterization will be performed using the processes described in Section 3.1.2 of Enclosure I. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

- An unreliability factor of three (3) will be used for the sensitivity studies described in Section 8 of NEI 00-04 for LSS components. The factor of three (3) was chosen as it is representative of the typical error factor of basic events used in the PRA model.

- NEI 00-04, Section 7, requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5, but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle Safety Evaluation [4] which states "...if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the DID assessment (Section 6), the associated system function(s) would be identified as HSS."

- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS Function components to LSS.

- With regard to the criteria that considers whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, WCNOC will not take credit for alternate means unless the alternate means are proceduralized and included in licensed operator training.

- WCNOC proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3, as discussed in Section 3.2.3 of this LAR.

- WCNOC proposes to apply an alternative screening approach to assess risks from high winds, as discussed in Section 3.2.4 of this LAR.

The risk analysis to be implemented for each modeled hazard is described below.

Note: The previous version of the internal events PRA models were determined to be acceptable by the NRC for the listed applications (e.g., Amendment No. 227 for adopting TSTF-425 [25]). At that time, the internal flood and fire PRAs were under development but have since been completed and peer reviewed with no outstanding peer review issues. Additionally, no PRA upgrades have been incorporated into the internal events, internal flood, or fire PRA models for this risk informed application which would have required a focused peer review consistent with RG 1.200 Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," as noted below.

- Internal Event Risks: WCNOC Internal Events PRA Model-of-Record (MOR) 10. It is noted that MOR 10 closed open items and incorporated routine maintenance on WCNOC 09. The WCNOC MOR 09 was previously accepted by NRC for License Amendment No. 227 [25].

- Internal Flooding Risks: WCNOC Internal Flooding PRA – Based on Internal Event PRA MOR 09.

- Fire Risks: WCNOC Fire PRA – Based on Internal Event PRA MOR 10.

- Seismic Risks: WCNOC will use the Alternative Approach in EPRI 3002017583 [30] for Tier 2 plants with the additional considerations discussed in Section 3.2.3 of this LAR.

- High Winds Risks: The WCNOC High Winds PRA model has not been completed and requires a focused scope peer review for an upgrade following the completion of an industry Newly Developed Method peer review. In lieu of the High Winds PRA, a conservative and bounding screening approach will be used to assess active component risk due to high winds and is further described in Section 3.2.

- Other External Risks (e.g., external floods): External Hazards are discussed in Section 3.2 and dispositioned in Attachment 4 of this enclosure. The other external hazards shown in Attachment 4 were determined to be insignificant contributors to plant risk and were screened in accordance with PRA Standard requirements.

- Low Power and Shutdown Risks: Qualitative DID shutdown model for shutdown configuration risk management (CRM) based on the framework for DID as provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" [3], which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from the Tier 2 alternate seismic approach to a seismic PRA approach) will not be used without prior NRC approval. The SSC categorization process documentation will include the following elements:

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

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology approved by the NRC in the ANO Unit 2 Approval of Request for Alternative ANO2-R&R-004, Revision 1, [5], consistent with the related Safety Evaluation (SE) issued by the Office of Nuclear Reactor Regulation.

The RI-RRA methodology is a risk-informed safety categorization and treatment program for repair/replacement activities for pressure retaining items and their associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., DID, safety margins) in determining safety significance. Consistent with the RI-RRA methodology approved by the NRC in the ANO Unit 2 Approval of Request for Alternative ANO 02-R&R-004 [5], pipe supports were not required to be in the scope of the evaluation but may be included in the scope at the licensee's discretion. Component supports, if categorized, are assigned a safety significance based upon one of the following approaches:

- Supports should have the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

- A combination of restraints or supports such that the LSS piping and associated SSCs attached to the HSS piping are included in the scope up to a boundary point that encompasses at least two supports in each of three orthogonal directions.

The RI-RRA methodology was also approved to be used for a 10 CFR 50.69 application by the NRC in License Amendment Nos. 173 and 155 to the Vogtle Electric Generating Plant, Units 1 and 2 [4]. The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant-specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment.

The passive categorization process is intended to apply the same risk-informed process approved by the NRC in the ANO Unit 2 Approval of Request for Alternative ANO2-R&R-004 [5] for the passive categorization of Class 2, 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in RG 1.147, Revision 15, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1. Both code cases employ a similar risk-informed safety categorization of SSCs in order to change the repair/ replacement requirements of the affected LSS components. All categorized ASME Code Class 1 SSCs with a pressure retaining function, as well as categorized supports, will be assigned HSS, for passive categorization which will result in HSS for its risk-informed safety categorization and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at WCGS for 10 CFR 50.69 SSC categorization.

3.2 Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii))

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs are adequate. The PRA models described below have been independently peer reviewed, and there are no PRA upgrades that have not been peer reviewed.

3.2.1 Internal Events and Internal Flooding

The WCNOC categorization process for the internal events and internal flooding hazards will use a technically acceptable and independently peer reviewed plant-specific PRA model consistent with RG 1.200 [7] and the 2009 ASME/ANS PRA Standard [10]. The WCNOC risk management process ensures that the PRA model used in this application reflects the as-built as-operated plant for WCGS, Unit 1. Attachment 2 of this enclosure identifies the applicable Internal Events (including internal flooding) PRA models.

3.2.2 Fire Hazards

The WCNOC categorization process for fire hazards will use a technically acceptable and independently peer reviewed plant-specific fire PRA model. The internal Fire PRA model was developed consistent with NUREG/CR 6850 [15], RG 1.200 [7] and the 2009 ASME/ANS PRA Standard [10]. The WCNOC risk management process ensures that the PRA model used in this application reflects the as-built, as-operated plant for WCGS, Unit 1. Attachment 2 of this enclosure identifies the applicable internal Fire PRA model.

3.2.3 Seismic Hazards

10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For other risk hazards, such as seismic, 10 CFR 50.69(b)(2) allows, and NEI 00-04 [1] summarizes, the use of other methods for determining SSC functional importance in the absence of a quantifiable PRA (such as SMA or IPEEE Screening) as part of an integrated, systematic process. For the WCGS seismic hazard assessment, WCNOC proposes to use a risk-informed graded approach that meets the requirements of 10 CFR 50.69(b)(2) as an alternative to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002017583 with the markups provided in Attachment 2 of References [33] and [34] and includes additional considerations that are discussed in this section.

> Note: The discussion below pertaining to EPRI 3002017583 includes the markups provided in Attachment 2 of References [33] and [34].
>
> EPRI 3002017583 [30] is an update to EPRI 3002012988, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,” July 2018 [36] which was referenced in the NRC-issued amendment and Safety Evaluation for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, to implement 10 CFR 50.69, as noted in Reference [37]. The technical criteria in EPRI 3002017583 are unchanged from its predecessor report EPRI 3002012988.
>
> This license amendment request (LAR) incorporates by reference the Clinton Power Station, Unit 1 response to request for additional information (RAI) ‘DRA/APLC RAI 03 – Alternate Seismic Approach’ included in the letter dated November 24, 2020 [38], in particular, the response to the question regarding the differences between the initial EPRI report (3002012988) and EPRI 3002017583.

The proposed categorization approach for WCGS is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA (SPRA). This approach relies on the insights gained from the SPRAs examined in EPRI 3002017583 and plant specific insights considering seismic correlation effects and seismic interactions.

Following the criteria in EPRI 3002017583, WCGS is considered a Tier 2 site because the site ground motion response spectrum (GMRS) to safe shutdown earthquake (SSE) comparison is above the Tier 1 threshold, but not high enough that the NRC required the plant to perform an SPRA to respond to Recommendation 2.1 of the Near-Term Task Force 50.54(f) letter [39]. EPRI 3002017583 also demonstrates that seismic risk is adequately addressed for Tier 2 sites by the results of additional qualitative assessments discussed in this section and existing elements of the 10 CFR 50.69 categorization process specified in NEI 00-04.

The trial studies in EPRI 3002017583, as amended by the RAI responses and NRC issued amendments [40], [41], [42], [43], [44], [45], [46], [47], [48] demonstrate that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 2 classification and resulting criteria is that consideration of the full range of the seismic hazard produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of EPRI 3002017583.

> At Tier 2 sites, there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. The special seismic risk evaluation process recommended using a Common Cause impact approach in the [full power internal events] FPIE PRA can identify the appropriate seismic insights to be considered with the other categorization insights by the [IDP] for the final HSS determinations.

At sites with moderate seismic demands (i.e., Tier 2 range) such as WCGS, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as EPRI NP-6041-SL [49]. Tier 2 seismic demand sites have a lower likelihood of seismically induced failures and less challenges to plant systems than trial study plants. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazards at WCGS.

Test cases described in Section 3 of EPRI 3002017583, as amended by their RAI responses and NRC issued amendments [40], [41], [42], [43], [44], [45], [46], [47], [48] demonstrated that there are very few, if any, SSCs that would be designated HSS for seismic unique reasons. The test cases identified that the unique seismic insights were typically associated with seismically correlated failures and led to unique HSS SSCs. While it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, it is prudent and recommended by EPRI 3002017583 to perform additional evaluations to identify the conditions where correlated failures and seismic interactions may occur and determine their impact in the 10 CFR 50.69 categorization process. The special sensitivity study recommended in EPRI 3002017583 uses common cause failures, similar to the approach taken in a full power internal events (FPIE) PRA and can identify the appropriate seismic insights to be considered by the IDP along with the other categorization insights for the final HSS determinations.

The test case information from EPRI 3002017583, developed by other licensees, including Case Study A [50], Case Study C [51], and Case Study D [52], as well as RAI responses and amendments [40], [41], [42], [43], [44], [45], [46], [47], [48] clarify aspects of these case studies and provide additional supporting bases for this application. Therefore, these case studies, RAI responses, and amendments are incorporated by reference into this amendment request.

As defined in EPRI 3002017583, WCGS meets the Tier 2 criteria for a “Moderate Seismic Hazard / Moderate Seismic Margin” site. The Tier 2 criteria are as follows:

> 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.

Note: EPRI 3002017583 applies to the Tier 2 sites in its entirety except for Sections 2.2 (Tier 1 sites) and 2.4 (Tier 3 sites).

For comparison, Tier 1 plants are defined as having a GMRS peak acceleration at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz and 10 Hz. Tier 3 plants are defined where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform an SPRA to respond to the Fukushima 10 CFR 50.54(f) letter [39].

The NRC issued its final determination of licensee SPRAs in a letter dated October 27, 2015 [53]. The letter informed power reactor licensees of the remaining seismic evaluations to be performed and specifically informed those licensees that would perform an SPRA. In the letter, NRC stated:

> If the seismic hazard exceedance, peak of the spectral acceleration, and the general estimation of the [seismic core damage frequency] SCDF were judged to be not significant, then the NRC staff concluded that a SPRA is not necessary for NRC's 50.54(f) letter-related regulatory decisions. Based on this additional assessment, the NRC staff has determined that SPRA are not warranted for 13 sites listed in Table 1a in Enclosure 1.

Note 3 of Table 1a identifies WCGS as a site where an SPRA or SMA were no longer expected.

As shown in Figure 1 (Figure 3.4-1 of Reference [55]), comparing the WCGS GMRS (derived from the seismic hazard) to the SSE (seismic design basis capability), the GMRS exceeds the SSE in the higher portion of the range between 1.0 Hz and 10 Hz. Note that Figure 1 also shows the GMRS as determined by the NRC, when the staff reviewed information related to the reevaluated seismic hazard for WCGS [54] and [55]. WCNOC's GMRS curve for WCGS ("Licensee GMRS" in Figure 1) is very similar to that determined by the NRC staff ("NRC GMRS" in Figure 1) and both curves exceed the SSE (both the Powerblock SSE and the non-Powerblock SSE) in the upper portion of the response spectrum between 1.0 and 10 Hz.

Therefore, the WCGS seismic hazard meets the criteria for Tier 2 from EPRI 3002017583. The basis for WCGS being classified as a Tier 2 site will be documented and presented to the IDP for each system that is categorized.

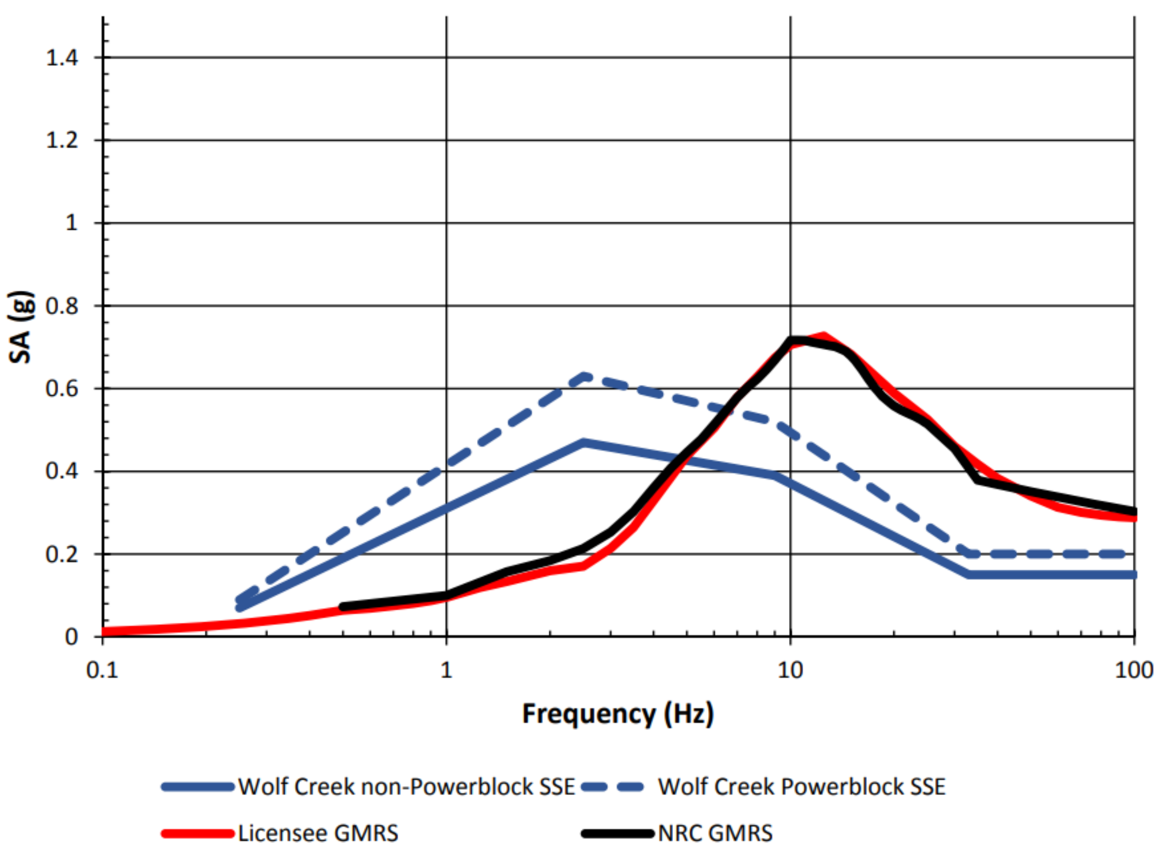

Figure 1: Comparison of the WCGS GMRS to the SSE

The following paragraphs describe additional background and the process to be utilized for the graded approach to categorize the seismic hazard for a Tier 2 plant.

Implementation of the Recommended Process

EPRI 3002017583 recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation Tiers in the report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in EPRI 3002017583 for identifying unique seismic insights.

The seismic fragility of an SSC is a function of the margin between an SSC's seismic capacity and the site-specific seismic demand. References such as EPRI NP-6041 [49] provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For example, a pump has a relatively high inherent seismic capacity based on its design and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand.

There are some plant features such as equipment anchorage that have seismic capacities more closely associated with the site-specific seismic demand since those specific features are specifically designed to meet that demand. However, even for these features, the design basis criteria have intended conservatisms that result in significant seismic margins within SSCs. These conservatisms are reflected in key aspects of the seismic design process. The SSCs used in nuclear power plants are intentionally designed using conservative methods and criteria to ensure that they have margins well above the required design bases.

In applying the EPRI 3002017583 process for Tier 2 sites to the WCGS 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the guidance and informed of plant SSC-specific seismic insights that the IDP may choose to consider in their HSS/LSS deliberations. As part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included that provides identified plant seismic insights as well as the basis for applicability of the EPRI 3002017583 study and the bases for WCGS being a Tier 2 plant. The discussion of the Tier 2 bases will include such factors as:

- The moderate seismic hazard for the plant,
- The definition of Tier 2 in the EPRI study, and
- The basis for concluding WCGS is a Tier 2 plant.

At several steps of the categorization process the categorization team will consider the available seismic insights relative to the system being categorized and document their conclusions in the SCD. Integrated importance measures over all the modeled hazards (i.e., internal events, internal flooding, and internal fire for WCGS) are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the specified criteria are preliminary HSS which cannot be changed to LSS. For HSS SSCs uniquely identified by the WCGS PRA models but having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, these will be addressed using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA.

For components that are HSS due to fire PRA but not HSS due to internal events PRA, 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 for presentation to the IDP as additional qualitative inputs, which will also be described in the SCD.

The categorization team will review available WCGS plant-specific seismic insights and other resources such as those identified above. The objective of the seismic review 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)

For each system categorized, the categorization team will evaluate correlated seismic failures and seismic interactions between SSCs. This process is detailed in Section 2.3.1 of EPRI 3002017583, including the markups provided in Attachment 2 of References [33] and [34], and as described in this request. The process is summarized in Figure 2.

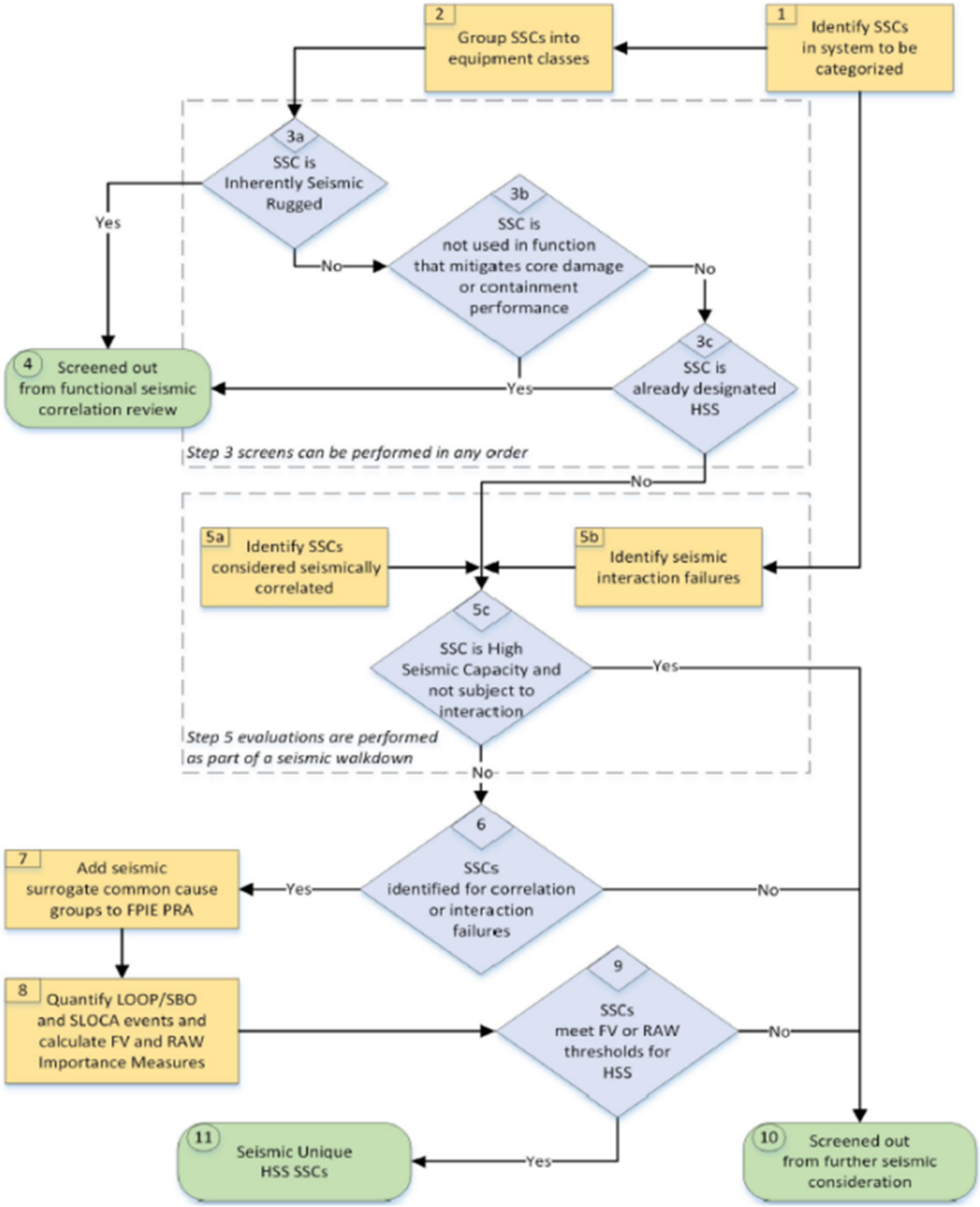

Figure 2: Seismic Correlated Failure Assessment for Tier 2 Plants

Determination of seismic insights will make use of the FPIE PRA model supplemented by focused seismic walkdowns. An overview of the process to determine the importance of SSCs for mitigating seismic events follows and is utilized on a system basis:

- Gather the population of SSCs in the system being categorized and review existing seismic information (Step 1 of Figure 2). This step may use the results of the required Tier 1 assessment that is performed along with the Tier 2 assessment. As stated in EPRI 3002017583, the technical basis for the Tier 1 approach generally applies for Tier 2 plants in addition to the additional sensitivity and walkdowns described herein.

- Assign seismic-based SSC equipment class and distributed system IDs, as used for SPRAs, for SSCs in the system being categorized (Step 2 of Figure 2).
- Perform a series of screenings to refine the list of SSCs subject to correlation sensitivity studies. Screens will identify (Steps 3a/3b/3c of Figure 2):
    - Inherently rugged SSCs
    - SSCs not in Level 1 or Level 2 PRAs
    - Components already identified as HSS components from the internal events PRA or integrated assessment
    - The above screened SSCs will still be evaluated for seismic interactions (Step 1 to Step 5b in Figure 2).
- SSCs identified in the above screening can be screened from consideration as functional correlation surrogate events. They are removed from the remainder of the process (can be considered LSS) unless they are subject to interaction source considerations (Step 4 of Figure 2).
- Perform Tier 2 walkdowns focusing on identifying seismic correlated or interaction SSC failures for SSCs that were not previously walked down (Steps 5a/5b of Figure 2).
- Screen out from further seismic considerations SSCs that are determined through the walkdown to be of high seismic capacity and not included in seismically correlated groups or correlated interaction groups since their non-seismic failure modes are already addressed for 50.69 categorization in the FPIE PRA and fire PRA. Those remaining components proceed forward for inclusion of associated seismic surrogate events in the Tier 2 Adjusted PRA Model (Steps 5c/6 of Figure 2).
- Develop a Tier 2 Adjusted PRA Model and incorporate seismic surrogate events into the model to reflect the potential seismically correlated and interaction conditions identified in prior steps (Steps 6/7 of Figure 2). The seismic surrogate basic events shall be added to the PRA under the appropriate areas in the logic model (e.g., given that the Tier 2 Adjusted PRA Model uses only loss of offsite power (LOOP) and small loss of coolant accident (LOCA) sequences, the seismic surrogate events should be added to system and/or nodal fault tree structures that tie into these sequence types). The probability of each seismic surrogate basic event added to the model should be set to 1.0E-04 (based on guidance in EPRI 3002017583).
- Quantify only the LOOP and small LOCA initiated accident sequences of the Tier 2 Adjusted PRA Model (Step 8 of Figure 2). The event frequency of the LOOP initiator shall be set to a value of 1.0 and the event frequency for the small LOCA initiator shall be set to a value of 1.0E-02. Remove credits for restoration of offsite power and other functional recoveries (e.g., Emergency Diesel Generator and DC power recovery).
- Utilize the importance measures from the quantification of the Tier 2 Adjusted PRA Model to identify appropriate SSCs (in the system being categorized) that should be HSS due to correlation or seismic interactions (Step 9 of Figure 2).
- SSCs screened out in Steps 5c, 6, or 9 in Figure 2 can be considered LSS (Step 10 of Figure 2).
- Prepare documentation of the Tier 2 analysis results, including identification of seismic unique HSS SSCs, for presentation to the IDP (Step 11 of Figure 2).

Seismic impacts would be compiled on an SSC basis. As each system is categorized, the system-specific seismic insights will be documented in the categorization report and provided to the IDP for consideration as part of the IDP review process. The IDP cannot challenge any candidate HSS recommendation for any SSC from a seismic perspective if they believe there is a basis, except for certain conditions identified in Step 10 of Section 2.3.1 of EPRI 3002017583 (including markups in References [33] and [34]). Any decision by the IDP to downgrade preliminary HSS components to LSS will consider the applicable seismic insights in that decision. SSCs identified from the fire PRA as candidate HSS, which are not HSS from the internal events PRA or integrated importance measure assessment, will be reviewed for their design basis function during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events. These insights will provide the IDP with a means to consider potential impacts of seismic events in the categorization process.

If the WCGS 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. Upon receipt of NRC approval for such a change, WCNOC 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 3002017583 SSC categorization criteria for the updated Tier. This includes use of the WCNOC corrective action process.

If the seismic hazard is reduced such that it meets the criteria for Tier 1 in EPRI 3002017583, WCNOC will implement the following process:

a. For previously completed system categorizations, WCNOC may review the categorization results to determine if use of the criteria in EPRI 3002017583 Section 2.2, "Tier 1 - Low Seismic Hazard / High Seismic Margin Sites," would lead to categorization changes. If changes are warranted, they will be implemented through the WCNOC design control and corrective action programs and NEI 00-04, Section 12.

b. Seismic considerations for subsequent system categorization activities will be performed in accordance with the guidance in 3002017583 Section 2.2, "Tier 1 - Low Seismic Hazard / High Seismic Margin Sites."

If the seismic hazard increases to the degree that an SPRA becomes necessary to demonstrate adequate seismic safety, WCNOC will implement the following process following completion of the SPRA, including adequate closure of Peer Review Findings and Observations:

a. For previously completed system categorizations, WCNOC will review the categorization results using the SPRA insights as prescribed in NEI 00-04 Section 5.3, "Seismic Assessment" and Section 5.6, "Integral Assessment." If categorization changes are warranted, they will be implemented through the WCNOC design control and corrective action programs and NEI 00-04 Section 12.

b. Seismic considerations for subsequent system categorization activities will follow the guidance in NEI 00-04, as recommended in EPRI 3002017583 Section 2.4, "Tier 3 - High Seismic Hazard / Low Seismic Margin Sites."

<u>Historical Seismic References for WCGS</u>

The WCGS GMRS and SSE curves from the seismic hazard and screening response are shown in Figure 1, as replicated from the seismic hazard and screening report [54]. The NRC staff assessment of the WCGS seismic hazard and screening response is documented in Reference [55]. In Section 3.3.3 of Reference [55], the NRC concluded that the methodology used by WCNOC adequately characterizes the seismic hazard for the WCGS site.

Section 1.1.3 of EPRI 3002017583 cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For WCGS, the specific seismic reviews prepared by WCNOC, and the NRC's staff assessments of those reviews, are provided in the following licensing documents.

1. Near-Term Task Force (NTTF) Recommendation 2.1 Seismic Hazard Screening [54], [55]
2. NTTF Recommendation 2.1 Spent Fuel Pool assessment [56], [57]
3. NTTF Recommendation 2.3 Seismic Walkdowns [58], [59], [60], [61]
4. NTTF Recommendation 4.2 Seismic Mitigation Strategy Assessment (S-MSA) [62], [63]

The following additional post-Fukushima seismic reviews were performed for WCGS:

5. NTTF Recommendation 2.1 Expedited Seismic Evaluation Process (ESEP) [64], [65]
6. NTTF Recommendation 2.1 Seismic High Frequency Evaluation [66], [67], [68]

Technical Information Incorporated by Reference

By letter dated January 31, 2020, Exelon Generation Company, LLC (EGC) submitted an LAR [31] to allow for the implementation of the provisions of 10 CFR 50.69 for LaSalle County Station (LaSalle), Units 1 and 2. Following the criteria in EPRI report 3002012988 [36], based on the GMRS-to-SSE comparison, the LaSalle site is considered a Tier 2 site, similar to the WCGS. For the LaSalle seismic hazard assessment, EGC also proposed the use of a risk-informed graded approach that meets the requirements of 10 CFR 50.69(b)(2) as an alternative to those listed in NEI 00-04. EGC provided responses to NRC RAIs pertaining to the alternative seismic approach in letters dated October 1, 2020, October 16, 2020, and January 22, 2021 [32], [33], [34]. Based on information provided in the LAR [31], as modified by the EGC RAI responses [32], [33], [34] the NRC issued the license amendments, approving the EGC request, on May 27, 2021 [35].

WCNOC will follow the same alternative seismic approach in the 10 CFR 50.69 categorization process for WCGS as that which was approved by the NRC staff for LaSalle [35], except for the site-specific LaSalle information (e.g., seismic capacity discussions, etc.). WCGS site-specific seismic capacity information is described above herein. The LaSalle RAI responses [32], [33], [34] are incorporated by reference into this LAR as they provide additional supporting bases for Tier 2 plants, such as WCGS, to adopt the alternative seismic methodology for use in the 10 CFR 50.69 categorization process. Note that EGC's October 1, 2020, letter [32] included a response to RAI 'APLC 50.69-RAI No. 12 (a)', which is not relevant to the alternative seismic approach.

In addition, References [37], [69], [70], and [71] are incorporated by reference into this LAR as they provide additional supporting bases for Tier 1 plants that are also used for Tier 2 plants.

Summary

WCGS is a Tier 2 plant for which there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs.

The special sensitivity study recommended using common cause failures, similar to the approach taken in a FPIE PRA, can identify the appropriate seismic insights to be considered with the other categorization insights by the IDP for the final HSS determinations. Use of the approach outlined in EPRI 3002017583 to assess seismic hazard risk for 10 CFR 50.69 with the additional reviews discussed above will provide a process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs that satisfies the requirements of 10 CFR 50.69(c).

3.2.4 High Winds Hazards

WCNOC has a High Wind (HW) PRA model that has been peer reviewed. All finding level facts and observations (F&Os) have been closed. However, a newly developed method, the EPRI Tornado Missile Strike Calculator (TMSC), still requires completion of the Newly Developed Method (NDM) peer review process with F&O closure. Once released, WCNOC will perform a focused scope independent peer review for this PRA upgrade. As a result, WCNOC will use a conservative approach to evaluate active component significance from HW hazards. This conservative approach is used in lieu of the existing HW PRA due to the delay in the full release of the EPRI TMSC tool, the corresponding Newly Developed Method Peer Review, and the subsequent focused peer review.

Note that there are existing significant design and risk informed insights which can be used to provide a qualitative and conservative active component significance assignment based on HW analyses performed to date. Specifically, a formal high wind fragility analysis has been performed which has evaluated plant equipment and has provided a component level screening for identifying those components susceptible to high wind hazards and those that are not (e.g., component is within a Seismic Category 1 Structure). Additionally, the WCNOC HW PRA is sufficiently developed to identify components that will be included in its scope (e.g., components in the HW PRA scope will be considered HSS).

WCNOC will use the above conservative bounding approach to identify components that are susceptible to failures due to high winds. Section 3.3 of this enclosure provides a discussion of the qualitative approach used to assess active component risk significance due to HW induced failures.

3.2.5 Other External Hazards

External events were initially screened and addressed per Generic Letter 88-20 [6]. Subsequently, external hazards were further screened for applicability to WCGS, Unit 1, per a plant-specific evaluation using the external hazard screening significance process identified in Part 6, Section 5.2 of the ASME/ANS PRA Standard RA-Sa-2009 [10]. The external hazards screening assessment was independently peer reviewed [21]. Attachment 4 of this enclosure provides a summary of the other external hazards screening results. Attachment 5 of this enclosure provides a summary of the progressive screening approach for external hazards.

3.2.6 Low Power and Shutdown

Consistent with NEI 00-04, the WCNOC categorization process will use the shutdown safety management plan described in NUMARC 91-06 [3] for evaluation of safety significance related to low power and shutdown conditions. The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.

NUMARC 91-06 specifies that a DID approach should be used with respect to each defined shutdown key safety function. The key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs.

SSCs that meet either of the two criteria (i.e., considered part of a “primary shutdown safety system” or a failure would initiate an event during shutdown conditions) described in Section 5.5 NEI 00-04 will be considered preliminary HSS.

3.2.7 PRA Maintenance and Updates

The WCNOC risk management process ensures that the applicable PRA models used in this application continue to reflect the as-built and as-operated plant. The WCNOC process delineates the responsibilities and guidelines for updating and maintaining the PRA models current with the design and operation of the plant and includes criteria for both regularly scheduled and interim PRA model updates. The WCNOC process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, equipment performance, errors or limitations identified in the model, and industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The WCNOC process directs the assessment of the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated and presented to the IDP.

In addition, WCNOC will implement procedural guidance for a process that addresses the requirements in NEI 00-04, Section 11, “Program Documentation and Change Control.” The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model change that is determined to meet the criteria of an upgrade in accordance with industry guidance [14] will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.8 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA model(s) used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the self-assessment and peer review processes as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5.

In the overall risk sensitivity studies, WCNOC will utilize a factor of three (3) to increase the unavailability or unreliability of LSS components consistent with NEI 00-04. Consistent with the NEI 00-04 guidance, WCNOC will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed, through WCNOC’s Corrective Action Program, before reaching the rate assumed in the sensitivity study.

The WCNOC IE PRA model and Fire PRA model and documentation were reviewed for plant-specific and generic modeling assumptions and related sources of uncertainty [7] [10]. The process to evaluate uncertainties is defined in NUREG-1855, Revision 1 [8] and EPRI Technical Reports TR-1016737 [9] and TR-1026511 [27]. Each PRA model includes an evaluation of the potential sources of uncertainty for the base case models using the approach that is consistent with the ASME/ANS RA-Sa-2009 [10] requirements for identification and characterization of uncertainties and assumptions.

Each PRA technical element notebook was also reviewed for assumptions and sources of uncertainties. The characterization of assumptions and sources of uncertainties are based on whether the assumption and/or source of uncertainty is key to the 10 CFR 50.69 application in accordance with RG 1.200 Revision 2.

Key WCNOC PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 6 of this enclosure. The conclusion of this review is that no additional sensitivity analyses are required to address WCNOC PRA model specific assumptions or sources of uncertainty.

3.2.9 Modeling of FLEX

No credit is taken for FLEX strategies within any WCNOC PRA model.

3.3 PRA Review Process Results (10 CFR 50.69(b)(2)(iii))

The PRA models identified in Section 3.2 have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 [7], consistent with NRC Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," [19] and through comment resolutions per NRC Letter [13]. Except as noted for the high winds PRA, all other PRA models used in the proposed 10 CFR 50.69 categorization process (i.e., internal events/floods, fire) have undergone independent peer review.

F&O closure reviews were conducted on the PRA models discussed in this section. Findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations (F&Os)" [11], as accepted by NRC in the letter dated May 3, 2017 [12]. The results of this review have been documented and are available for NRC audit.

3.3.1 Internal Events (IE) and Internal Flooding (IF) PRA Models

The current IE (MOR 10) and IF (MOR 09) PRA models have been peer reviewed and have no findings remaining open against them. The initial peer review of PRA technical adequacy was for the 2019 WCGS IE PRA [16]. This peer review was performed during the week of June 17-21, 2019. The assessment evaluated the IE PRA (including internal flooding) against the requirements published in the current version of the ASME/ANS PRA Standard (RA-Sa-2009) [10] and RG 1.200 [7]. This independent peer review was followed up with an initial facts and observation closure effort [17]. Final closure of remaining finding level F&Os is documented in [18].

3.3.2 Fire PRA Model

The WCNOC Fire PRA peer review [20] was performed against the requirements of Part 4 of the ASME/ANS PRA Standard (RA-Sa-2009) [10]. The peer review also included the clarifications and qualifications provided in the NRC endorsement of the PRA Standard, contained in Revision 2 to RG 1.200 [7]. This peer review was performed using the process defined in NEI 17-07 [22]. An independent peer review was documented in "Peer Review for Fire PRA" [20]. Subsequently, a focused peer review was performed [24]. The focused scope peer review evaluated the specific PRA Standard Supporting Requirements (SRs) pertaining to the Seismic Fire element published in the current version of the ASME/ANS PRA Standard (RA-Sa-2009) [10] as clarified by RG 1.200. Subsequently, an independent assessment of the F&O closures was performed in October 2022 [23]. This led to a separate evaluation of seismic-fire interactions where no additional findings were identified [28].

A focused scope peer review of the technical adequacy of the seismic fire interactions element was required as this was not in scope PWROG-21032-P [20]. No Finding F&O was issued; however, two suggestions were issued.

3.3.3 High Winds PRA Model

The HW PRA model is still pending review for an upgrade following completion of an industry newly developed method review [14]; however, the HW PRA, is a highly developed PRA model that meets PRA Standard requirements, has been independently peer reviewed, and has closed all peer review F&Os. What remains is the peer review for the industry's newly developed method for assessing turbine missile strike probabilities. The current HW PRA contains sufficient information and risk insights to support a qualitative but conservative approach to assess component risk significance due to high winds. Thus, in lieu of using quantitative results from the HW PRA, a conservative and bounding qualitative approach will be used for determining SSC active risk significance. If at some point in the future it is desired to use quantitative results from the HW PRA during 10 CFR 50.69 categorization, then prior NRC approval will be requested, in accordance with 10 CFR 50.90.

The qualitative approach described below will be used for establishing a High Wind qualitative component level active risk significance:

1. Per High Wind Fragility Analysis, if a component **is** "high wind impact screened" with justification that the component is within a Seismic Category 1 structure, then it is not susceptible to a High Winds failure, and thus, is assigned HW LSS.

   Remark: Components within Seismic Category 1 structures do not have failures due to high wind exposure. This is true for all components within Seismic Category 1 structures with the exception of certain components in the BN system (RWST and Isolation Valve Room) which are in Category 1 seismic structures but are not designated "high wind impact screened." This is due to conservative modeling of Refueling Water Storage Tank (RWST) failure due to high wind failure of the Radwaste Building wall.

2. 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 not** included in the High Wind Plant Response Model and Quantification, then the component is assigned HW LSS.

   Remark: This accounts for components that are mostly balance of plant equipment or are items not generally modeled in the PRA. The WCNOC HW PRA is technically sufficient such that if a component is not modeled, then it is not a risk significant contributor and is not required to achieve a safe stable plant state. So, these components would be assigned HW LSS.

3. 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, then the component is assigned HW HSS.

   Remark: This accounts for those components that **are not** "high wind impact screened" per the High Wind Fragility Analysis **but are included** in the High Wind Plant Response Model and Quantification. This represents the conservative nature of this approach since quantified PRA results might indicate certain components in the HW PRA scope could have had importances meeting LSS criteria.

3.4 Risk Evaluations (10 CFR 50.69(b)(2)(iv))

The WCNOC 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions and meets the requirements of § 50.69(b)(2)(iv). Sensitivity studies described in NEI 00-04, Section 8 will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, and human errors, as applicable). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.

3.5 Feedback and Adjustment Process

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, a timely evaluation and review will be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information will be performed during the normally scheduled periodic review cycle.

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 discussed in Section 3.2.3 of this submittal, implementation of the WCNOC 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).

The performance monitoring process will be described in WCNOC's 10 CFR 50.69 program documents. The program requires that the periodic review assess changes that could impact the categorization results and provides the IDP with an opportunity to recommend categorization and treatment adjustments. Personnel from engineering, operations, PRA, regulatory affairs, and others have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability, to help identify and reverse negative performance trends and take corrective action if necessary.

The WCNOC configuration control process requires that changes to the plant, including physical changes to the plant and changes to documents, are evaluated to determine the impacts to drawings, design bases, licensing documents, programs, procedures, and training.

WCNOC has a comprehensive problem identification and corrective action program that requires the identification and resolution of issues. Issues that may impact the 10 CFR 50.69 categorization process will be identified and addressed through the problem identification and corrective action program, including seismic-related issues.

The WCNOC 10 CFR 50.69 program requires that system categorization cannot be approved by the IDP until the panel's comments have been resolved to the satisfaction of the IDP. This includes issues related to system-specific seismic insights considered by the IDP during categorization.

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. This review will include:

- A review of plant modifications since the last review that could impact the SSC categorization,
- A review of plant-specific operating experience that could impact the SSC categorization,
- A review of the impact of the updated risk information on the categorization process results,
- A review of the importance measures used for screening in the categorization process, and
- An update of the risk sensitivity study performed for the categorization.

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is upgraded, a review of the SSC categorization will be performed.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following NRC requirements and guidance documents are applicable to the proposed Change:

- The regulations in Title 10 of the Code of Federal Regulations (10 CFR) Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."

- Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006 [2].

- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 [29].

- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009 [7].

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 No Significant Hazards Consideration Analysis

Wolf Creek Nuclear Operating Corporation (WCNOC) proposes to modify the licensing basis of Wolf Creek Generating Station (WCGS) to allow for the voluntary implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

WCNOC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of Structures, Systems and Components (SSCs) subject to Nuclear Regulatory Commission (NRC) special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

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

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

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC.

Under the proposed change, no additional plant equipment will be installed. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any Safety Limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, WCNOC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.3 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categoric exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.

2. NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.

3. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.

4. NRC letter to Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant Units 1 and 2 - Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC Nos. ME9472 and ME94473)," December 17, 2014 (ADAMS Accession No. ML14237A034).

5. NRC letter to Entergy Operations, Inc., "Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems (TAC No. MD5250)," April 22, 2009 (ADAMS Accession No. ML090930246).

6. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f) (Generic Letter No. 88-20, Supplement 4)," US NRC, June 28,1991.

7. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

8. NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking, Revision 1, March 2017.

9. EPRI TR-1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, December 2008.

10. ASME/ANS RA-Sa-2009, Standard for Level I/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008, February 2009.

11. NEI Letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," February 21, 2017 (ADAMS Accession No. ML17086A450 and ML17086A451).

12. NRC Letter to NEI, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," May 3, 2017 (ADAMS Accession No. ML17079A427).

13. NRC Letter to ASME, "U.S. Nuclear Regulatory Commission (NRC) Comments on 'Addenda to a Current ANS: ASME RA-SB - 20XX, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment For Nuclear Power Plant Applications," July 6, 2011 (ADAMS Accession No. ML111720076).

14. PWROG-20037-WCNOC, PRA Upgrade/Maintenance and Newly Developed Method Examples PA-RMSC-1647, Maioli, Revision 0-B, March 2022.

15. NUREG/CR 6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, EPRI 1011989, September 2005.

16. WCNOCPES029-REPT-001, Revision 0, "Wolf Creek Internal Events Probabilistic Risk Assessment Peer Review," September 9, 2019.

17. PWROG-19038-P, Revision 0, "Independent Assessment of Facts & Observations Closure of the Wolf Creek Probabilistic Risk Assessment," March 2020.

18. PWROG-23024-P, Revision 0, "Independent Assessment of Facts & Observations Closure of the Wolf Creek Probabilistic Risk Assessments," November 2023.

19. NRC RIS 2007-06, "NRC Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," March 22, 2007."

20. PWROG-21032-P, Revision 0, "Peer Review of the Wolf Creek Fire Probabilistic Risk Assessment," Rev. 0, January 2022.

21. PWROG 15082-P, Rev. 0, "Peer Review of Wolf Creek Generating Station External Events Screening and High Winds Probabilistic Risk Assessment," December 2015.

22. NEI 17-07, Revision A, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," Nuclear Energy Institute, December 2017.

23. WCNOC-22026-P, "Independent Assessment of Facts and Observations Closure of the Wolf Creek Fire PRA," Rev 0, November 2022.

24. NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Revision 1, June 2010.

25. NRC Letter to WCNOC, "Wolf Creek Generating Station, Unit 1 – Issuance of Amendment No. 227 Re: Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program Based on TSTF-425 (EPID L-2020-LLA-0091)," April 8, 2021, (ADAMS Accession No. ML21053A117).

26. Not used

27. EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," December 2012.

28. WCNOC-00076-REPT-005, Revision 0, Special Scope Seismic-Fire Interaction - "Wolf Creek Generating Station Fire Probabilistic Risk Assessment Focused Scope Peer Review," November 2023.

29. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.

30. Electric Power Research Institute (EPRI) 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Technical Update, February 2020 (ADAMS Accession No. ML21082A170).

31. Exelon Generation Company (LaSalle) Letter to NRC, "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," January 31, 2020 (ADAMS Accession No. ML20031E699).

32. Exelon Generation Company (LaSalle) Letter to NRC, "Response to Request for Additional Information regarding LaSalle License Amendment Request to Renewed Facility Operating Licenses to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," October 1, 2020 (ADAMS Accession No. ML20275A292).

33. Exelon Generation Company (LaSalle) Letter to NRC, "Response to Request for Additional Information regarding LaSalle License Amendment Request to Renewed Facility Operating Licenses to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors'," October 16, 2020 (ADAMS Accession No. ML20290A791).

34. Exelon Generation Company (LaSalle) Letter to NRC, "Response to Request for Additional Information Regarding the License Amendment Request to Adopt 10 CFR 50.69," January 22, 2021 (ADAMS Accession No. ML21022A130).

35. NRC Letter to Exelon Generation Company, "LaSalle County Station, Unit Nos. 1 and 2 - Issuance of Amendment Nos. 249 and 235 Related to Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors'," May 27, 2021 (ADAMS Accession No. ML21082A422).

36. Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," July 2018.

37. NRC letter to Exelon Generation Company, "Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Issuance of Amendment Nos. 332 and 310 Re: Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, February 28, 2020 (ADAMS Accession No. ML19330D909).

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39. NRC Letter to all Power Reactor Licensees, “Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 12, 2012 (ADAMS Accession No. ML12053A340).

40. Exelon Generation Company (Peach Bottom) Letter to NRC, "Seismic Probabilistic Risk Assessment Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," August 28, 2018 (ADAMS Accession No. ML18240A065).

41. NRC Letter to Exelon Generation Company, “Peach Bottom Atomic Power Station, Units 2 and 3 – Staff Review of Seismic Probabilistic Risk Assessment Associated With Reevaluated Seismic Hazard Implementation of the Near-Term Task Force Recommendation 2.1: Seismic,” June 10, 2019 (ADAMS Accession No. ML19053A469).

42. NRC Letter to Exelon Generation Company, “Peach Bottom Atomic Power Station, Units 2 and 3 – Correction Regarding Staff Review of Seismic Probabilistic Risk Assessment Associated With Reevaluated Seismic Hazard Implementation of the Near-Term Task Force Recommendation 2.1: Seismic,” October 8, 2019 (ADAMS Accession No. ML19248C756).

43. Southern Nuclear Operating Company Letter to NRC, “Vogtle Electric Generating Plant – Units 1 and 2 License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process,” June 22, 2017 (ADAMS Accession No. ML17173A875).

44. NRC Letter to Southern Nuclear Operating Company, “Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Application of Seismic Probabilistic Risk Assessment into the Previously Approved 10 CFR 50.69 Categorization Process,” August 10, 2018 (ADAMS Accession No. ML18180A062).

45. Tennessee Valley Authority Letter to NRC, “Seismic Probabilistic Risk Assessment for Watts Bar Nuclear Plant, Units 1 and 2 - Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” June 30, 2017 (ADAMS Accession No. ML17181A485).

46. Tennessee Valley Authority Letter to NRC, “Tennessee Valley Authority (TVA) - Watts Bar Nuclear Plant Seismic Probabilistic Risk Assessment Supplemental Information,” April 10, 2018 (ADAMS Accession No. ML18100A966).

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54. WCNOC Letter WO 14-0042 to NRC, "Wolf Creek Nuclear Operating Corporation's Seismic Hazard and Screening Report (CEUS Sites), Response NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 31, 2014 (ADAMS Accession No. ML14097A020).

55. NRC Letter to WCNOC, "Wolf Creek Generating Station - Staff Assessment of Information Provided Pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," August 12, 2015 (ADAMS Accession No. ML15216A320).

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57. NRC Letter to WCNOC, "Wolf Creek Generating Station - Staff Review of Spent Fuel Pool Evaluation Associated with Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1," December 9, 2016 (ML16335A371).

58. WCNOC Letter ET 12-0032 to NRC, "Wolf Creek Nuclear Operating Corporation 180-Day Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.3 (Seismic) of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, November 27, 2012 (ADAMS Accession No. ML123420431).

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69. Exelon Generation Company (Calvert Cliffs) Letter to NRC, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," July 1, 2019 (ADAMS Accession No. ML19183A012).

70. Exelon Generation Company (Calvert Cliffs) Letter to NRC, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," July 19, 2019 (ADAMS Accession No. ML19200A216).

71. Exelon Generation Company (Calvert Cliffs) Letter to NRC, “Revised Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors’,” August 5, 2019 (ADAMS Accession No. ML19217A143).

Attachment 1: List of Categorization Prerequisites

WCNOC will establish procedure(s) prior to the use of the categorization process on a plant system. The procedure(s) will contain the elements/steps listed below.

- Integrated Decision-Making Panel (IDP) member qualification requirements.

- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significance (HSS) or Low Safety Significance (LSS) based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.1 of this enclosure for this license amendment request). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting, an LSS function are categorized as preliminary LSS.

- Component safety significance assessment. Safety significance of active components is assessed through a combination of Probabilistic Risk Assessment (PRA) and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.

- Assessment of defense-in-depth (DID) and safety margin. Safety-related components that are categorized as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.

- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.

- Risk sensitivity study. For PRA-modeled components, an overall bounding risk sensitivity study, which conservatively assumes increased component failure rates for LSS components, is used to confirm that the population of preliminary LSS components with increased failure rates results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.

- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those Structures, Systems, and Components that have been categorized.

- Documentation requirements per Section 3.1.1 of this enclosure.

Attachment 2: Description of Probabilistic Risk Assessment (PRA) Models of Record

PRA Model of Record (MOR)	Baseline CDF	Baseline LERF	Comments
WCNOC Internal Events (IE) PRA MOR 10	7.11E-06	7.87E-08	All finding level F&Os closed.
WCNOC Internal Flood (IFPRA) PRA MOR 09	9.16E-06	3.77E-08	All finding level F&Os closed.
WCNOC Fire (FPRA) PRA MOR 10	4.27E-05	7.76E-07	All finding level F&Os closed.

Attachment 3: DISPOSITION AND RESOLUTION OF OPEN PEER REVIEW FINDINGS AND SELF-ASSESSMENT OPEN ITEMS

Internal Events

F&O	Status	SRs	F&O Description	Disposition
All finding level F&Os have been closed per industry and regulatory guidance.				

Internal Flood PRA

F&O	Status	SRs	F&O Description	Disposition
All finding level F&Os have been closed per industry and regulatory guidance.				

Fire PRA

F&O	Status	SRs	F&O Description	Disposition
All finding level F&Os have been closed per industry and regulatory guidance.				

Attachment 4: EXTERNAL HAZARDS SCREENING

A calculation "Other External Hazards Screening" was developed to evaluate the applicability for other external hazards.
All the external hazards that could potentially affect the WCGS site were screened out based on the criteria specified in SRs EXT-B1 or EXT-C1 of the ASME/ANS PRA Standard, [10].

See Table 4-4 below. This table uses screening criteria that is described in the ASME/ANS PRA Standard 2009 [10]. A peer review was performed on the "Other External Hazards Screening" [21]. Additionally, bounding support calculations were performed to evaluate and close remaining external event screenings related to aircraft impacts, lightning strikes, and local intense precipitation.

Note: References to "Sections" in the "Disposition for 10 CFR 50.69" column are with respect to the "Other External Hazards Screening" analysis.

Table 4-4: Hazard Dispositions

Hazard	50.69 Screening Criterion	Disposition for 10 CFR 50.69
Avalanche	3	Topography is such that no avalanche is possible.
Biological Events	5	There is sufficient time to respond to these hazards before Service Water System or Essential Service Water System operation would be jeopardized.
Coastal Erosion	3	The site is inland and not subject to this hazard.
Landslides	3	Not considered possible because of the absence of topographic and geologic features conducive to landslide formation in the vicinity of the plant site.
Sinkholes	3	The closest known sinkholes are significantly more than 5 miles from the site.
Soil Shrink-Swell	1	The hazard was considered in the construction and operation of the WCGS buildings and structures.
Drought	1, 5	The hazard was considered in the design of the Ultimate Heat Sink (UHS). In addition, there would be sufficient time to respond to the hazard before plant operation would be jeopardized.
External Flooding(Local) Intense Precipitation)	Screened on quantitative criterion C of SR EXT-C1	A bounding analysis for the LIP hazard was performed and documented in accordance with industry standards. This bounding analysis demonstrates that the CDF for this hazard is $<10^{-6}$/yr. Therefore, this hazard can be screened out from further consideration based on Criterion C of SR EXT-C1.
High Winds,	N/A	WCNOC will use an alternative screening approach that is conservative and bounding, which

Attachment 4: EXTERNAL HAZARDS SCREENING

Hazard	50.69 Screening Criterion	Disposition for 10 CFR 50.69
Extreme Winds, Hurricanes, and Tornadoes		assesses active component risks due to high winds, as described in the LAR.
Fog	4	Fog affects the frequency of occurrence of air, land, and water transportation hazards and is indirectly considered in those hazards.
External Fire: Forest Fire / Grass Fire	3	For Forest Fire, the event cannot occur close enough to the plant to affect it. See Section 8.1.2.1. For Grass Fire, the event cannot occur close enough to the plant to affect it. Also, the plant design and fire-protection provisions are adequate to mitigate the effects.
Frost	1	Frost has lesser damage potential than snow and ice.
Frazil Ice	5	The effects of frazil ice are mitigated by diffusing warmed water in front of the ESW trash racks. In addition, the CCWS, in conjunction with the ESWS, provides sufficient heat energy to maintain the ESWS inlet trash racks from being blocked with frazil ice. Operations procedure STS CR-001 directs operations personnel to check the temperature of the UHS and ESW two (2) times per day. Therefore, the event is slow in developing and there would be sufficient time to respond to it.
Hail	Screened on quantitative criterion C of SR EXT-C1	Screened by calculating a conservative bounding CDF assuming a LOOP initiating event and bounded by the consequences of an EF0 tornado.
High Summer Temperature	1	USAR Section 2.3.2.3 states that temperature extremes data were used in the design of safety-related equipment exposed to ambient environment conditions. Therefore, the high summer temperature hazard can be screened out based on Criterion 1 (i.e., the event is of equal or lesser damage potential than the events for which the plant has been designed).
High Tide / High Lake Level	3	Does not apply to the site.
Ice Cover	1, 4	See “Snow and Ice” disposition.
Military Facility	3	An accidental explosion from a nearby military facility will not pose a hazard to the plant.

Attachment 4: EXTERNAL HAZARDS SCREENING

Hazard	50.69 Screening Criterion	Disposition for 10 CFR 50.69
Accident		
Industrial Accident	1,3	An accidental explosion from a nearby industrial facility will not pose a hazard to the plant. In addition, a toxic chemical release from a nearby industrial facility will not pose a hazard to the plant.
Internal Fire	Detailed PRA	WCNOC will use its approved WCNOC Fire PRA.
Internal Flooding	Detailed PRA	WCNOC will use its approved WCNOC Internal Flooding PRA
Lightning	Screened on quantitative criterion C of SR EXT-C1	This hazard could be screened out using Criterion 1 or Criterion 4 of SR EXT-B1. Despite that, it was decided to present this hazard as failing the preliminary screening, thus requiring a bounding analysis. This was done to provide a stronger quantitative reasoning for screening out this hazard. Reference [21] provides the peer review screening analysis performed for external events. Per Reference [21]: "Section 4.2 of Other External Hazards: Bounding Support Calculations, provides the methodology for performing the Lightning Strike (LS) CDF calculation to support quantitative screening (i.e., < 1.0E-06). The limiting scenario was assumed to be a LOOP with a loss of both DC Trains. This implies that the CCDP is based a loss of both DC Trains." "Section 8.2 of Other External Hazards: Bounding Support Calculations, provides the Lightning Strike (LS) CDF calculation to support quantitative screening (i.e., < 1.0E-06). The calculations actually use bounding CCDP values based on the plant-specific PRA model." Therefore, this hazard can be screened out from further consideration based on Criterion C of SR EXT-C1.
Low lake or river water level	1	Accounted for in the design of the UHS.
Low winter temperature	1	Accounted for in the design of safety-related equipment exposed to ambient environment conditions.

Attachment 4: EXTERNAL HAZARDS SCREENING

Hazard	50.69 Screening Criterion	Disposition for 10 CFR 50.69
Meteorite/Satellite strikes	2	The event has a low initiating event frequency (less than 1.0E-9).
Mining Accident	3	Nearest mining facility is more than 5 miles from the plant and therefore, this hazard can be screened out based on Criterion 3 of SR EXT-B1 (i.e., the event cannot occur close enough to the plant to affect it).
Pipeline Accident	1	The bounding analysis discussed in USAR Section 2.2.3.1.2 remains valid. This analysis demonstrated a probability of 1.3×10^{-10} per year for the worst case pipeline accident. See Section 8.1.9.4.
Release of Chemicals from On-site Storage	1	Per the "Other External Hazards Screening," there are no control room habitability problems from the potential release of hazardous chemicals from the onsite storage facilities.
River Diversion	1	The hazard was considered in the design of the UHS.
Sandstorm	3	Per the NOAA website search documented in Appendix A.2, no dust or sandstorms have been recorded in the area near the plant.
Seiche	3	This hazard is not applicable to the site.
Seismic Activity	N/A	The seismic hazard will be assessed using the Alternative Approach for Tier 2 plants as described in EPRI 3002017583 and as described in this LAR.
Snow and Ice	1,4	USAR Section 2.4.2.3.3 states that the design of the safety-related structures at the plant considered maximum snow loadings on the roofs and the clogging of inlets and certain size culverts by ice. In addition, USAR Section 2.4.7.1 states that the potential for ice flooding in the site area is minimal and not a safety-related factor.
Storm Surge	3	This hazard is not applicable to the site.
Toxic Gas	4	This hazard is included under the "Release of Chemicals from Onsite Storage" hazard.
Air Transportation: Aircraft Impacts	Bounding Calculation performed	This hazard could not be screened out based on the SR EXT-B1 criteria. The bounding analysis demonstrates that the CDF for this hazard is $<10^{-6}$/yr. Therefore, this hazard can be screened out from further consideration based on Criterion C of SR EXT-C1. A bounding analysis for the Aircraft Impact hazard is documented in the "Other External Hazards Screening Evaluation." This bounding analysis demonstrates that the CDF for this hazard is $<10^{-6}$/yr. Therefore, this hazard can be screened out from further consideration based

Attachment 4: EXTERNAL HAZARDS SCREENING

Hazard	50.69 Screening Criterion	Disposition for 10 CFR 50.69
		on Criterion C of SR EXT-C1 of the ASME/ANS PRA Standard [10].
Land Transportation: Roads/Highways: Vehicle Impact	3	Physical damage to the plant caused by a truck colliding with plant structures is considered minimal due to the distance between main roads and highways and the plant structures
Land Transportation: Roads/Highways: Vehicle Explosion	3	Due to the long distance from the closest road (rural or highway) to the plant, any accidental explosion will not endanger the safe operation of the plant.
Railroads: Railroad Explosion	3	Due to the long distance from the nearest existing railroad to the plant (9.5 miles), any accidental explosion will not endanger the safe operation of the plant.
Water Transportation: Collisions with Intake Structures	3	There is no potential for collision of vessels with the intake structures.
Water Transportation: Explosion	3	There is no potential for explosions from vessels.
Tsunami	3	This hazard is not applicable to the site.
Turbine-generated Missiles	1	With the new monoblock turbine rotor design, the annual probability of unacceptable damage resulting from a turbine missile (P4) was reduced to 2.44×10^{-9} per year, which is less than the 10^{-7} per year limit of RG 1.115 and NUREG-1048.
Volcanic Activity	3	The site is not located near any active volcano.
Waves	1,4	The effects of wind-generated waves and flood waves caused by the potential failures of dams on the Coffey County Lake (formerly Wolf Creek Lake) water level are incorporated into the Probable Maximum Flood (PMF) for the site, which is used in bounding external flooding calculations. In addition, a flood wave produced by a landslide in the area adjacent to the cooling lake is not considered possible.

Attachment 5: PROGRESSIVE SCREENING APPROACH FOR ADDRESSING EXTERNAL HAZARDS

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	Event damage potential is ≤ events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	Event has a significantly lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	SR EXT-C1, Criterion A. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	SR EXT-B2 Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	SR EXT-C1, Criteria B & C Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	Criterion C of SR EXT-C1 Bounding mean CDF is < 1E-6/y.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
Detailed PRA	Screening not successful. Probabilistic Risk Assessment (PRA) needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Attachment 6: DISPOSITION OF KEY ASSUMPTIONS/SOURCES OF UNCERTAINTY

The WCNOC probabilistic risk assessment (PRA) models and documentation were reviewed for plant-specific modeling assumptions and related sources of uncertainty. The WCNOC Assumptions and Sources of Uncertainty and Fire PRA uncertainty notebooks document sources of PRA modeling uncertainty. To assess seismic risks, the process will use the EPRI alternative seismic approach for Tier 2 plants. In Tier 2, there may be a limited number of unique seismic insights to consider. These insights would most likely be attributed to the possibility of seismically correlated failures or seismic interaction related failures. Therefore, a special sensitivity study is performed to identify unique seismic insights, which are considered when making final HSS determinations. The approach for High Winds is conservative in that components within seismic category 1 structures are screened out from a wind fragility perspective. Those components that are not screened (i.e., they are assigned a wind fragility group) and that are included in the High Wind plant response model and quantification are assigned HSS regardless of their PRA importance. The deterministic screening of components being in a Category 1 structure combined with being identified for inclusion in WCNOC's near term High Wind PRA provides a reasonable conservative and bounding approach relative to uncertainty.

The Internal Events and Fire PRAs identify assumptions and determine if those assumptions are related to sources of model uncertainty and characterize that uncertainty, as necessary. The identified uncertainties were reviewed for this application. These PRA models include an evaluation of the potential sources of uncertainty for the base case models using the approach that is consistent with the ASME/ANS RA-Sa-2009 [10] requirements for identification and characterization of uncertainties and assumptions. This evaluation identifies those sources of uncertainty that are important to the PRA results and may be important to PRA applications which meets the intent of steps C-1 and E-1 of NUREG-1855, Revision 1 [8].

The results of the base PRA evaluations were reviewed to determine which potential uncertainties could impact the 10 CFR 50.69 categorization process results. This evaluation meets the intent of the screening portion of steps C-2 and E-2 of NUREG-1855, Revision 1 [8].

The evaluation of uncertainty for internal events and fire addressed generic sources of uncertainty and assessed other uncertainty sources specific to the PRA modeling of the WCNOC design and operation. The degree to which uncertainties impact a specific accident progression varies but was determined to not be significant overall. The impact of uncertainty could be greater on "accident sequence specific" components or human actions but would not have a significant impact to the overall PRA results.

Markup of Appendix D of Facility Operating License (FOL) for Proposed Change

Amendment Number	Additional Condition	Implementation Date
179 (Cont'd)	(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from August 16, 2004, the date of the most recent successful tracer gas test, as stated in the November 16, 2004, letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.	
	(c) The first performance of the periodic measurement of control room pressure, Specification 5.5.18.d, shall be within 18 months, plus the 138 days allowed by SR 3.0.2, as measured from February 2, 2007, the date of the most recent successful pressure measurement test.	
213	Automated Statistical Treatment of Uncertainty Method (ASTRUM), as corrected for thermal conductivity degradation (TCD) including the use of PAD 4.0 + TCD, has specifically been approved for use in the WCGS licensing basis analyses. Upon NRC approval of a revised generic best-estimate loss-of-coolant accident (LOCA) analysis methodology and fuel performance analysis methodology that accounts for TCD and is applicable to the fuel in use at WCGS, WCNOC will within 6 months, either: (a) Demonstrate that the WCGS safety analyses remain conservatively bounded in licensing basis analyses when compared to the new generically approved version of the LOCA analysis methodology and fuel performance analysis methodology that accounts for TCD, or (b) Provide a schedule for re-analysis of any of the affected licensing basis analyses using the new generically approved version of the LOCA analysis methodology and fuel performance analysis methodology that accounts for TCD.	Within 6 months of NRC approval of a revised methodology that accounts for TCD

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XXX	WCNOC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC) RISC-1, RISC-2, RISC3, and RISC-4 structures, systems, and components (SSCs) using: probabilistic risk assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2, Class 3, and non-class SSCs and their associated supports; 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 ET 25-000779 dated January 30, 2025; as specified in License Amendment No. [XXX] dated [DATE]. 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.