2807 West County Road 75
Monticello, MN 55362

800.895.4999
xcelenergy.com

March 28, 2018

L-MT-18-010
10 CFR 50.90
10 CFR 50.69

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

Monticello Nuclear Generating Plant
Docket No. 50-263
Renewed Facility Operating License No. DPR-22

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 (FOL) of the Monticello Nuclear Generating Plant (MNGP).

The proposed amendment would modify the MNGP 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 MNGP Renewed FOL. 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 Probabilistic Risk Assessment (PRA) models described within this license amendment request (LAR) are the same as those described within NSPM submittal of the LAR dated December 19, 2017 to adopt TSTF-425 (ADAMS Accession No. ML17353A189), with routine updates applied. NSPM requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application 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.

NSPM requests approval of the proposed change by April 30, 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 there are any questions or 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 March 28, 2018.

Christopher R. Church
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure

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

ENCLOSURE

MONTICELLO NUCLEAR GENERATING PLANT

Evaluation of the Proposed Change

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LIST OF ATTACHMENTS

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 Monticello Nuclear Generating Plant (MNGP).

2.3 DESCRIPTION OF THE PROPOSED CHANGE

NSPM proposes the addition of the following condition to the Renewed Facility Operating License of the MNGP 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 March 28, 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).

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 PRA models described within this license amendment request (LAR) are the same as those described within NSPM submittal of the LAR dated December 19, 2017, to adopt

TSTF-425 (Reference 2), with routine maintenance updates applied. NSPM requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application 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.

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 3). 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.

Table 3-1 – IDP Changes from Preliminary HSS to LSS

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 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.

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 4) 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 3.4. The internal events PRA model described within this LAR is the same as the one described within NSPM's submittal of the LAR to adopt TSTF-425 (Reference 2).

- Fire Risks: Fire PRA model Revision 4. This PRA model was credited in the MNGP LAR to extend the integrated leak rate testing (ILRT) interval, as accepted by the NRC in Reference 5. Furthermore, the Fire PRA model described within this LAR is the same as the one described within NSPM's submittal to adopt TSTF-425 (Reference 2).

- Seismic Risks: SSEL referenced in the Individual Plant Examination of External Events (IPEEE) seismic analysis accepted by NRC Staff Evaluation Report dated April 14, 2000 (Reference 6).

- Other External Risks (e.g., tornados, external floods): Using the IPEEE screening process as approved by the NRC in Reference 6. 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 7), 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 8.

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 4. The RI-RRA method, as approved for use at Vogtle for 10 CFR 50.69, does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to 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 Class 2, 3, and non-class components. Consistent with the ANO RI-RRA method, Class 1 pressure retaining SSCs in the scope of the system being categorized will be assigned HSS and cannot be changed by the IDP. Therefore, the RI-RRA methodology for passive categorization is acceptable and appropriate for use at MNGP 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 MNGP 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 MNGP 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 9) and only utilizes methods previously accepted by the NRC. 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 fire PRA model.

3.2.3 Seismic Hazards

The MNGP categorization process will use the seismic margins analysis (SMA) performed for the IPEEE in response to Generic Letter (GL) 88-20, Supplement 4 (Reference 10), 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 MNGP 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 11). 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 MNGP 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 self-assessment and peer review processes as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the 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 4. 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 12) and Section 3.1.1 of EPRI TR-1016737 (Reference 13). 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 MNGP 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 MNGP 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 no additional sensitivity analyses are required to address MNGP 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 14), consistent with NRC Regulatory Issue Summary (RIS) 2007-06 (Reference 15).

The internal events PRA model, including internal flooding, was subject to a self-assessment and full-scope peer review conducted in accordance with RG 1.200, Revision 2, in April 2013. Additionally, a focused scope peer review was conducted in April 2017 to review the convolution analysis portion of the PRA model.

The fire PRA model was subject to a self-assessment and full-scope peer review conducted in accordance with RG 1.200, Revision 2, in March 2015. Additionally, a focused scope peer review was conducted in December 2016, to account for enhanced fire modeling methods that were incorporated into the model.

Finding closure reviews were conducted on the internal events, including internal flooding, and fire PRA models in August and October 2017, respectively. 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 16) as accepted by the NRC in Reference 17. 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 MNGP fire PRA model peer review. (Note: All internal events PRA model, including internal flooding, findings were closed during the finding closure review.)

- 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 MNGP 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 3).

- 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 18).

- RG 1.200, “An Approach for Determining Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2, March 2009 (Reference 14).

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 Monticello 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 regulations. 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 regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

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

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

   Response: No

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

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

Based on the above, 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 December 19, 2017 (ADAMS Accession No. ML17353A189)

3. 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)

4. 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)

5. NRC letter to NSPM, “Issuance of Amendment Re: Technical Specification 5.5.11, ‘Primary Containment Leakage Rate Testing Program’ (CAC No. MF7359)”, dated April 25, 2017 (ADAMS Accession No. ML17103A235)

6. NRC letter to NSP, “Review of Monticello Individual Plant Examination of External Events (IPEEE) Submittal (TAC No. M83644)”, dated April 14, 2000

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

8. 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)

9. 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)

10. 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)

11. 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

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

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

14. 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)

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

16. 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)

17. 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)

18. 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)

19. NSPM letter to NRC, “Monticello Nuclear Generating Plant: Response to Post-Fukushima Near-Term Task Force (NTTF) Recommendation 2.1, Flooding – Flood Hazard Reevaluation Report”, dated May 12, 2016 (ADAMS Accession No. ML16145A179)

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 3.2 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.

Attachment 2: Description of PRA Models Used in Categorization

Model	Baseline CDF	Baseline LERF	Comments
Internal Events PRA Model, including Internal Flooding, Revision 3.4, dated June 1, 2017. Full scope peer review against RG 1.200, Revision 2, in April 2013. Focused scope peer review in April 2017..	8.06E-06	6.95E-07	This model represents the current Internal Events, including Internal Flooding, PRA Model of Record.
Fire PRA Model, Revision 4.0, dated January 30, 2017. Full scope peer review against RG 1.200, Revision 2, in March 2015. Focused scope peer review in December 2016.	5.24E-05	6.39E-06	This model represents the current Fire PRA Model of Record.

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
2-1 (PP-B3) (PP-C3)	The requirement for this Supporting Requirement (SR) is to JUSTIFY credited spatial separation. This section does not provide the required justification. The documentation should include an evaluation that establishes why the separation provided by "space" will ensure that the adverse effects of fire will be substantially contained in each of the adjacent Physical Analysis Units (PAUs).	The fire PRA model was revised to evaluate spatially-separated fire areas in the multi-compartment analysis with the barrier failure probabilities set to 1.0. The F&O finding closure review team determined that the revised approach was acceptable. However, several isolated errors were identified where barrier probabilities were set to lower values. A documentation discrepancy was also identified. These will be corrected in a future update.	As recommended by the Closure Review Team, the remaining few isolated cases where the barrier failure probability is not set to 1.0 for spatially separated fire compartments will be corrected. These cases will be corrected by setting their barrier failure probability to 1.0 in the Multi-Compartment Analysis. These changes are not expected to have a significant impact on CDF or LERF.
2-5 (IGN-A7)	Section 5.6 discusses the apportionment of generic transient fire ignition frequencies and the development of the influencing factors for each area. The influencing factors were assigned by the Fire PRA analysts based on engineering judgment and a set of rules documented in Section 5.6.2 of the Ignition Frequency Notebook. Assignment of these values resulted in a comparatively low result. Based on the information contained in the Fire Modeling Database the influencing factors average as follows: Maintenance 1.7; Occupancy 2.2; and Storage 1.8. It's typically assumed that these factors will produce an average value, i.e., Medium or 3, by definition. Based on the values stated it appears	A comprehensive review of the influence factors was conducted in response to this F&O including a review by plant personnel. The maintenance influence factors were adjusted based on this review. The F&O finding closure review determined that the revised factors were acceptable. However, better justification of application of a "very low" factor in two compartments needs to be provided.	These changes are not expected to have any impact on CDF or LERF since the recommendations are associated with documentation changes to better explain modeling rationale. If the "very low" factor cannot be justified for compartments 8 and 33 then a different factor will be appropriately applied with a justification.

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
2-5 (cont.)	that the influencing factors may have been underestimated. To increase the accuracy and reliability it's suggested that these values be set or validated by plant operations and maintenance personnel. For example, numerous fire zones were assigned LOW maintenance factors including H2 Seal Oil/Condensate Pump Area, Turbine Condenser Area, Air Ejector Room, Admin Bldg. HVAC Room, Engineered Safety Features (ESF) Motor Control Center, 13.8 kV Switchgear Rooms, Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Rooms, Diesel Fuel Oil Pump House, etc. as these zones contain pumps, motors, electrical equipment that would require maintenance. LOW storage factor was assigned in numerous fire zones including Lube Oil Storage Room, Contaminated Equipment Storage Area, etc. which appear to be defined storage areas in the plant. Additionally, there are only 13 fire zones that are assigned storage factors greater than LOW.		
3-6 (PRM-B10)	The Fire PRA plant response model was not successfully modified to fail SSCs not selected in the ES element. Representative examples of this include: 1. In the PRM notebook, the basis for exclusion of MSO 5j is that 'Monticello does not credit operation of service water', however, service water is not failed in the logic model. 2. In section 3.2 of the PRM notebook Water, it	Documentation and model changes were made to the fire PRA to address this F&O. The F&O Closure review determined that the changes made largely address the issues identified in the F&O. However, there are still about ten component-level basic events that are not yet treated as "guaranteed failures" in the fire model. A sensitivity study was	The Closure Review Team Recommendation will be addressed by including the specified basic events in the fire failed events flag file. These changes are not expected to have a significant impact on total CDF or LERF since the current fire model contribution to total CDF and

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
3-6 (cont.)	states that use of the Fire Water System as a back-up to LPCI is not credited, however, this is not failed in the model. 3. In both the ES notebook and PRM notebook, it is stated that CRDH and SLBC were not used in the fire PRA, however, these are not successfully failed in the model. They were failed by putting appropriate flags set to 1.0 in the model. However, basic events for SBLCS components (L) and HFE's appear in the results. Basic events for CRD pump random failures (J) also appear in the model, with random failure probabilities. If the systems are correctly FLAGGED out, there should not be random failures of these systems. If the correct component is flagged, the logical 1.0 should propagate to the top of the tree, eliminating all other random failures. The fact that random events for these systems appear in cutsets indicate the correct basic event has not been chosen to be flagged. This particular example is not expected to be risk significant. 4. Individual components identified in Table D-1 of the ES notebook as not credited were not failed in the PRM [e.g., FPAP1AXXXR12-S - CONDENSATE PUMP P-1A FAILS TO RUN (SHORT TERM)] 5. Conversely - Basic events that were not failed in the model, yet were not included in table C-1 as credited [e.g., ABSLPCIAXG - LPCI MCC FAULT (MCC-133A)]	performed which demonstrated that inclusion of these events result in a change in CDF and LERF of 0.19% and 1.41%, respectively.	LERF is 0.19% and 1.41%, respectively.
4-11	An initial ambient temperature of 20°C was utilized in the fire modeling calculations for all	Validation studies of the three fire modeling models used in the fire	The Closure Review Team recommendation will be

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
4-11 (cont.) (FSS-D4)	MNGP fire zones. This ambient temperature does not appear to be appropriate for areas that are not temperature controlled such as the Turbine Building, Diesel Generator Building, and areas of the Reactor Building.	PRA were performed. In each case, the model biases are dispositioned as reasonable for their use based upon Chapter 4 of NUREG-1934. The F&O closure review team found the validation studies to be appropriate for cases in which the ambient temperature is 20°C or less. Additional justification is required for plant areas which may have higher ambient temperatures.	addressed by revising the fire models using expected plant ambient temperatures for each fire zone. These changes are not expected to have a significant impact on total CDF or LERF due to the conservative fire modeling methods used.
4-20 (FSS-C5) (FSS-D9)	Although the damage criteria for sensitive electronics is defined in the Single Compartment Analysis Notebook 016015-RPT-06 and zones of influence (critical distances) are calculated in the Fire Modeling Database, there is no specific discussion of how specific sensitive electronics at Monticello are analyzed in the FPRA.	The fire PRA Single Compartment Analyses have been updated to document the analysis associated with the treatment of sensitive electronics. The F&O closure review team determined that the methods used for all areas other than the main control room are either in agreement with FAQ 13-0004 or conservative in comparison to the FAQ. However, additional verification and documentation of the main control board configuration for sensitive electronics was determined to be required by the F&O closure team to fully resolve this F&O.	These changes are not expected to have any impact on CDF or LERF since the recommendations are associated with documentation changes to better explain modeling rationale.
4-29 (FSS-A1) (FQ-A3)	Appendix E of the Single Compartment Analysis Notebook 016015-RPT-06 identifies that scenarios for cable fires caused by welding and cutting and self-ignited cable fires result in high total CDF contributions and further evaluation	For fire compartments resulting in CDF greater than 1.0E-08/year, the process in FAQ 13-0005 was applied for cable fires caused by welding. The Single	These changes are not expected to have any impact on CDF or LERF since the recommendations are associated with documentation

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
4-29 (cont.)	and refinement will be completed after risk reduction activities are completed. These scenarios are not currently quantified in the FPRA model.	Compartment Analysis notebook has been updated to document the process used to treat cable fires due to hotwork, self-ignited cable fires, and junction box fires. The F&O closure review team determined that the model changes were appropriate. However, the F&O remains open since the documentation of the process used should be enhanced.	changes to better explain modeling rationale.
4-33 (FSS-A5)	Wall and corner effects are not accounted for in the FLASH-CAT modeling for heat release rate calculations that are used for the CFAST hot gas layer models.	Justification has been added to the fire PRA documentation to demonstrate that the FLASH-CAT analyses would bound as-built conditions. However, the F&O finding closure review team determined that the results may not be bounding for cable trays in wall or wall-corner locations. Verification that FLASH-CAT results were not used for such configurations needs to be performed.	The Closure Review Team recommendation will be addressed by reviewing the detailed modeled fire scenarios to determine which ones meet the definition of wall or corner and revising their model to address wall and corner effects or provide justification that the approach was bounding for the expected ignition sources in the fire zone. These changes are not expected to have a significant impact on total CDF or LERF due to the conservative fire modeling methods used.
6-3 (IGN-A1)	Battery Chargers have been counted as either Bin 10 or Bin 15 in the IGN development. It appears that well sealed low voltage panels (e.g. lighting panels) have been included in the	Many of the battery chargers have been re-assigned to Bin 10 for ignition frequency determination, but some still need to be re-assigned to the	The Closure Review Team recommendation will be addressed by correctly assigning battery chargers D70, D80, and D90 to bin 15

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
6-3 (cont.)	bin 15 count that should be excluded.	correct bin. A sensitivity study shows that a change in CDF and LERF of -0.38% and -0.16%, respectively will result once the battery charger re-assignments are completed. The F&O closure team reviewed the battery charger re-binning. The team did not identify any specific issues with inappropriate counting of sealed low voltage panels. This F&O remains open since not all battery chargers have yet been reassigned.	rather than bin 10. These changes are not expected to have a significant impact on total CDF or LERF since their current fire model contribution to total CDF and LERF is -0.38% and -0.16%, respectively.
6-9 (SY-A19) (PRM-B9) (SY-A1) (SY-A16) (SY-A14) (SY-B9) (SY-B5) (DA-E2)	Common cause and test and maintenance, and pre-initiator human error basic events for core spray and RHR are missing from the ASD Logic. Additionally, the alternate shutdown modeling of core spray train B is also missing the failure mode for: 'CS Pump P-208B to run after the first hour.' Also, the review found that power supplies for some of the active components were missing from the alternate shutdown logic.	Common cause failure, test & maintenance, and pre-initiator basic events for core spray were added. RHR modeling under multiple gates for the various RHR functions were also added; however, the F&O closure review team identified that pump maintenance unavailability events needed to be added to some of the RHR fault tree logic. Power supplies were reviewed and added to the model as appropriate.	The Closure Review Team recommendation will be addressed by correctly adding in the remaining one or two basic events related to RHR “B” pump maintenance unavailability that were not included under the alternate shutdown logic. These changes are not expected to have a significant impact on total CDF or LERF since the few common cause and maintenance events that will be included are not affected by a fire and are generally much less likely than the equipment failure probability.

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
6-11 (PRM-B9) (SY-A1) (SY-A2) (UNC-A2) (DA-E2)	Basic events added to the FPRA associated with the following failure modes have failure probabilities set to zero: • UPS panel fault • Circuit breaker fails to remain open • Circuit breaker fails to open • Fused disconnect switch, fuse spuriously fails • Transformer fault • CS pump fails to start • CS pump fails to run 1st hour • MOV fails to remain open • MOV fails to open • MOV fails to close • 125 VDC distribution panel fault • AOV fails to remain closed • AOV fails to remain open • Level transmitter spurious operation • RHR Pump fails to run • RHR Pump fails to start • Solenoid valve fails to transfer	Many of the identified issues have been corrected with model changes. However, a number of "breaker fail to remain open" events and other miscellaneous events that require data have not yet been updated in the model.	The Closure Review Team recommendation will be addressed by including the specified basic events in the fire failed events flag file. These changes are not expected to have a significant impact on total CDF or LERF from the specified basic events since their current fire model contribution to total CDF and LERF is 0.00%.
7-3 (PRM-A4) (PRM-C1) (SY-A1) (PRM-B9)	The purpose SR PRM-A4 is to confirm that the plant response model is constructed in such a manner that it reflects the failure of identified equipment due to the loss of the associated equipment selected cables. Based on the review by peers, the following issues were identified. These are based on limited time to review and are only examples. The fault tree modeling of essential cues for HFE HPI-CNTRLY is not correct. The cues are modeled under gate F-HEP-CNTRLML, and ANDed with the medium LOCA initiator	Corrections were made to the Feedwater (FW) level control logic. Corrections were also made to the FRANX mapping and fault tree models to consider dependency on both ADS channels in the Alternate Shutdown logic. Corrections were made to the Main Steam Isolation Valve (MSIV) pseudo-component logic. A review of the pseudo-component and interlock	The Closure Review Team recommendation will be addressed by adding the HEP HPI-CNTRLY under and OR gate with L-RPV-INSTR: Loss of RPV Instrumentation in the fault tree logic. These changes are not expected to have a significant impact on total CDF or LERF from the specified basic events since their current fire model contribution to total

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
7-3 (cont.)	IE_MLOCA 2.72E-4/yr (with no FPRA IE modeled there). The FPRA development team concurred that the cues modeling gate LRPV-INSTR should be input into OR gate F020 (ORed with HPI-CNTRLY). Equipment Selection report 016015-RPT-03 Table B-2 identifies ADS-CHANNEL-A:Avail:Non-Spur and ADS-CHANNEL-B:Avail:Non-Spur low level pseudo functions and equipment dependencies. It was determined during CS that the cables were properly mapped to the ADS pseudo component. Equipment SV271A, C, and D are dependent on both ADS A and B channel cables. However, review of the FRANX database FPRA CDF 2-2 determined improper Component to basic event mapping was made to the PRM. ADS channel A is mapped to SV271A and ADS channel B is mapped to the remaining SV271C and D. A review of pseudo components MSIV-ISOL-A:Avail:Avail and LLS-DIV-B1:Avail:Non-Spur as identified in the ES procedure determined there was no modeling of the component to basic event relationship. From peer discussion it was determined that the noted pseudo-components were determined not required in the model following cut set review by the utility. In addition, there was no evidence that the interlocks on the cable selection data worksheets were reviewed and properly incorporated into the PRM.	modeling was performed and changes were incorporated into the model. However, the F&O finding closure review team identified some residual modeling issues with the HFE HPI-CNTRLY that is located in the FW level control logic which have not yet been corrected.	CDF and LERF is 0.00%.
7-4	It is not clear if the internal events PRA initiating events and accident sequences applicable to	The fire PRA model has been updated to use an adjusted Large	The Closure Review Team recommendation will be

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
7-4 (cont.) (SY-A1) (PRM-B1) (PRM-B9) (PRM-C1) (FQ-A2)	two or more SRVs open similar to LLOCA have been correctly applied. If it is deemed that opening of two or more SRVs does not need to mimic LLOCA, then provisions for a new initiating event, success criteria and accident sequence is required. PRM calculation 016015-RPT-05, MSO 3a and 3b for potential opening of two or more SRVs added additional logic to the PRM. Potential opening of all SRVs mimics sequences similar to Large LOCA. Review of the PRM calculation noted in section 6.0 that the initiating events and accident sequences embodied in the MNGP internal events model are used as the basis for development of the FPRA model. Additional information received from the utility representative regarding the review of internal events initiating events determined that Large LOCA was deemed not applicable to the FPRA. Review of the PRM CAFTA model located MSO failure of more than one SRV via gate F_SORV_2of8 with parents to gates different from LLOCA. If it is deemed that opening of all SRVs does not need to mimic LLOCA, then provisions for a new initiating event, success criteria and accident sequence is required.	LOCA event tree for a spurious opening of 2 or more SRVs that do not reclose due to an MSO event. However, the F&O finding closure review team identified additional locations in the model where the revised logic model still needs to be added to fully account for these fire-induced SRV opening scenarios.	addressed by performing thermal hydraulic MAAP analysis to determine the success criteria for the opening of 2 or more SRVs. The fault tree model will be revised to reflect the determined success criteria.
FO-1 (FSS-D3) (FSS-D4)	The analysis results of the thermal heat soak method appear to credit ventilation limited burning in several PAUs without providing sufficient basis. An example is the Group 1 scenarios listed in Table J-6 of 016015-RPT-06. Each of the four fire case CFAST results have sensitivity cases due to the development of ventilation limited conditions. The baseline CFAST results do not result in damage to a	Documentation of the results of a sensitivity analysis that was conducted to exclude credit of ventilation-limited burning was added. This analysis demonstrates that the oxygen-limited fire reaches the damage threshold for the cable at an earlier time. This information was	The Closure Review Team recommendation will be addressed by reviewing the cable heat soak fire modeling that credits ventilation limited burning. Credit for ventilation limited burning will be removed.

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
FO-1 (cont.)	generic target over a 60 min time interval. The CFAST sensitivity cases that were originally run with additional ventilation to verify constant exposure damage times would likely result in damage to a generic TP target when assessed in the heat soak model	used to justify the use of the ventilation-limited cases for estimating the time to hot gas layer formation. However, the F&O finding closure review team identified issues with the sensitivity case and its applicability in certain situations. Additional justification concerning the treatment of the ventilation-limited modeling for those areas needs to be developed.	These changes are not expected to have a significant impact on total CDF or LERF due to the conservative fire modeling methods used.
FO-2 (FSS-D3) (FSS-D4) (FSS-H4) (FSS-H5) (FSS-H9)	A number of documentation issues have been identified. These include: A. There are a number of scenarios that appear to credit the thermal heat soak method listed in the FMDB but the HGL times do not match any scenario listed in Report 016015-RPT-06. An example is Equipment C-18 in tblIgnitionScenarios of the fire modeling database. Scenario 2 and the corresponding comment indicates HGL time is 25 minutes based on heat soak time. Table J-6 in Section J-6 of Report 016015-RPT-06 does not list any damage times from any ignition source – secondary combustible grouping of 25 minutes. The database should be checked for additional examples and addressed as necessary. B. There are a number of scenarios listed in the FMDB indicating HGL timings but there is inconsistent indication for when a scenario credits the thermal heat soak method. The only method to verify that the thermal heat soak method was applied in the FMDB was to query	The various documentation issues identified in this F&O have been addressed. However, the F&O finding closure review team determined that additional information needs to be included in the documentation concerning the impacts of accumulation of damage at low temperature on cables and on the impacts of cable size on the heat soak methodology.	These changes are not expected to have any impact on CDF or LERF since the recommendations are associated with documentation changes to better explain modeling rationale.

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
FO-2 (cont.)	the results in TblIgnitionScenarios and match HGL timing to those reported in Table J-6 in Section J-6 of Report 016015-RPT-06, or to search for comment fields in IgScnComment. C. Description of the method in which the results from the thermal heat soak analysis is incorporated in the MCA. It is not clear where the MCA heat soak calculations or their direct inputs are among the reviewed material. Section 5.4 of the MCA Report 016015-RPT-08 points to Table J-6 in the SCA Report 016015-RPT-06 for heat soak results. However, the compartments listed in Table J-6 do not completely match the compartments that were screened from the MCA using the heat soak method. This suggests that there may be other heat soak results that are not documented. For example, the MCA screens combinations involving compartments 19B and 32A, but the SCA does not indicate that thermal heat soak analysis was performed for these compartments. In addition it appears that the heat soak method was used to increase the HGL timing for combinations involving compartment 32A, but there is no documentation of the results used to justify this timing. D. It is difficult to link the CFAST Group and the Fire Case as listed in Table J-6 in the SCA Report 016015-RPT-06 with the damage integral result listed in the database for the SCA and MCA where applied. There is no consolidated table which includes the CFAST Group and the Fire Case as applied to a given scenario in the FMDB.		

F&O Number (SR)	Description	Resolution	Disposition for 10 CFR 50.69
FO-2 (cont.)	E. The thermal heat soak method does not fully document the approach for target damage accumulation at low temperatures. No technical deficiencies were noted in the method review; however, the treatment of the low temperature damage accumulation can have a significant influence on the overall result and should be clearly discussed. F. Additional documentation of the limits of applicability for the thermal heat soak method is needed in Report 016022-RPT-01. For example, is there a maximum exposure temperature or maximum/minimum cable size over which the results can be used? G. Documentation of sources of model uncertainty and its treatment in the analysis is needed to achieve a Cat II for FSS-H5 and FSS-H9. Since the heat soak method is an interpolation of the generic cable damage times listed in NUREG/CR-6850, there is no new uncertainty introduced with the heat soak method, except for the damage accrual estimates at temperatures below the damage threshold. H. Reports 016015-RPT-06, Rev. 4 and 016015-RPT-08, Rev. 4 were draft at the time of the review. They will need to be finalized and signed.		

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 45 miles from the site. The IPEEE reports results from bounding assessments to demonstrate that the risk due to this hazard is less than 1.0E-06 per year.
Avalanche	Y	C3	The topography surrounding MNGP precludes the possibility of a snow avalanche.
Biological Event	Y	C5	Actions committed to and completed by MNGP in response to Generic Letter 89-13 (Service Water System Problems Affecting Safety-Related Equipment) provide on-going control of biological hazards. These controls are described in MNGP procedure EWI-08.22.01, “Generic Letter 89-013.” Additionally, actions taken in response to INPO SOER 07-2 (Intake Structure Blockage) provide an additional layer of biological hazard management. Based on these actions, the hazard is slow to develop and can be identified via monitoring and managed via standard maintenance processes.
Coastal Erosion	Y	C3	The mid-western location of MNGP precludes the possibility of coastal erosion.
Drought	Y	C5	These effects would take place slowly allowing time for orderly plant reductions including shutdowns.
External Flooding	Y	C1	The external flooding hazard at MNGP 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 12, 2016 (Reference 19). 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 was evaluated and will not challenge any safety functions at MNGP.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Extreme Wind or Tornado	Y	C1 PS4	Wind damage is bounded by tornadoes, and the tornado wind speed corresponding to the 1.0E-06 per year exceedance frequency is less than the MNGP design value. Therefore, damage due to the forces associated with extreme winds or tornadoes can be screened. For tornado missiles, those areas housing critical equipment required to assure safe shutdown were designed to prevent penetration of exterior walls by two types of missiles that could be generated by a tornado: A) A utility pole 35-feet long by 14-inches in diameter and a unit weight of 35 pounds per cubic foot having a velocity of 200 mph; and B) A one ton missile, such as a compact type automobile, traveling at 100 mph at a maximum height of 25-feet above grade and with a contact area of 25 square feet. The CDF associated with tornado missiles is less than 1.1 E-07 per year.
Fog	Y	C1 C4	The principal effects of such events (such as freezing fog) 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 MNGP.
Forest or Range Fire	Y	C1 C3 C4	The site landscaping and lack of forestation prevent such fires from posing a threat to MNGP. Furthermore, 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 MNGP.
Frost	Y	C1 C4	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 MNGP.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Hail	Y	C1 C4	Hail is bounded by other events 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 MNGP.
High Summer Temperature	Y	C1 C4 C5	The principal effects of such events would result in elevated river temperatures which are monitored by station personnel. Should the ultimate heat sink temperature exceed the Technical Specification limit, an orderly shutdown would be initiated. These effects would take place slowly allowing time for orderly plant reductions, including shutdowns. Another potential consequence would be to cause a loss of offsite power. This consequence is addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for MNGP.
High Tide, Lake Level, or River Stage	Y	C3 C4	High tide or lake level not applicable to the site because of location. Impact of High River Stage is included as an impact to external flooding.
Hurricane	Y	C3 C4	The mid-western location of MNGP precludes the possibility of a hurricane. Additionally, hurricanes would be covered under Extreme Wind or Tornado and Intense Precipitation.
Ice Cover	Y	C1 C4	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 MNGP. Furthermore, ice induced flooding was evaluated in the FHRR and determined to be bounded by External Flooding.
Industrial or Military Facility Accident	Y	C3 C4	There are no military facilities within the proximity to the plant (the closest is a National Guard facility at MSP airport, ~45 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 MNGP internal events PRA addresses risk from internal flooding events.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Internal Fire	N	N/A	The MNGP internal fire PRA addresses risk from internal fire events.
Landslide	Y	C3	Not applicable to the site because of topography.
Lightning	Y	C1 C4	Lightning strikes can result in losses of offsite power or surges in instrumentation output if grounding is not fully effective. The latter events often lead to reactor trips. Both results are incorporated into the MNGP internal events PRA model through the incorporation of generic and plant-specific data.
Low Lake Level or River Stage	Y	C5	These effects would take place slowly over time allowing for orderly plant reductions, including shutdowns.
Low Winter Temperature	Y	C1 C4 C5	The principal effects of such events would be to cause a loss of offsite power. These effects would take place slowly allowing time for orderly plant reductions, including shutdowns. At worst, the loss of offsite power events would be subsumed into the base PRA model results.
Meteorite or Satellite Impact	Y	PS4	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	The nearest hazardous material or natural gas pipeline is located more than one mile south of the plant. The effects on plant structures due to a pipeline explosion located approximately one mile from the site are bounded by tornado loadings.
Release of Chemicals in Onsite Storage	Y	C4 PS1	Chlorination of water systems is performed using a hypochlorite system. No chlorine gas is stored on-site. The control room envelope habitability program, as described in Technical Specification 5.5.13 ensures NSPM retains the ability to safely operate the plant in the event of a chemical release onsite or in the vicinity of the plant. Onsite and offsite chemical hazards are periodically re-evaluated to ensure continued compliance with this specification.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
River Diversion	Y	C3 C5	Diversion of the Mississippi River was reviewed under the Fukushima 10 CFR 50.54(f) Request for Information and the FHRR was submitted to the NRC for review in Reference 19. The location of the MNGP site is adjacent to the natural channel of the Mississippi River. A review of the United States Geological Survey from 1961 and 2013 showed no change in the course of the Mississippi River channels in the site vicinity. The river channel in the area of the site does not include prominent bluffs or other features that could be susceptible to landslide which could potentially result in migration of the channel more directly towards the site. There are no man-made channels, canals, diversions, or permanent levees used for conveyance of water and flood protection near the site.
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 MNGP site is located on the Mississippi River and is not susceptible to a seiche.
Seismic Activity	N	N/A	See information in Section 3.2.3 of the enclosure.
Snow	Y	C1 C4	Snow cover is included as an input to the probable maximum flood. Potential impacts are covered under external flooding. Snow loading on buildings was considered during the design of the plant and maintenance procedures exist to ensure snow loading remains within the design load.
Soil Shrink-Swell Consolidation	Y	C3	Excluded based on structures founded on bedrock and/or engineered fill.
Storm Surge	Y	C3	The mid-western location of MNGP precludes the possibility of a sea level driven storm surge.
Toxic Gas	Y	C4	The hazards associated with toxic gas are screened elsewhere in this table (e.g., Release of Chemicals in Onsite Storage)

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Transportation Accident	Y	C1 C3 C4	Land Transportation – Based on the proximity of the nearest major roadways, potential impacts are covered by Extreme Wind or Tornado as well as Release of Chemicals in Onsite Storage. 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 – MNGP is located near the headwaters of the Mississippi River. Therefore, the river is shallow near the plant limiting the activity primarily to pleasure craft.
Tsunami	Y	C3	The mid-western location of MNGP precludes the possibility of a tsunami.
Turbine-Generated Missiles	Y	PS4	Turbine-Generated Missiles are evaluated in USAR Section 12.2.3. The probability of unacceptable damage from turbine missile has been calculated to be 1.76E-08 per year.
Volcanic Activity	Y	C3	Not applicable to MNGP.
Waves	Y	C3 C4	Waves associated with adjacent large bodies of water are not applicable to MNGP. The potential impacts of waves associated with External Flooding was evaluated and wave runup was determined to be below flood protection barriers.
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 RA-Sa-2009	
	C2	Event has lower mean frequency and no worse consequences than other events analyzed	NUREG/CR-2300 ASME/ANS PRA Standard RA-Sa-2009	
	C3	Event cannot occur close enough to the plant to affect it	NUREG/CR-2300 ASME/ANS PRA Standard RA-Sa-2009	
	C4	Event is included in the definition of another event	NUREG/CR-2300 ASME/ANS PRA Standard RA-Sa-2009	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 RA-Sa-2009	
Progressive Screening	PS1	Design basis hazard cannot cause a core damage accident	ASME/ANS PRA Standard RA-Sa-2009	
	PS2	Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan	NUREG-1407 ASME/ANS PRA Standard RA-Sa-2009	
	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 RA-Sa-2009	
	PS4	Bounding mean CDF is < 1.0E-06 per year	NUREG-1407 ASME/ANS PRA Standard RA-Sa-2009	
Detailed PRA		Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 ASME/ANS PRA Standard RA-Sa-2009	

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

Assumption/Uncertainty	Discussion	Disposition
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.	This uncertainty has the potential to push more components into the HSS categorization than LSS. Future model enhancement incorporating NUREG-2169 will reduce this uncertainty, but not eliminate it, due to larger data set developing frequencies.
Fire PRA Cable Selection is based on risk significance of PRA component.	Cable Selection for PRA components was limited based on risk significance of the Internal Events component's impact to Fire PRA risk.	This uncertainty has the potential to push more components into the HSS categorization than LSS. Further cable selection to credit PRA equipment in the plant will be prioritized by the risk significance.
Common cause failures are developed using available industry data.	This uncertainty potentially affects all SSCs evaluated during 50.69 categorization.	As directed by NEI 00-04, common cause 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 common cause failure probabilities are accounted for in the categorization process.

Assumption/Uncertainty	Discussion	Disposition
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 primary conservatism introduced by this task is associated with the heat release rates specified in NUREG/CR-6850.	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.	This uncertainty has the potential to push more components into the HSS categorization than LSS. Future model enhancement incorporating NUREG-2178 will reduce this uncertainty, but not eliminate it, due to larger data set developing frequencies.
Human Reliability Analysis	Human Reliability Analysis (HRA) is a continually evolving discipline. The human error probabilities 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.	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 Human Error Probabilities are accounted for in the 50.69 application.