1717 Wakonade Drive
Welch, MN 55089

800.895.4999
xcelenergy.com

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L-PI-18-012
10 CFR 50.90
10 CFR 50.69

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

Prairie Island Nuclear Generating Plant, Units 1 and 2
Docket Nos. 50-282 and 50-306
Renewed Facility Operating License Nos. DPR-42 and DPR-60

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

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, is requesting an amendment to the Renewed Facility Operating License (RFOL) of the Prairie Island Nuclear Generating Plant (PINGP).

The proposed amendment would modify the PINGP licensing basis, by the addition of a License Condition, 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." 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.

The enclosure to this letter provides the basis for the proposed change to the PINGP RFOL. The categorization process being implemented through this change is consistent with Nuclear Energy Institute (NEI) Report NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline", Revision 0, dated July 2005, which was endorsed by the NRC in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance", Revision 1, dated May 2006. Attachment 1 of the enclosure provides a list of categorization prerequisites. Use of the categorization process on a plant system will only occur after these prerequisites are met.

The internal events, including internal flooding, Probabilistic Risk Assessment (PRA) model described within this license amendment request (LAR) is the same as the one described within NSPM submittal of the LAR dated March 15, 2018, to adopt TSTF-425 (ADAMS Accession No. ML18074A308) and the LAR dated May 18, 2018, to modify the list of required NFPA 805 modifications (ADAMS Accession No. ML18138A402). The fire PRA model described within this LAR is the same as the one referenced in the NSPM submittal of the LAR dated May 18, 2018, as supplemented on July 10, 2018, to modify the list of required NFPA 805 modifications. NSPM requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the applications currently in-process. This would reduce the number of NSPM and NRC resources necessary to complete the review of the applications. This request should not be considered a linked request as the details of the PRA models in each LAR are complete, which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other applications.

NSPM requests approval of the proposed change by August 31, 2019, with an implementation period of 90 days.

In accordance with 10 CFR 50.91(b)(1), a copy of this application, with attachments, is being provided to the designated Minnesota Official.

If additional information is required, please contact Mr. Shane Jurek at (612) 330-5788.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on July 20, 2018.

Scott Sharp

Scott Sharp
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

ENCLOSURE

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Evaluation of the Proposed Change

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1. 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 low safety significant (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be high safety significant (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. 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 the DBEs are defined as “safety-related.” These SSCs are the subject of many 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 50, is not explicitly defined.

2.2 REASON FOR 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 systems 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 LSS, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be 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) Report NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline” (Reference 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 SSC 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 Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, to improve focus on equipment that has safety significance resulting in improved safety at the Prairie Island Nuclear Generating Plant (PINGP).

2.3 DESCRIPTION OF THE PROPOSED CHANGE

NSPM proposes the addition of the following condition to the Renewed Facility Operating License (RFOL) of the PINGP to document the NRC's approval of the use of 10 CFR 50.69.

> NSPM is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3 and RISC-4 structures, systems and components specified in the license amendment request dated July 20, 2018.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).
>
> NSPM shall complete the modifications listed in Table A-1 of the license amendment request dated July 20, 2018, prior to implementation.

3. 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.
>
> (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.
>
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).
>
> (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).

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

The internal events, including internal flooding, PRA model described within this license amendment request (LAR) is the same as the one described within NSPM's submittal of the LAR dated March 15, 2018, to adopt TSTF-425 (Reference 2) and the LAR dated May 18, 2018, as supplemented on July 10, 2018, to modify the list of required NFPA 805 modifications (References 3 and 4). The fire PRA model described within this LAR is the same as the one referenced in NSPM's submittal of the LAR submitted in References 3 and 4. NSPM requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the applications currently in-process. This would reduce the number of NSPM and NRC resources necessary to complete the review of the applications. These requests should not be considered linked requests as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other applications.

3.1 CATEGORIZATION PROCESS DESCRIPTION (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

NSPM 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" (Reference 5). 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. 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. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and, as long as they are all completed, they may even be performed in parallel. 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 evaluations (e.g., the internal events, internal flooding, and fire PRAs)
2. Non-PRA approaches (e.g., fire safe shutdown equipment list (SSEL), seismic SSEL, other external events screening, and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04

4. The defense in depth 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. 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 only 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. This is also summarized in Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

The mapping of components to system function 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 defense in depth 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 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). These components from the component level assessments will remain HSS (i.e., IDP cannot override) regardless of the significance of the functions to which they are mapped. Therefore, if a HSS component is mapped to a LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven to HSS based on Table 3-1, or may remain LSS.

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 Events Base Case (Sections 5.2, 5.3 and 5.4)		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment (Section 5.6)		Not Allowed	Yes
Risk (Non-Modeled)	Fire, Seismic and Other External Hazards (Sections 5.2, 5.3 and 5.4)	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*	N/A
Passive	Passive (Section 4)	Segment/Component	Not Allowed	No

* The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2 of NEI 00-04. 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. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that 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 of 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 details 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 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 (mechanical and electrical) 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 defense in depth philosophy and requirements to maintain this philosophy.

- The decision criteria for the IDP for categorizing SSCs as HSS or LSS pursuant to 10 CFR 50.69(f)(1) will be documented in NSPM procedures. Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. If a resolution cannot be achieved concerning the safety significance of an SSC, then the SSC will be classified as HSS.

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

- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 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 NRC Safety Evaluation (SE) (Reference 6) approving the Vogtle license amendment to adopt 10 CFR 50.69, 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 defense in depth 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 consider whether the active function is called out or relied upon in the plant Emergency/Abnormal Operation Procedures, NSPM will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

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

- Internal Event Risks: Internal events, including internal flooding, PRA model Revision 5.3. The internal events, including internal flooding, PRA model described within this LAR is the same as the one described within the NSPM submittal of the LAR to adopt TSTF-425 (Reference 2) as well as the in the LAR to modify the list of required NFPA 805 modifications (References 3 and 4).

- Fire Risks: Fire PRA model Revision 5.3-APP1. The fire PRA model described within this LAR is the same as the one described within the NSPM submittal of the LAR to modify the list of required NFPA 805 modifications (References 3 and 4).

- Seismic Risks: SSEL referenced in the Individual Plant Examination of External Events (IPEEE) seismic analysis accepted by NRC Staff Evaluation Report dated May 29, 2001 (Reference 7).

- Other External Risks (e.g., tornados, external floods): Using the IPEEE screening process as approved by the NRC in Reference 7. The other external hazards were determined to be insignificant contributors to plant risk.

- Low Power and Shutdown Risks: Qualitative defense in depth shutdown model for shutdown configuration risk management based on the framework for defense in depth provided in Nuclear Management and Resource Council (NUMARC) Report NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" (Reference 8), which provided 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 a seismic margins 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 support 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 as approved by the NRC in Reference 9.

The RI-RRA methodology is a risk-informed safety classification 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., defense in depth, safety margins) in determining safety significance. Component supports are assigned the same safety significance as the highest passively ranked segment 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.

The use of this method was previously approved to be used for a 10 CFR 50.69 application by the NRC in Reference 6. 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 change in treatment. The passive categorization process is intended to apply the same risk-informed process approved for use at ANO for passive categorization of American Society of Mechanical Engineers (ASME) Code 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. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned HSS for passive categorization. This results in an HSS risk-informed safety classification that cannot be changed by the IDP. Therefore, the RI-RRA methodology and scope for passive categorization is acceptable and appropriate for use at PINGP 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 peer reviewed and there are no PRA upgrades that have not been peer reviewed.

3.2.1 Internal Events and Internal Flooding

The PINGP categorization process for internal events and flooding hazards will use the plant-specific PRA model. The NSPM risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant. Attachment 2 to this enclosure identifies the applicable internal events PRA model, which encompasses internal flooding.

3.2.2 Fire Hazards

The PINGP categorization process for fire hazards will use a peer-reviewed, plant-specific fire PRA model. The internal fire PRA model was developed consistent with NUREG/CR-6850 (Reference 10) and only utilizes methods previously accepted by the NRC. The NSPM risk management process ensures that the PRA model used in this application will reflect the as-built and as-operated plant prior to categorization.

The NRC approved PINGP's transition to a fire protection program in accordance with NFPA-805 in Reference 11. As a result of that approval, the PINGP RFOLs contain License Condition 2.C(4)(c)2, which specifies modifications that must be completed to fully implement the fire protection program. NSPM has reviewed the list of modifications required by the license condition and identified ten risk-significant modifications that are modeled in the current PINGP fire PRA but not yet installed in the plant. Therefore, completion of the modifications listed in Attachment 1, Table A-1, shall be a prerequisite for implementation. The modification list in Table A-1 is a modified version of the list submitted in Reference 12. It does not include those modifications that have already been installed or those that are not significant contributors to overall risk. Attachment 2 to this enclosure identifies the applicable fire PRA model.

3.2.3 Seismic Hazards

The PINGP categorization process will use the seismic margins analysis (SMA) performed for the IPEEE in response to Generic Letter (GL) 88-20, Supplement 4 (Reference 13), for evaluation of safety significance related to seismic hazards. No plant specific approaches were utilized in development of the SMA. The NEI 00-04 approved use of the SMA SSEL as a screening process results in the identification of all system functions and associated SSCs that are involved in the seismic margin success path as HSS. Since the analysis is being used as a screening tool, importance measures are not used to determine safety significance. The NEI 00-04 approach using the SSEL identifies credited equipment as HSS regardless of their capacity, frequency of challenge, or level of functional diversity.

An evaluation was performed of the as-built, as-operated plant against the SMA SSEL. The evaluation compared the as-built, as-operated plant to the plant configuration originally assessed by the SMA. Differences were reviewed to identify any potential impacts to the equipment credited on the SSEL. Appropriate changes to the credited equipment were identified and documented. This documentation is available for audit. The NSPM risk management program will ensure that future changes to the plant will be evaluated to determine their impact on the SMA and risk categorization process.

3.2.4 Other External Hazards

All other external hazards (i.e., not seismic or fire hazards) were screened from applicability to PINGP per a plant-specific evaluation in accordance with GL 88-20, Supplement 4, and updated to use the criteria in the ASME PRA Standard RA-Sa-2009 (Reference 14). Attachment 4 to this enclosure provides a summary of the other external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

As part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS. All remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

3.2.5 Low Power and Shutdown

Consistent with NEI 00-04, the PINGP categorization process will use the shutdown safety management plan described in NUMARC 91-06 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 defense in depth 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 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 of NEI 00-04 will be considered preliminary HSS.

3.2.6 PRA Maintenance and Updates

The NSPM risk management process ensures that the applicable PRA models used in this application continue to reflect the as-built and as-operated plant. The process delineates the responsibilities and guidelines for updating the PRA models, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, 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 process will assess the impact of these changes on the plant PRA model in a timely manner which is typically considered to be once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

In addition, NSPM will implement 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 upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.7 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA models used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the peer review process as discussed in Section 3.3 of this enclosure.

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

In the overall risk sensitivity studies, NSPM will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference 6. Consistent with the NEI 00-04 guidance, NSPM 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 before reaching the rate assumed in the sensitivity study.

The detailed process of identifying, characterizing, and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 (Reference 15) and Section 3.1.1 of EPRI TR-1016737 (Reference 16). The process in these references was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

The list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of this application. If the PINGP PRA model used a non-conservative treatment, or methods that are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application. Only those assumptions or sources of uncertainty that could significantly impact the risk calculations were considered key for this application.

Key PINGP PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 6 to this enclosure. The conclusion of this review is that sensitivity analyses will be performed, as necessary, to address PINGP PRA model specific assumptions or sources of uncertainty.

3.3 PRA REVIEW PROCESS RESULTS (10 CFR 50.69(b)(2)(iii))

The PRA models described 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" (Reference 17), consistent with NRC Regulatory Issue Summary (RIS) 2007-06 (Reference 18).

The internal events PRA model was subject to a full-scope peer review conducted in accordance with RG 1.200, Revision 2, in November 2010. The internal flooding portion of the PRA model was not available for peer review at that time. Subsequently, a focused scope peer review was performed on the internal flooding portions of the internal events PRA model against RG 1.200, Revision 2, in September 2012. An additional focused scope peer review of the internal events PRA model was conducted in May 2014 to evaluate the model changes made to address the incorporation of Flowserve N9000 Reactor Coolant Pump seals against RG 1.200, Revision 2.

The fire PRA model was subject to a full-scope peer review conducted in accordance with RG 1.200, Revision 2, in May 2012. Additionally, two focused scope peer reviews against RG 1.200, Revision 2, have been conducted to review changes to the fire PRA model. The first was conducted in November 2013 and reviewed the upgrade to the correlations used in determining hot gas layer temperatures. The second was conducted in December 2017 and reviewed the apportioning of the main control board fire frequency and the implementation of NUREG/CR-6850 thermal response time to cable damage.

Finding closure reviews were conducted on the internal events, including internal flooding, and fire PRA models in October 2017. Closed 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" (Reference 19), as accepted by the NRC in Reference 20. The results of this review have been documented and are available for NRC audit.

Attachment 3 to this enclosure provides a summary of the remaining findings and open items, including:

- Open findings and disposition of the PINGP PRA model peer reviews.
- Identification of and basis for any sensitivity analysis needed to address open findings.

The attachment identified above demonstrates that the PRA is of sufficient quality and level of detail to support the categorization process, and has been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required in 10 CFR 50.69(c)(1)(i).

3.4 RISK EVALUATIONS (10 CFR 50.69(b)(2)(iv))

The PINGP 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 10 CFR 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). 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.

4. REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The following NRC requirements and guidance documents are applicable to the proposed change.

- The regulations in 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors.”
- RG 1.201, “Guidelines for Categorizing Structures, Systems and Components in Nuclear Power Plants According to their Safety Significance”, Revision 1, May 2006 (Reference 5).
- RG 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 (Reference 21).
- RG 1.200, “An Approach for Determining Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2, March 2009 (Reference 17).

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

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, proposes to modify the licensing basis of the Prairie Island Nuclear Generating Plant 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.

NSPM 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 NRC special treatment requirements and to implement alternative treatments per the regulation. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensure 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 regulation. 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 regulation. 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, NSPM 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. 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 categorical 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. REFERENCES

1. NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline”, Revision 0, dated July 2005 (Agencywide Document Access and Management System (ADAMS) Accession No. ML052910035)

2. NSPM letter to NRC, “License Amendment Request: Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program”, dated March 15, 2018 (ADAMS Accession No. ML18074A308)

3. NSPM letter to NRC, “License Amendment Request to Revise License Condition Associated with Implementation of NFPA 805”, dated May 18, 2018 (ADAMS Accession No. ML18138A402)

4. NSPM letter to NRC, “Supplement to License Amendment Request to Revise License Condition Associated with Implementation of NFPA 805 (EPID L-2018-LLA-0147)”, dated July 10, 2018 (ADAMS Accession No. ML18191B265)

5. NRC Regulatory Guide 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance”, Revision 1, dated May 2006 (ADAMS Accession No. ML061090627)

6. NRC letter to Southern Nuclear Operating Company, “Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC Nos. ME9472 and ME9473)”, dated December 17, 2014 (ADAMS Accession No. ML14237A034)

7. NRC letter to Nuclear Management Company, “Review of Individual Plant Examination of External Events (IPEEE) Submittal (TAC Nos. M83663 and M83664)”, dated May 29, 2001

8. NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management”, dated December 1991 (ADAMS Accession No. ML14365A203)

9. NRC letter to Entergy Operations, Inc., “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)”, dated April 22, 2009 (ADAMS Accession No. ML090930246)

10. NRC NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities”, Volumes 1 and 2, dated September 2005 (ADAMS Accession Nos. ML15167A401 and ML15167A411)

11. NRC letter to NSPM, “Issuance of Amendments Re: Transition to NFPA-805 ‘Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants’ (CAC Nos. ME9734 and ME9735)”, dated August 8, 2017 (ADAMS Accession No. ML17163A027)

12. NSPM letter to NRC, “License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors – Response to Request for Additional Information (CAC Nos. ME9734 and ME9735)”, dated December 14, 2016 (ADAMS Accession No. ML16350A105)

13. NRC Generic Letter 88-20, Supplement 4, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f)”, dated June 28, 1991 (ADAMS Accession No. ML031150485)

14. ASME Standard ASME/ANS RA-Sa-2009, “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications”, dated February 2, 2009

15. NRC NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making”, dated March 2009 (ADAMS Accession No. ML090970525)

16. EPRI Report TR-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments”, dated December 2008

17. NRC Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2, dated March 2009 (ADAMS Accession No. ML090410014)

18. NRC Regulatory Issue Summary 2007-06, “Regulatory Guide 1.200 Implementation”, dated March 22, 2007 (ADAMS Accession No. ML070650428)

19. 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)”, dated February 21, 2017 (ADAMS Accession No. ML17086A450)

20. 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)”, dated May 3, 2017 (ADAMS Accession No. ML17079A427)

21. NRC 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, dated January 2018 (ADAMS Accession No. ML17317A256)

22. NSPM letter to NRC, “Prairie Island Nuclear Generating Plant, Units 1 and 2, Response to March 12, 2012, Request for Information Enclosure 2, Recommendation 2.1, Flooding, Required Response 2, Flood Hazard Reevaluation Report”, dated May 9, 2016 (ADAMS Accession No. ML16133A041)

23. NSPM letter to NRC, “Prairie Island Nuclear Generating Plant, Units 1 and 2, Response to March 12, 2012, Request for Information Enclosure 2, Recommendation 2.1, Flooding, Required Response 3, Focused Evaluation”, dated December 13, 2016 (ADAMS Accession No. ML16351A209)

Attachment 1: List of Categorization Prerequisites

NSPM 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.

- IDP member qualification requirements.
- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary HSS or LSS based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.1 of this enclosure). 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 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 and safety margin. Safety-related components that are categorized as preliminary LSS are evaluated for their role in providing defense in depth 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 risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to CDF and LERF and meets the acceptance guidelines of RG 1.174.
- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.
- Documentation requirements per Section 3.1.1 of this enclosure.

The risk-significant modifications reflected in the fire PRA model but not yet installed in the plant, as identified below in Table A-1, shall be completed.

Table A-1 – Risk Significant Modifications Related to Implementation of NFPA 805

Item*	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
1	High	1	A fire could damage Train B 12 Motor Driven Auxiliary Feedwater Pump (MDAFWP) and the control switches for the 11 Turbine Driven Auxiliary Feedwater Pump (TDAFWP) discharge valves (MV-32238 & MV-32239). Fire damage to CS-51003 could cause spurious closure of MV-32238 which would isolate the 11 TDAFWP flow to the credited 11 Steam Generator. Fire damage to control switch CS-51005 could prevent closing MV-32239 which could divert the 11 TDAFWP flow to the non-credited 12 Steam Generator. The NFPA 805 Nuclear Safety Performance Goal Criteria is not met for Decay Heat Removal.	Modify equipment in FA 31 to ensure that Train "A" equipment is available for fire safe shutdown. The controls and associated cables for the Unit 1 Train "A" AFWP discharge valves will be moved to Fire Area 32 so they are not damaged by a fire in Fire Area 31	Yes	Yes	The modifications proposed by Items 1 and 3 will reduce risk by modifying FAs 31 and 32 to ensure that each FA has either A-train or B-train related equipment unaffected by a fire. This will limit the number of fire scenarios that could damage both trains of equipment. Compensatory measures in accordance with the Current Fire Protection Licensing Basis are being maintained. Compensatory measures will continue to remain in effect after the NFPA 805 fire protection program becomes effective until this modification is complete.
3	High	1	A fire could damage the 11 TDAFWP (Train A) and the control switches for the 12 MDAFWP discharge valves (MV-32381 & MV-32382). Fire damage at the Train B	Modify equipment in FA 32 to ensure that Train "B" equipment is available for fire safe shutdown. The controls, MCC power supply, and associated	Yes	Yes	The modifications proposed by Items 1 and 3 will reduce risk by modifying FAs 31 and 32 to ensure that each FA has either A-train or B-train related equipment

*This table is a modified version of Table S-2, "Plant Modifications Committed" as submitted in Reference 12. It only includes those modifications that are not yet installed in the plant and significantly contribute to overall plant risk. The numbers listed in this table correspond to the item numbers listed in the table submitted in Reference 12.

Table A-1 – Risk Significant Modifications Related to Implementation of NFPA 805

Item*	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
3 cont.			Hot Shutdown Panel or MCC 1A2 could affect MV-32381 (12 MDAFWP to 11 SG) or MV-32382 (12 MDAFWP to12 SG). A fire at MCC 1A2 could affect MV-32027 (12 MDAFWP suction from Cooling Water), MV-32335 (12 MDAFWP suction from CST), MV-32381 (12 MDAFWP to 11 SG) and MV-32382 (12 MDAFWP to 12 SG). The NFPA 805 Nuclear Safety Performance Goal Criteria is not met for Decay Heat Removal.	cables for the Unit 1 “B” AFWP discharge and suction valves will be moved out of Fire Area 32 so they are not damaged by a fire in Fire Area 32. The cables going to Unit 1 Train “B” AFW discharge valves (MV-32381 and MV-32382) will be modified so that the MOV will not spuriously close due to a fire in Fire Area 32.			unaffected by a fire. This will limit the number of fire scenarios that could damage both trains of equipment. Compensatory measures in accordance with the Current Fire Protection Licensing Basis are being maintained. Compensatory measures will continue to remain in effect after the NFPA 805 fire protection program becomes effective until this modification is complete.
10	High	1, 2	A fire could damage DC control cables for 4 KV breakers which could cause the tripping control power fuses to clear which would prevent the breaker from tripping on over-current. The fire could then damage the 4 KV power cable, but since the breaker can't trip, the cable would be subjected to an over-current condition up to the full fault current available to the bus. If the cable is not sized large enough to carry this amount of current, the cable could	Modify 4160 volt switchgear control circuits so that faults on the control cables will not prevent the over-current trip relay from protecting the cable from over-current conditions that could lead to cable damage and secondary fires or loss of bus coordination.	Yes	Yes	The FPRA assumes coordination of credited buses. This modification ensures there are no secondary fires. Compensatory measures in accordance with the Current Fire Protection Licensing Basis are being maintained. Compensatory measures will continue to remain in effect after the NFPA 805 fire protection program becomes effective until this

Table A-1 – Risk Significant Modifications Related to Implementation of NFPA 805

Item*	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
10 cont.			be damaged and start a fire in other fire areas where it is routed. Affected Breakers: BKR 15-1, BKR 15-4, BKR 15 5, BKR 15-9, BKR 16-1, BKR 16-5, BKR 16-6, BKR 16-7, BKR 16-10, BKR 16-12, BKR 25-7, BKR 25-8, BKR 25-9, BKR 25-10,BKR 25-13, BKR 26-5, BKR 26-9, BKR 26-11				modification is complete.
14	Medium	1, 2	A fire in FA 13/18 could damage cables causing multiple spurious operations that could damage D1 Emergency Diesel Generator. If fire induced cable damage caused multiple spurious operations that caused D1 (034-011) to spuriously start with no cooling water (11 MDCLP MTR 13-8, 12 DDCLP 145-392, 21 MDCLP MTR 23-4, 22 DDCLP 245-392) then the EDG could be damaged.	Modify control circuits for the Diesel Driven Cooling Water Pump to eliminate the current required recovery action of sending an operator to the D1 Room and Screenhouse.	Yes	No	This modification will reduce risk by simplifying restoration of Cooling Water to provide cooling to D1 Emergency Diesel Generator and a backup water supply to the Aux Feedwater Pumps.
23	Medium	2	A fire in Bus 27 room (Fire Area 127) could damage DC control power to Bus 25 or Bus 26.	Install fuses to provide coordination so that a fire in the Bus 27 room will not affect DC control power to Bus 25 or Bus 26.	Yes	Yes	The Fire PRA assumes proper coordination of these power supplies

Table A-1 – Risk Significant Modifications Related to Implementation of NFPA 805

Item*	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
24	High	1, 2	A fire in the Bus 15 (Fire Area 81) or Bus 16 (Fire Area 20) room could damage the cables and bus duct that supply off-site power (CT11 and 1R transformers) to Bus 15 and Bus 16 due to common power supply. The redundant diesel generator remains unaffected by a fire to re-power the unaffected 4 kv safeguards bus (Bus 15 or Bus 16), but risk is higher than desired. Unit 2 is similar to Unit 1. A fire in Fire Area 117, BUS 25 or Fire Area 118, BUS 26, could damage cables for both off-site power sources (2RY and CT12 transformer). The redundant Emergency Diesel Generator (D5/D6) remains unaffected by a fire, but the risk is higher than desired.	Provide fuse/breaker coordination for the CT11 supply to Bus 15 and Bus 16 so that the CT11 source remains available to Bus 15 if a fire damages Bus 16 or to Bus 16 if a fire damages Bus 15. Provide fuse/breaker coordination for the CT12 supply to Bus 25 and Bus 26 so that the CT12 source remains available to Bus 25 if a fire damages Bus 26 or to Bus 26 if a fire damages Bus 25. Modify associated control cables (1CS-1, 1CS-2, 1CS-3, and 1CS-4) so the CT11/CT12 source remains available for the opposite Bus room.	Yes	No	The proposed modification will reduce risk by ensuring one off-site power source to the safeguards 4 kV Bus remains unaffected by a fire in the event of a fire in the opposite train safeguards 4 kV Bus room.
25	Medium	1	A fire in Fire Area 32 could damage cables required to open MV-32077 and MV-32078 to provide recirculation from Sump B.	Re-power MV-32078 from an MCC that is not located in Fire Area 32 to re-gain the ability to recirculate water from Sump B.	Yes	No	This will reduce risk by ensuring a fire in FA 32 does not damage the ability to recirculate water from Sump B.

Table A-1 – Risk Significant Modifications Related to Implementation of NFPA 805

Item*	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
34	Medium	1, 2	A fire in FA 13, 18, 32 or 58 could damage cables and cause spurious closure of the Emergency Diesel Generator output breaker. This could cause a lockout of the 4kv safeguards Bus which powers one train of safeguards equipment.	Modify cables to prevent spurious closure from risk significant fire initiators.	Yes	No	This will reduce risk by making modifications to reduce the number of fire scenarios that could cause fire damage to a 4kV safeguards bus.
35	Medium	1	A fire in Fire Area 32 or 58 could damage cables which support operation of the 1RY offsite power sources to BUS 15 (BKR-15-3) and BUS 16 (BKR-16-2).	Modify cable 1C-332 from fire damage in Fire Area 32 and 58 to ensure BUS 16 can be powered from the 1RY transformer.	Yes	Yes	The proposed modification will reduce risk because it will reroute cables associated with the opposite train of equipment to another FA. This will limit the number of fire scenarios that could damage both trains of equipment. Compensatory measures in accordance with the Current Fire Protection Licensing Basis are being maintained. Compensatory measures will continue to remain in effect after the NFPA 805 fire protection program becomes effective until this modification is complete.

Table A-1 – Risk Significant Modifications Related to Implementation of NFPA 805

Item*	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
41	Medium	1, 2	A fire in Fire Area 13 or 18 could damage cables that could over-torque motor operated valves; MV-32006, MV-32010, MV-32021, MV-32022, MV-32238, and MV-32246 which are credited in the Fire PRA to be locally operated to perform Recovery Actions.	Re-wire the torque and limit switches so fire induced damage to cables in FA 13 and 18 cannot bypass the torque and limit switches and subsequently over-torque the MOV.	Yes	Yes	The proposed modification will allow the valve to be locally operated to credit this recovery action in the PRA. Compensatory measures in accordance with the current Fire Protection Licensing Basis are being maintained. Compensatory measures will continue to remain in effect after the NFPA 805 fire protection program becomes effective until this modification is complete.

Attachment 2: Description of PRA Models Used in Categorization

Model	Baseline CDF	Baseline LERF	Comments
Internal Events PRA Model, including Internal Flooding, Revision 5.3, dated November 30, 2017. Full scope peer review (excluding Internal Flooding), against RG 1.200, Revision 2, in November 2010. Focused scope peer review of the Internal Flooding portion of the Internal Events PRA Model in September 2012. Additional focused scope peer review in May 2014.	1.28E-05 (Unit 1) 1.25E-05 (Unit 2)	2.15E-07 (Unit 1) 1.86E-07 (Unit 2)	This model represents the current Internal Events, including internal flooding, PRA Model of Record.
Fire PRA Model, Revision 5.3-APP1, dated June 25, 2018. Full scope peer review against RG 1.200, Revision 2, in May 2012. Focused scope peer reviews in November 2013 and December 2017.	6.60E-05 (Unit 1) 6.59E-05 (Unit 2)	9.60E-07 (Unit 1) 9.26E-07 (Unit 2)	This model is an application-specific fire PRA model that was created for the LAR to change the list of NFPA 805 modifications.

Attachment 3: Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
	Internal Events PRA Model Open Findings		
6-13 (QU-C2)	***From 2010 Full Scope PRA Peer Review:*** PRA-PI-QU, Section 4.3.4 discusses the process used to adjust multiple HFEs [human failure events] using HRA [human reliability analysis] calculator, EXCEL spreadsheets and utility programs. PRA-PI-HRA, section 3.4.2 and Attachment E address the dependent HFE analysis and resultant values which were used in the final quantification. This should be referenced in the QU notebook. These are included in Appendix F in the HEPCombos.txt file. There is no discussion of the details of the adjustments made to the dependent HFEs to justify that the combination HFEs and in many cases have extremely low values well below 1.0E-05. These low values and their impact on CDF/LERF should be justified and evaluated.	A minimum joint human error probability (HEP) of 1.0E-07 was applied to joint dependent HEPs in the dependency analysis. The HRA Notebook and Quantification Notebook were revised to discuss the minimum joint HEP limit. The F&O finding closure team reviewed the resolution and determined that the finding should remain open pending the inclusion of additional justification for the 1.0E-07 value that was used. Subsequent to the F&O closure review, a new internal events PRA model (Revision 5.3) was approved that incorporated additional changes to address this finding. This included applying a minimum joint HEP of 1.0E-06 and adding a justification of risk-significant joint HEP combinations less than 1.0E-05 in the QU Notebook.	The open issue identified by the F&O closure team has been addressed and is included in the internal events PRA model that will be used for 50.69 categorization. Therefore, this issue will not impact 50.69 categorization.
SY-A17-01 (SY-A17)	***From 2014 Focused Scope Peer Review for Flowserve N-9000 Reactor Coolant Pump Seal Loss of Coolant Accident Modeling:*** Subsection 1.8.1 of AC System notebook, “PRA-PI-SY-AC, Rev. 2.1a” indicates safeguards 4kV buses do not result in RCP [reactor coolant pump] trip. Failure in both 4Kv buses (bus 15 and 16)	A sensitivity analysis was performed that demonstrated failure to trip the RCP with a loss of all 4 kV safety buses would result in a negligible (less than 1.0E-08 per year) increase in CDF and represents a negligible source of uncertainty for the base PRA model. The F&O finding closure review	It is expected that this F&O finding can be considered to be closed once the underlying RCP seal model has been approved. As demonstrated by the

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
SY-A17-01 (continued)	which is a cause of 1AC requires RCP trip to prevent RCP seal failure, but 1N9-SBO gate does not include the operator action. Cause(s) of loss of 1AC which do not result in RCP trip requires RCP trip within 2 hours to prevent RCP LOCA.	team concurred with this assessment. However, since the N-9000 RCP seal model must obtain NRC review and approval, the closure review team determined that the F&O should remain open until the underlying RCP seal model is approved.	sensitivity study, this issue will not impact 50.69 categorization.
Fire PRA Model Open Findings			
ES-C1-01 (ES-C1) (CS-A1)	***From 2012 Full Scope Fire PRA Peer Review:*** HFEs created specifically for Fire Scenarios (identified in FPRA-PI-FHRA) and their credited instrumentation have not been included in the Equipment Selection documentation. Related to this, not all instrumentation for those HFEs are cable selected. Standard requires that all HFEs and associated instrumentation required for FPRA [fire PRA] are identified in the Equipment Selection task (and it is noted that Equipment Selection is an iterative process). Also, associated instrumentation requires cable selection if cable routing is not available. Include all HFEs in the Equipment Selection documentation. Cable selection will be required on instrumentation if not already available. It has been noted that there is a revision plan in place to address instrumentation currently not cable selected in the HRA.	The F&O finding closure team reviewed the resolution and determined that the finding should remain open pending the inclusion of additional documentation for screening additional instrumentation for HFEs. Subsequent to the F&O closure review, a new fire PRA model (Revision 5.3) was approved that incorporated documentation justifying the exclusion of instrumentation for HFEs. Criteria have been provided for a minimum level of redundancy and diversity to meet the intent of the ASME PRA standard with respect to determining if instrumentation needs to be modeled. Each instrument documented is evaluated to determine the dependencies (i.e. AC bus or DC power panel) for the instrument in question. The review looks at the overall number of instruments and power supplies to determine if sufficient diversity and redundancy is present to screen out modeling of the	The open issue identified by the F&O closure team has been addressed and is included in the Fire PRA model that will be used for 50.69 categorization. Therefore, this issue will not impact 50.69 categorization.

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
ES-C1-01 (continued)		instrument. Any HFE that has insufficient diversity or redundancy in instrumentation is either assumed failed or the instrumentation is explicitly modeled.	
CS-A10-01 (CS-A10)	***From 2012 Full Scope Fire PRA Peer Review:*** Open item No. 1 on Page 17 of 17 of FPRA-PI-CS, Revision C, states that "in order to fully comply with Capability Category II, cables that are routed through fire compartments 2A, 41B-1, 46A, 58A, 58B, 58C, 58D, 76A, 78E, 86, 94A, 94B, 94C, 94D, 94E, and 94F need to be identified. This will be accomplished by identifying routing on electrical drawings, and with walkdowns performed as needed." Since this has not yet been completed, and the methodology not specified, this SR is provisionally met. Completeness of Fire PRA. Route cables through the listed fire compartments, and perform walkdowns to confirm accuracy of the routing. Utilize EPM Division Procedure EPM-DP-EP-005, Revision 1-Post-Fire Safe Shutdown Cable Routing and Component Location, and EPM Division Procedure EPM-DP-EP-004, Revision 2-Post Fire Safe Shutdown Cable Identification-February 2011.	The F&O finding closure team reviewed the resolution and determined that the finding should remain open pending the identification of cable routing in fire compartments that are a subset of larger fire areas. Subsequent to the F&O closure review, a new fire PRA model (Revision 5.3) was approved that incorporated additional plant walkdowns and drawing reviews that were performed to refine cable routing reflected in the PINGP fire PRA model to resolve this issue. The cable to fire compartment table was updated as well as the documentation of the issue. The cables in these fire compartments are now mapped to fire compartment; not fire area.	The open issue identified by the F&O closure team has been addressed and is included in the Fire PRA model that will be used for 50.69 categorization. Therefore, this issue will not impact 50.69 categorization.
FSS-D7-01 (FSS-D7) (PRM-A2)	***From 2012 Full Scope Fire PRA Peer Review:*** The Non-Suppression probability (NSP) for the deluge system appears to be calculated incorrectly in several places in the FPRA-PI-SS report. In particular, in Table 20 on page 25 of 34 in report FPRA-PI-SS, Revision A, the unreliability of the detection system required to activate the deluge valve is not factored into the calculation. Additionally, It appears that the event tree in	The F&O finding closure team reviewed the resolution and determined that the finding should remain open pending update of the unreliability used for the pre-action suppression system. Subsequent to the F&O closure review, a new fire PRA model (Revision 5.3) was approved that	The open issue identified by the F&O closure team has been addressed and is included in the Fire PRA model that will be used for 50.69 categorization. Therefore, this issue

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
FSS-D7-01 (continued)	section P.1.3 and Figure P-1of NUREG/CR-6850 has been solved incorrectly. It also appears that the bullet titled "Pr(failure auto det):" and the bullet titled "Pr(failure auto supp):"on page P-6 of NUREG/CR-6850v2 is being interpreted incorrectly. This impacts the accuracy of the FPRA. Revise the calculation method on page 25-26 of 34 in report FPRA-PI-SS, Revision A and document its accuracy.	incorporated the updated unreliability for the pre-action suppression system. The process used to calculate the fire ignition frequencies for structural steel fire scenarios was re-performed. The non-suppression probability for the deluge systems was revised to correct the identified errors. All frequencies reported in the Structural Steel documentation now match the Fire PRA model.	will not impact 50.69 categorization.
FSS-F1-01 (FSS-F1)	***From 2012 Full Scope Fire PRA Peer Review:*** During walkdowns it was determined that hydrogen lines in the turbine building have the capability of initiating a fire that would directly impinge on the building structural steel. In many plants, the use of fork trucks is not allowed in the area of exposed hydrogen pipelines. However, based on the walkdown performed, fork truck use is allowed in the area of exposed hydrogen pipelines at PINGP. The potential for a fork lift inadvertently impacting an exposed hydrogen line resulting in a direct hydrogen flame on exposed structural steel in the Turbine Building needs to be addressed. Additionally, the Hydrogen Storage Tanks outside the Turbine Building are stored in a physical configuration such that a fire in the storage tank area that impacts the valves and ends of the tanks could result in the tanks becoming missiles and punching through the concrete wall separating them from the Turbine Building. The potential for and impact of this scenario needs to be addressed. Evaluate and document the missing hydrogen scenarios.	The F&O finding closure team reviewed the resolution and determined that the finding should remain open pending assessment of whether this potential plant-specific issue should be considered or modeled. Subsequent to the F&O closure review, a new fire PRA model (Revision 5.3) was approved that incorporated the hydrogen fire scenario in the turbine building. Ignition frequency Bin 34 turbine generator hydrogen fire scenarios were added to the turbine building fire scenarios. These scenarios include the probability of hydrogen fires in the Turbine Building regardless of how the fire is initiated (including due to fork truck movements). Hydrogen tank generated missiles were determined to have an insignificant probability and impact to the plant’s fire PRA.	The open issue identified by the F&O closure team has been addressed and is included in the Fire PRA model that will be used for 50.69 categorization. Therefore, this issue will not impact 50.69 categorization.

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
IGN-A7-01 (IGN-A7) (IGN-B3)	***From 2012 Full Scope Fire PRA Peer Review:*** Bin 29, Yard Transformers (Others) is not filled. Additionally, Bin 15.1 Electrical Cabinets Non-HEAF [high energy arc fault] may not be filled correctly. Fire Compartment 20 (Switchgear Room) has more HEAF electrical cabinet counts (13) than non-HEAF counts (7). This suggests that electrical cabinets with the potential to have a HEAF event are not assigned a non-HEAF fire event. Thus the failure mode of electrical cabinet fire may not be counted and underestimate risk in compartments with the potential for HEAF. This may inadvertently skew risk away from important areas such as transformer yard and switchgear rooms. Fill Bin 29 similar to how bins 27 and 28 are filled. If Electrical Cabinet can have a HEAF failure mode, ensure that both Bins 15.1 and 15.2 are filled. Also, for Bin 21 (Pumps), believe that the split fractions in [NUREG/CR] 6850 electrical/oil are still valid.	The F&O finding closure team reviewed the resolution and determined that the finding should remain open pending documentation of ignition source counting. Subsequent to the F&O closure review, a new fire PRA model (Revision 5.3) was approved that updated high voltage and low voltage HEAF ignition source counts. Ignition source counts were revisited for bins 15, 16, 21, and 29 for all fire compartments. During the internal review, it was discovered that some components are being conservatively counted as both low and high voltage HEAF sources. This double counting of ignition sources was corrected using the NUREG/CR-6850, Supplement 1 for binning/counting electrical cabinets based on voltage rating.	The open issue identified by the F&O closure team has been addressed and is included in the Fire PRA model that will be used for 50.69 categorization. Therefore, this issue will not impact 50.69 categorization.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	PS4	The nearest major airport is Minneapolis-St. Paul International (MSP) which is located approximately 30 miles from the site. There only three airports within 10 miles of the plant and they have been screened out from further consideration or analyzed to be so small as to not pose a hazard for PINGP. Of the airports greater than 10 statute miles from the PINGP site, they have either been screened out or demonstrated to not pose a hazard for PINGP. A reevaluation of external events demonstrated that the risk due to this hazard of aircraft-induced radiological consequences is less than 1.0E-07 per year. If it is conservatively assumed that LERF is a surrogate for the radiological consequence and CDF is typically an order of magnitude greater for PWRs, this would imply that CDF is less than 1.0E-06 per year.
Avalanche	Y	C3	The topography surrounding PINGP precludes the possibility of a snow avalanche.
Biological Event	Y	C5	Actions committed to and completed by PINGP in response to GL 89-13 (Service Water System Problems Affecting Safety-Related Equipment) provide on-going control of biological hazards. These controls are described in PINGP procedure H21, “Generic Letter 89-13 Implementing Program”. Additionally, actions taken in response to INPO SOER 07-2 (Intake Structure Blockage) provide an additional layer of biological hazard management. The hazard is slow to develop and can be identified by monitoring and managed through standard maintenance processes.
Coastal Erosion	Y	C3	The mid-western inland location of PINGP precludes the possibility of coastal erosion.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Drought	Y	C1 C5	These effects would take place slowly allowing time for orderly plant reductions including shutdowns. Also, the design of the cooling water supply is such that adequate water will be delivered into the plant under any condition.
External Flooding	Y	PS1	The external flooding hazard at PINGP was recently updated as a result of the post-Fukushima 50.54(f) Request for Information and the flood hazard reevaluation report (FHRR) was submitted to the NRC for review on May 9, 2016 (Reference 22). The results indicate that flooding from rivers and streams are bounded by the current licensing basis and do not pose a challenge to the plant. Flooding from local intense precipitation (LIP) was also evaluated and the focused evaluation (Reference 23) affirms that during LIP events the site has effective flood protection through the determination of Available Physical Margin and the reliability of protection features and will not challenge any safety functions at PINGP.
Extreme Wind or Tornado	Y	PS2 PS4	Wind damage is bounded by tornadoes, and the tornado wind speed corresponding to the 1.0E-07 per year exceedance frequency is less than the PINGP design value. Therefore, damage due to the forces associated with extreme winds or tornadoes, including missiles, can be screened..
Fog	Y	C4	The principal effects of such events would be to indirectly cause a loss of offsite power due to the occurrence of other hazards, such as highway accidents, aircraft landing and take-off accidents, and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Forest or Range Fire	Y	C1 C3 C4	The site landscaping and lack of forestation nearby prevent such fires from posing a threat to PINGP. Furthermore, the principal effects of such events would be to cause a loss of offsite power, which is addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP, and smoke and gases entering the control room. If the latter were to occur, operators would have sufficient time to take action, such as donning protective air masks within the control room if the concentration of smoke begins to increase.
Frost	Y	C4	The effects of frost are bounded by snow and ice. The principal effect of such events would be to cause a loss of offsite power and is addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP.
Hail	Y	C1 C4	Hail is bounded by other events, such as tornado missiles, for which the plant is designed. The principal effects of such events would be to cause a loss of offsite power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP.
High Summer Temperature	Y	C1 C5	The principal effects of such events would result in elevated river temperatures which are monitored by station personnel. The design maximum temperature for the Cooling Water System is 85^{o}F and the average monthly temperature at St. Paul, which is typically 2 to 3 degrees higher than at the site, typically does not approach that value. Safeguards components are operable with Cooling Water inlet temperature up to 95^{o}F. The climatology at PINGP is such that extreme heat would have an insignificant effect on plant operations.
High Tide, Lake Level, or River Stage	Y	C3 C4	High tide or lake level are not applicable to the site because of location. Impact of High River Stage is included as an impact in the external flooding analysis.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Hurricane	Y	C3	The mid-western location of PINGP precludes the possibility of a hurricane. Additionally, hurricanes would be covered under Extreme Winds and Tornados and Local Intense Precipitation.
Ice Cover	Y	C1 C4	Plant piping and equipment located outside of plant buildings are protected by heat tracing to prevent adverse effects from severe cold. Furthermore, the capacity reduction of the ultimate heat sink (UHS) due to extreme cold would be a slow process that would allow plant operators sufficient time to take proper actions, such as reducing plant power output level or achieving safe shutdown. The principal effects of such events would be to cause a loss of offsite power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP.
Industrial or Military Facility Accident	Y	C3 C4	There are no military facilities within five miles of the plant (the closest is the Red Wing National Guard Armory, ~7.5 miles away). The hazards associated with an industrial facility accident are screened elsewhere in this table (e.g., transportation accident, pipeline accident).
Internal Flooding	N	N/A	The PINGP internal events PRA addresses risk from internal flooding events.
Internal Fire	N	N/A	The PINGP NFPA-805 fire PRA addresses risk from internal fire events.
Landslide	Y	C3	In the immediate vicinity of the PINGP, there are no steep hills. Therefore, it is not applicable to the site because of topography.
Lightning	Y	C1 C4	Lightning strikes can result in losses of offsite power. This is incorporated into the PINGP internal events PRA model through the incorporation of generic and plant-specific data.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Low Lake Level or River Stage	Y	C1 C5	PINGP uses water from the reservoir upstream of Lock and Dam Number 3 on the Mississippi River for UHS. An accident at the dam concurrent with normal river flow would provide a level of water 10.9 feet deep at the circulating water intake. The design of the cooling water system is such that it will deliver adequate water to the plant under any condition. Other reductions in river level would take place slowly over time allowing for orderly plant reductions, including shutdowns.
Low Winter Temperature	Y	C1 C4 C5	Plant piping and equipment located outside of plant buildings are protected by heat tracing to prevent adverse effects from severe cold. The principal effects of such events would be to cause a loss of offsite power. The effects of weather-related losses of offsite power are included in the PINGP PRA models. These effects would take place slowly allowing time for orderly plant power reductions, including shutdowns.
Meteorite or Satellite Impact	Y	C2	The frequency of a meteorite or satellite strike is judged to be very low such that the risk impact from such events is insignificant.
Pipeline Accident	Y	C1	A 4-inch natural gas supply line terminates outside the northwest corner of the Owner Controlled Area for PINGP. The effects on plant structures due to a blast release due to a Vapor Cloud Explosion are bounded by tornado loadings.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Release of Chemicals in Onsite Storage	Y	C3 C4 PS1	No chlorine gas is stored on-site. The newly installed natural gas supply line was also evaluated for its effect on control room habitability and diesel generator operation where it was determined that natural gas concentrations resulting from a leak would neither challenge the environment of the control room nor would it challenge the operability of the safety-related diesel generators. In addition, it was determined there was no asphyxiation hazard posed by the rupture of the largest nitrogen tank onsite (3000 gallons). Chemical hazards stored and transported in the vicinity of the plant are analyzed in conformance with the guidance set forth by RG 1.78 and NUREG-0570.
River Diversion	Y	C1	In the event of Mississippi River diversion, the water in the intake canal and the emergency intake line provide enough cooling water to enable safe shutdown of both units.
Sand or Dust Storm	Y	C1 C3 C4	The frequency of a loss of offsite power accounts for severe weather, including sand or dust storms.
Seiche	Y	C3	The PINGP site is located on the Mississippi River. Gantenbein Lake and Larson Lake are both more than 1 kilometer from the site and Sturgeon Lake is approximately 1/2 kilometer from the site. Therefore, no large body of water is close enough for the site to be susceptible to a seiche.
Seismic Activity	N	N/A	See information in Section 3.2.3 of this enclosure.
Snow	Y	C1 C4	The average snowfall per year in Red Wing, Minnesota is 32 inches. The maximum recorded snowfall from a single storm in Minnesota occurred near Finland and measured 46.5 inches. One inch of snowfall weighs approximately 1 psf, which means the estimated weight from a postulated maximum snowfall would be 46.5 psf. The design basis roof live load is 50 psf, which is within the design basis.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Soil Shrink-Swell Consolidation	Y	C3	The soil at PINGP is sandy alluvium. Due to the very permeable nature of the granular soils at the site, the soil is resistant to shrink-swell.
Storm Surge	Y	C4	The potential storm surge from Sturgeon Lake was evaluated in the FHRR and determined to be bounded by External Flooding.
Toxic Gas	Y	C4	The hazards associated with toxic gas are screened elsewhere in this table (e.g., Industrial and Military Facility Accidents, Release of Chemicals in Onsite Storage)
Transportation Accident	Y	C1 C3 C4	Land Transportation – Based on the proximity of the nearest major roadways, truck explosions pose no danger to PINGP. Rail Transportation – Based on the proximity of the nearest commercial railroad line, potential impacts are covered by Extreme Wind or Tornado as well as Release of Chemicals in Onsite Storage. Water Transportation – PINGP is located along the Mississippi River, the main channel of which is ~0.5 miles from the site. Based on that proximity, potential impacts are covered by Extreme Wind or Tornado as well as Release of Chemicals in Onsite Storage.
Tsunami	Y	C3	The mid-western location of PINGP precludes the possibility of a tsunami.
Turbine-Generated Missiles	Y	PS4	The probabilistic analysis performed for postulated failures of turbines in PINGP has shown that the overall probability of turbine missile damage is less than the NRC accepted value of 1.0E-07 per year.
Volcanic Activity	Y	C3	Not applicable to PINGP as the site is not close to any active volcanoes.
Waves	Y	C4	The potential impacts of waves were evaluated in the FHRR and determined to be bounded by External Flooding.
Note a – See Attachment 5 for descriptions of the screening criteria.			

Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion		Source	Comments
Initial Preliminary Screening	C1	Event damage potential is less than events for which plant is designed	NUREG/CR-2300 ASME/ANS PRA Standard	
	C2	Event has lower mean frequency and no worse consequences than other events analyzed	NUREG/CR-2300 ASME/ANS PRA Standard	
	C3	Event cannot occur close enough to the plant to affect it	NUREG/CR-2300 ASME/ANS PRA Standard	
	C4	Event is included in the definition of another event	NUREG/CR-2300 ASME/ANS PRA Standard	Not used to screen. Used only to include within another event.
	C5	Event develops slowly allowing adequate time to eliminate or mitigate the threat	ASME/ANS PRA Standard	
Progressive Screening	PS1	Design basis hazard cannot cause a core damage accident	ASME/ANS PRA Standard	
	PS2	Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan	NUREG-1407 ASME/ANS PRA Standard	
	PS3	Design basis event mean frequency is < 1.0E-05 per year and the mean conditional core damage probability is < 0.1	NUREG-1407 ASME/ANS PRA Standard	
	PS4	Bounding mean CDF is < 1.0E-06 per year	NUREG-1407 ASME/ANS PRA Standard	
Detailed PRA		Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 ASME/ANS PRA Standard	

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

Assumption/ Uncertainty	Discussion	Disposition
Internal Events PRA Model Key Assumptions/Sources of Uncertainty		
Screening exclusion of some components and failure modes in the main feedwater (MFW) system model.	Certain components/ failure modes were not modeled for MFW because they were of lower probability of occurrence than other similar failure modes already in the system models.	Categorization evaluations for this specific system will perform sensitivity studies, as necessary, to confirm that the screening exclusions do not impact the results of the individual component categorizations.
Not all flow diversion pathways in the residual heat removal (RHR) system were modeled.	The flow diversion paths are assumed to result in insignificant flow diversion and were not specifically modeled for RHR system.	Categorization evaluations for this specific system will perform sensitivity studies, as necessary, to confirm that the screening exclusions do not impact the results of the individual component categorizations.
Potential recovery actions not modeled for the RHR system.	Operator response to RHR heat exchanger bypass valve failure to close is not modeled for the RHR Shutdown Cooling Operation.	Categorization evaluations for the RHR system will include sensitivity studies, as necessary, to evaluate the impact of this non-credited HFEs on individual component categorizations.
Potential recovery actions not modeled for the Circulating Water System	Failure to trip the circulating water pumps on low water level in the intake bay was not modeled. This is only an issue on a Loss of Circulating Water initiating event or a Non-Loop Initiating event with subsequent random failure of intake screenhouse traveling screens bays.	Categorization evaluations for the Circulating Water System and Cooling Water Systems will include sensitivity studies, as necessary, to evaluate the impact of this non-credited HFE on individual component categorizations.

Assumption/ Uncertainty	Discussion	Disposition
Basis for Human Error Probabilities (EPRI-identified generic source of modeling uncertainty)	HRA is a continually evolving discipline. The HEPs were obtained using the current EPRI HRA calculator consistent with the Fire HRA Methodology described in NUREG-1921. The Internal Events human error probabilities were obtained using guidance from NUREG/CR-1278 and NUREG/CR-4772. Given the methodologies used, the impact of any remaining uncertainties is expected to be small.	The PINGP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of uncertainty. As directed by NEI 00-04, human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values as part of the required 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and, therefore, the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.
Thermally-induced Steam Generator Tube Rupture (TI-SGTR) (EPRI-identified generic source of modeling uncertainty)	The results of the generic event tree quantification reported in WCAP-16341 are applicable to PINGP. The TI-SGTR impact on LERF is low. However, due to significance to LERF results for non-Station Blackout (SBO) sequences, TI-SGTR impacts may be a source of uncertainty for some non-SBO sequences.	Since the treatment of TI-SGTR is primarily a phenomenological uncertainty for LERF, this is not expected to have any impact on the 50.69 categorization for PINGP plant systems.

Assumption/ Uncertainty	Discussion	Disposition
Interfacing system loss of coolant accident (ISLOCA) (low pressure RHR piping outside containment assumed failed if exposed to reactor coolant system (RCS) pressure)	The plant-specific ISLOCA screening and modeling for PINGP is based on the guidelines in Nuclear Safety Analysis Center (NSAC) 154, NUREG/CR-5102, and NUREG/CR-5744. The ISLOCA fault tree models are quantified dynamically within the single-top model fault tree models for CDF and LERF for each unit. PINGP piping is designed with significant margin to failure such that it is possible that the low pressure piping (600 psig design pressure) of concern for this analysis may be able to withstand exposure to RCS pressure. However, the PRA assumes the piping will fail if exposed to RCS pressure since no data exists to support the function of the RHR pump discharge check valves under RCS pressure conditions and their failure was assumed which leads to the assumed failure of the RHR pump seals and the subsequent release of RCS fluid outside of containment.	PINGP piping is designed with margin to failure such that it is possible that the low pressure piping (600 psig design pressure) of concern for this analysis may be able to withstand exposure to RCS pressure. However, the PRA assumes the piping will fail if exposed to RCS pressure. The current approach used will result in conservative evaluations. This conservativism would tend to result in additional SSCs being categorized as HSS in the 50.69 categorization process. The potential for masking of other HSS SSCs as a result of this conservative treatment can be assessed on a case-by-case basis as needed.
Fire PRA Model Key Assumptions/Sources of Uncertainty		
Fire PRA Cable Selection	Fire PRA Cable Selection is performed with a level of detail commensurate with the risk significance of the component. Higher risk components have higher fidelity of cable selection. The selection of cables to be considered in the analysis identified using industry guidance documents such as NUREG/CR-6850.	Based on the discussion of sources of uncertainty it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this is not a source of uncertainty for the 50.69 categorization.

Assumption/ Uncertainty	Discussion	Disposition
Fire Ignition Frequencies	Ignition source counting is an area with inherent uncertainty; however, the results are not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the frequency values from NUREG/CR-6850, Supplement 1, which result in uncertainty due to variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates, based on limited fire events and fire test data. The conservatism in the ignition frequency data, which is also linked to conservatism in non-suppression probability data specified in NUREG/CR-6850, appears to introduce a significant conservatism. The current fire PRA utilizes the bin frequencies from NUREG/CR-2169, which represents the most current approved bin frequencies. As such, some of the inherent conservatism associated with bin frequencies from NUREG/CR-6850 has been removed.	The current approach employed is an industry-accepted method for ignition source counting. If new industry accepted methods are developed for counting ignition sources, then they will be incorporated into the fire PRA model. Since current industry data is being used in accordance with accepted methods, this is not judged to create a source of uncertainty that would impact the 50.69 categorization.

Assumption/ Uncertainty	Discussion	Disposition
Detailed Fire Modeling	The approach taken for the Detailed Fire Modeling was: 1) the use of generic fire modeling treatments in lieu of conservative scoping analysis techniques, and 2) limited detailed fire modeling was performed to refine the scenarios developed using the generic fire modeling solutions. The employment of generic fire modeling solutions did not introduce any significant conservatism. Detailed fire modeling was only applied where the reduction in conservatism was likely to have a measurable impact. The NUREG/CR-6850 heat release rates introduce significant conservatism given the limited fire test data available to define the heat release rates and the associated fire development timeline.	Detailed fire modeling was performed only on those scenarios which otherwise would have been notable risk contributors and only where removal of conservatism in the generic fire modeling solution was likely to provide benefit either via a smaller zone of influence or to allow credit for automatic suppression. Fire modeling was used to evaluate the time to abandonment for control room fire scenarios for a range of fire heat release rates. The analysis methodology conservatism is primarily associated with conservatism in the heat release rates and manual non-suppression probability data specified in NUREG-2178. Based on the use of an industry accepted method, this is not expected to impact the 50.69 categorization.