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

August 29, 2019

Mr. Christopher Church
Senior Vice President and Chief
Nuclear Officer
Northern States Power Company - Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT NO. 203 RE: ADOPTION OF 10 CFR 50.69 – "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURE, SYSTEMS AND COMPONENTS OF NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0076)

Dear Mr. Church:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 203 to Renewed Facility Operating License No. DPR-22, for the Monticello Nuclear Generating Plant (MNGP). The amendment consists of changes to the technical specifications in response to your application dated March 28, 2018, as supplemented by letters dated March 13, 2019, and May 15, 2019.

The amendment adds a condition to the MNGP renewed facility operating license to allow the implementation of Title 10 of the *Code of Federal Regulations*, Section 50.69, "Risk informed categorization and treatment of structures, systems and components for nuclear power reactors."

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

Sincerely,

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosure:
1. Amendment No. 203 to DPR-22
2. Safety Evaluation

cc: Listserv

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 203
Renewed License No. DPR-22

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

   A. The application for amendment by Northern States Power Company – Minnesota dated March 28, 2018, as supplemented by letters dated March 13, 2019 and May 15, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Lisa M. Regner, Acting Branch Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: August 29, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 203

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-263

Renewed Facility Operating License No. DPR-22

Replace the following pages of the Renewed Facility Operating License No. DPR-22 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating area of change:

Remove	Insert
License	License
Page 10	Page 10
Page 11	Page 11
Page 12	Page 12

2. Level 1 performance criteria; and

3. The methodology for establishing the limit curves used for the Level 1 and Level 2 performance.

(a) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report that includes a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the ACE 2.0 and ACE2.0-SPM specific bias and uncertainties. The report will be provided within 90 days of the completion of EPU power ascension testing.

(b) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer in accordance with the inspection guidelines provided to the NRC.

(c) The results of the visual inspections of the steam dryer shall be reported to the NRC staff within 90 days following startup from the respective refueling outage.

(d) At the end of the second refueling outage, following the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results for NRC review and approval.

The license conditions described above shall expire (1) upon satisfaction of the requirements in Paragraphs 15(f) and 15(g), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and (2) upon satisfaction of the requirements specified in Paragraph 15(h).

16. <u>Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"</u>

NSPM is approved to implement 10 CFR 50.69 using the approaches for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding and internal fire, with the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards (e.g., external flooding and high winds) updated using the external hazard screening significance criteria identified in

ASME/ANS PRA Standard RA-Sa-2009, as endorsed in RG 1.200, Revision 2; as specified in MNGP License Amendment No. 203 dated August 29, 2019.

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

D. NSPM shall immediately notify the NRC of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

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

F. NSPM shall observe such standards and requirements for the protection of the environment as are validly imposed pursuant to authority established under Federal and State law and as determined by the Commission to be applicable to the facility covered by this renewed facility operating license.

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

H. The Updated Safety Analysis Report supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. NSPM shall complete these activities no later than September 8, 2010, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

I. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

J. Upon implementation of Amendment No. 160 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with TS 5.5.13.c(i), the assessment of CRE habitability as required by Specification 5.5.13.c(ii), and the measurement of CRE pressure as required by Specification 5.5.13.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.4.4, in accordance with Specifications 5.5.1.3.c(i), shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from June 4, 2004, the date of the most recent successful tracer gas test, as stated in the letter (L-MT-04-049, dated November 18, 2004) in response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specifications 5.5.13.c(ii), shall be within 3-years, plus the 9-month allowance of SR 3.0.2, as measured from June 4, 2004, the date of the most recent successful tracer gas test, as stated in the letter (L-MT-04-049, dated November 18, 2004) in response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.13.d, shall be within 24 months, plus the 184 days allowed by SR 3.0.2, as measured from October 17, 2008, the date of the most recent pressure measurement test, or within 184 days if not performed previously.

K. This renewed operating license is effective as of the date of issuance and shall expire at midnight, September 8, 2030.

FOR THE NUCLEAR REGULATORY COMMISSION
/RA/
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A - Technical Specifications
2. Appendix B - (Deleted per Amendment 15, 12/17/82)
3. Appendix C - Additional Conditions

Date of Issuance: November 08, 2006

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated March 28, 2018 (Reference 1), as supplemented by the letters dated March 13, 2019 (Reference 2), and May 15, 2019 (Reference 37), Northern States Power Company, a Minnesota corporation doing business as Xcel Energy (NSPM, the licensee), submitted a license amendment request (LAR) for the Monticello Nuclear Generating Plant (Monticello, MNGP). The amendment requested a new license condition be added to the Monticello Renewed Facility Operating License (RFOL) to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed process that includes several approaches and methods for categorizing SSCs according to their safety significance.

On January 31, 2019 (Reference 17), the U.S. Nuclear Regulatory Commission (NRC or Commission) staff requested additional information from the licensee. The licensee responded to the NRC's requests for additional information (RAIs) with supplemental letters dated March 13, 2019 (Reference 2), and May 15, 2019. The supplemental letters provided additional information that clarified the application; did not expand the scope of the application as originally noticed; and did not change the NRC staff proposed no significant hazards consideration determination as published in the *Federal Register* on May 22, 2018 (83 FR 23735).

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Categorization and Treatment of SSC

A risk-informed approach to regulation enhances and extends the traditional deterministic regulation by considering risk in a comprehensive manner. Specifically, a probabilistic approach allows 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. Probabilistic risk assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures (CCFs). To take advantage of the safety enhancements available through the use of PRA, the NRC promulgated a new regulation, 10 CFR 50.69, in the *Federal Register* on November 22, 2004 (69 FR 68008), which became effective on December 22, 2004. The provisions of 10 CFR 50.69 allow for adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design basis functions. For SSCs categorized as low safety significance (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety significance (HSS), requirements may not be changed.

Regulation 10 CFR 50.69 contains requirements regarding how a licensee categorizes SSCs using a risk-informed process; adjusts treatment requirements consistent with the relative significance of the SSC; and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decisionmaking process (IDP) which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability and is a function of the 10 CFR 50.69 categorization process results and associated bases. Finally, periodic assessment activities are conducted to adjust the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable functional requirements.

Regulation 10 CFR 50.69 does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. Important-to-safety electrical equipment, that is covered within the scope of 10 CFR 50.69 and relied on for mitigating the design basis accident, will continue to have evidence of environmental qualification demonstrating that the equipment can perform its safety function during and following a design basis accident. For SSCs that are categorized as HSS, existing treatment requirements are maintained or potentially enhanced. Conversely, for SSCs categorized as LSS that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable, level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that has HSS.

2.2 Licensee Proposed Changes

The May 15, 2019, supplemental letter proposed, in response to the NRC's RAI 07, to amend the MNGP RFOL by adding the following license condition that would allow for the implementation of 10 CFR 50.69:

> NSPM is approved to implement 10 CFR 50.69 using the approaches for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events,

including internal flooding and internal fire, with the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE [individual plant examination of external events] Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards (e.g., external flooding and high winds) updated using the external hazard screening significance criteria identified in ASME/ANS [American Society of Mechanical Engineers/American Nuclear Society] PRA Standard RA-Sa-2009, as endorsed in [Regulatory Guide] RG 1.200, Revision 2; as specified in MNGP License Amendment No. [XXX] dated [DATE].

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

Section 3.1.2 of the LAR proposes using a categorization method for passive components not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference 5), for passive component categorization, but was approved by the NRC for ANO-2. The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for Class 2 and 3 pressure retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1" (Reference 27).

2.3 Regulatory Review

The NRC staff reviewed the LAR as supplemented to determine whether: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) activities proposed will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendments will not be inimical to the common defense and security, or the health and safety of the public. The NRC staff considered the following regulatory requirements and guidance during its review of the proposed changes.

Regulatory Requirements

Regulation 10 CFR 50.69 provides an alternative approach for establishing requirements for treatment of SSCs for nuclear power reactors using an integrated and systematic risk-informed process for categorizing SSCs according to their safety significance. Specifically, for SSCs categorized as LSS, alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of HSS, requirements may not be changed.

Regulation 10 CFR 50.69(c) requires use of an IDP to categorize safety-related and nonsafety-related SSCs according to the safety significance of the functions they perform into one of the following four RISC categories, which are defined in 10 CFR 50.69(a), as follows:

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

The SSCs are classified as having either HSS functions (i.e., RISC-1 and RISC-2 categories) or LSS functions (i.e., RISC-3 and RISC-4 categories). For HSS SSCs, 10 CFR 50.69 maintains current regulatory requirements (i.e., it does not remove any requirements from these SSCs) for special treatment. For LSS SSCs, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d). For RISC-3 SSCs, licensees can replace special treatment with an alternative treatment. For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements, and RISC-4 SSCs are removed from the scope of any applicable special treatment requirements identified in 10 CFR 50.69(b)(1).

Regulation 10 CFR 50.69(c)(1) states that SSCs must be categorized as RISC-1, RISC-2, RISC-3, or RISC-4 SSCs using a categorization process that determines if an SSC performs one or more safety significant functions and identifies those functions. The process must:

(i) Consider results and insights from the plant-specific PRA. This PRA must, at a minimum, model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(ii) Determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

(iii) Maintain defense-in-depth (DID).

(iv) Include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of 10 CFR 50.69(b)(1) and 10 CFR 50.69 (d)(2) are small.

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

(v) Be performed for entire systems and structures, not for selected components within a system or structure.

Regulation 10 CFR 50.69(c)(2) of states:

> The SSCs must be categorized by an IDP staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.

Regulation 10 CFR 50.69(b)(3) states that the Commission will approve a licensee's implementation of this section by issuance of a license amendment if the Commission determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As stated in 10 CFR 50.69(b), after the NRC approves an application for a license amendment, a licensee may voluntarily comply with 10 CFR 50.69, as an alternative to compliance with the following requirements for LSS SSCs:

(i) 10 CFR Part 21,
(ii) A portion of 10 CFR 50.46a(b),
(iii) 10 CFR 50.49,
(iv) 10 CFR 50.55(e),
(v) Specified requirements of 10 CFR 50.55a,
(vi) 10 CFR 50.65, except for paragraph (a)(4),
(vii) 10 CFR 50.72,
(viii) 10 CFR 50.73,
(ix) Appendix B to 10 CFR Part 50,
(x) Specified requirements for containment leakage testing, and
(xi) Specified requirements of Appendix A to 10 CFR Part 100.

Guidance

NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 3), describes a process for determining the safety significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69. This categorization process is an IDP that incorporates risk and traditional engineering insights. Revision 0 of NEI 00-04 provides options for licensees implementing different approaches depending on the scope of their PRA models. It also allows for the use of non-PRA approaches when PRAs models have not been developed. The NEI 00-04 guidance identifies non-PRA methods to be used as an approach, such as fire-induced vulnerability evaluation (FIVE) to address internal fire risk, SMA to address seismic risk, and guidance in the Nuclear Management and Resource Council (NUMARC) report NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," (Reference 4), to address shutdown operations. As stated in RG 1.201, Revision 1, such non-PRA-type evaluations will result in more conservative categorization, in that special treatment requirements will not be allowed to be relaxed for SSCs that are relied upon in such evaluations. The degree of relief that the NRC will accept under 10 CFR 50.69 (i.e., SSCs subject to relaxation of special treatment requirements) will be commensurate with the assurance provided by the evaluations performed to assess and characterize the SSCs risk.

Sections 2 through 10 of NEI 00-04, Revision 0, guidance describes a steps/elements of the SSC categorization process to be performed for meeting the requirements of 10 CFR 50.69, as follows:

- Sections 3.2 and 5.1 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(i).
- Sections 3, 4, 5, and 7 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(ii).
- Section 6 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iii).
- Section 8 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iv).
- Section 2 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(v).
- Sections 9 and 10 provide specific guidance corresponding to 10 CFR 50.69(c)(2).

Additionally, Section 11 of NEI 00-04, Revision 0, provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f). Section 12 of NEI 00-04, provides guidance on periodic review related to the requirements in 10 CFR 50.69(e). Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience, as required by 10 CFR 50.69(c)(1)(ii).

RG 1.201, Revision 1, endorses the categorization process described in NEI 00-04, Revision 0, and states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events. RG 1.201, Revision 1, clarifies that the NRC staff expects that licensees proposing to use non-PRA approaches in their categorization should provide a basis in the submittal for why the approach, and the accompanying method employed to assign safety significance to SSCs, is technically acceptable. The guidance further states that as part of the NRC's review and approval of an LAR to adopt 10 CFR 50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from an SMA to a seismic PRA (SPRA)), the licensee or applicant will need to seek NRC approval via a license amendment of the implementation of the new approach or method in its categorization process. In addition, RG 1.201, Revision 1, states that all aspects of NEI 00-04, Revision 0, must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 6), describes an acceptable approach for determining whether the acceptability of the base PRA, in total or the parts, that is used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors. The guidance endorses the ASME/ANS PRA standard, ASME/ANS RA-Sa-2009 (Reference 7). Revision 2 of RG 1.200 provides guidance for determining the acceptability of a PRA by comparing the PRA to the relevant parts of ASME/ANS RA-Sa-2009 using a peer review process. The guidance stipulates that peer reviews are performed to assess PRA upgrades. A PRA upgrade is defined in ASME/ANS RA-Sa-2009 as "the incorporation into a PRA model of a new methodology, or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences."

RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 8), provides guidance on the use of PRA findings and risk insights in support of changes to a plant's licensing basis.

Revision 3 of RG 1.174, provides risk acceptance guidelines for evaluating the results of such evaluations.

NUREG 1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (Reference 22), provides guidance on how to treat uncertainties associated with PRA in risk-informed decisionmaking. The guidance fosters an understanding of the uncertainties associated with PRA and their impact on the results of the PRA and provides a pragmatic approach to addressing these uncertainties in the context of the decisionmaking.

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

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

3.2 Overview of the Categorization Process (NEI 00-04, Revision-0, Section 2)

Section 2 of NEI 00-04, Revision 0, states that the categorization process should include eight primary steps:

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

Section 3.1.1 of the LAR states that the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, will be implemented.

Further discussion of specific elements within the 10 CFR 50.69 categorization process is provided in the LAR. Elements of the categorization process for which the LAR provided clarity are bulleted below. A more detailed review of those specific elements in the categorization process is discussed in the applicable sections of this SE.

- Passive Characterization: Passive components are not modeled in the PRA. Therefore, a different method is used to assess the safety significance of these components, as described in Section 3.5.4 of this SE. The process used addresses those components that have only a pressure retaining function and the passive function of active components, such as the pressure/liquid retention of the body of a motor-operated valve (MOV).

- Qualitative Characterization: System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04, Revision 0. (Refer to Section 3.9 of this SE).

- Cumulative Risk Sensitivity Study: For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of LSS components results in acceptably small increases to CDF and LERF and meets the acceptance guidelines of RG 1.174, Revision 3. (Refer to Section 3.8 of this SE).

- 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. (Refer to Section 3.9 of this SE).

Attachment 1 of the LAR provides a list of steps/elements that will be included in the 10 CFR 50.69 programmatic procedures prior to the use of the categorization process. A more detailed review of the steps/elements for the programmatic procedures is discussed in Section 3.10 of this SE.

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

Regulation 10 CFR 50.69(c)(1)(ii) requires that licensees determine SSC functional importance using an integrated, systematic process for addressing initiating events (i.e., internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design-basis functions and functions credited for mitigation and prevention of severe accidents.

Section 3 of NEI 00-04, Revision 0, states that the assembly of plant-specific inputs involves the collection and assessment of the key inputs to the risk-informed categorization process. This includes design and licensing information, PRA analyses, and other relevant plant data sources. This step also includes the critical evaluation of plant-specific risk information to ensure that the PRA is modeled adequately to support the risk characterization of SSCs for this application. In addition, Section 4 of the NEI 00-04 guidance states, in part, "that the next step is the identification of system functions, including design-basis and beyond-design-basis functions identified in the PRA, and that system functions should be consistent with the functions defined in design-basis documentation and maintenance rule functions."

Furthermore, the guidance in Section 3 of NEI 00-04, Revision 0, summarizes the use of PRA if such PRA models exist, or, in the absence of quantifiable PRA, the use of other methods to evaluate risk include the FIVE methodology, SMA, IPEEE screening, and shutdown safety plan. The NRC staff acknowledges that elements of the categorization process are not always performed in chronological order and may be performed in parallel, such that, the systematic process for evaluating the plant-specific PRA may include other aspects of the categorization process (e.g., system selection, system boundary definition, identification of system functions, and mapping of components to functions) that is further discussed in Section 3.4 of this SE. The licensee's risk categorization process uses PRAs to assess risks from internal events (includes internal floods), and internal fire hazards. For the other applicable risk hazard groups, the licensee's process uses non-PRA methods for risk characterization, including the NSPM SMA to assess seismic risk and IPEEE screening to assess the risk from external hazards (i.e., high wind and external flood hazards) and other hazards.[2] To assess risk from shutdown operations, the licensee's categorization process uses the shutdown safety management plan. The NRC staff review of the quality and level of detail for the acceptability of the PRAs at the time of the submittal and non-PRA methods is provided in Sections 3.5.1 and 3.5.3 of this SE, respectively.

Regulation 10 CFR 50.69(c)(1)(v) requires that SSC categorization be performed for entire systems and structures, not for selected components within a system or structure. Section 3.1.1, "Overall Categorization Process," of the LAR, states, in part, that "NSPM will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201." The NRC staff finds that the process described in the LAR, as supplemented in letters dated March 13, 2019, and May 15, 2019 (References 2 and 37, respectively), is consistent with NEI 00-04 as endorsed by the NRC staff in RG 1.201, Revision 1, and capable of collecting and organizing information at the system level for defining boundaries, functions, and components.

3.4 System Engineering Assessment (NEI 00-04, Revision 0, Section 4)

Regulation 10 CFR 50.69(c)(1)(ii) requires, in part, the functions to be identified and considered in the categorization process include design basis functions and functions credited for mitigation and prevention of severe accidents. Revision 0 of NEI 00-04, includes guidance to identify all functions performed by each system and states that the IDP will categorize all system functions. All system functions include all functions involved in the prevention and mitigation of accidents and may include additional functions not credited as hazard mitigating functions depending on the system. The guidance in NEI 00-04 also includes consideration of interfacing functions. Section 4 of the NEI 00-04 provides guidance for circumstances when the categorization of a candidate LSS SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system. The guidance states, in part, "[i]n this case, the SSC will remain uncategorized until the interfacing system is categorized." Furthermore, Section 7.1 of the NEI 00-04 guidance states in part, "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC or part thereof should be assigned the highest risk significance for any function that the SSC or part thereof supports."

Section 2.2 of the LAR states, "[t]he safety functions include the design basis functions, as well as functions credited for severe accidents (including external events)." Section 3.1.1 of the LAR summarizes the different hazards and plant states for which functional and risk significant information will be collected. Section 3.1.1 of the LAR confirmed that the SSC categorization process documentation will include, among other items, system functions, identified and categorized with the associated bases, and mapping of components to support function(s). Therefore, the NRC staff finds that the process described in the LAR is consistent with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

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[2] Other hazards include any internal or external hazards that are not considered as part of the development of an internal events, internal flood, internal fire, seismic, high wind, or external flood PRA model using the applicable parts of the ASME/ANS PRA standard, as endorsed by the NRC.

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

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

- Internal Events
- Internal Fire Events
- Seismic Events
- External Hazards (e.g., high wind, external floods, etc.)
- Other Hazards
- Shutdown Events
- Passive Component Categorization[3]

Regulation 10 CFR 50.69(c)(1)(i) requires, in part, the use of PRA to assess risk from internal events as a minimum. The rule further specifies that the PRA used in the categorization process must be of sufficient quality and level of detail and subject to an acceptable peer review process. For the hazards other than internal events, internal flood, internal fire, seismic, other external hazards (e.g., high winds, external floods), and for the shutdown mode of operation, 10 CFR 50.69(b)(2) allows, and the guidance in NEI 00-04, Revision 0, summarizes the use of PRA, if such PRA models exist. Also, NEI 00-04, Revision 0, summarizes the use of other methods (e.g., FIVE, SMA, IPEEE, screening, and shutdown safety management plan) in the absence of quantifiable PRA as permitted by 10 CFR 50.69(b)(2).

Sections 3.1.1, 3.2.1, and 3.2.2 of the LAR describes that the categorization process uses PRA modeled hazards to assess risks for the internal events (includes internal flood) and internal fire hazards. For the other risk contributors, the process uses the following non-PRA methods to characterize the risk:

- Seismic: SMA performed for the NSPM IPEEE (Reference 9).
- External Hazards: Screening analysis performed for the IPEEE for high winds and external flood updated using Part 6 of ASME/ANS RA-Sa-2009, as endorsed by the NRC.
- Other Hazards: Screening analysis performed for the IPEEE for other hazards updated using Part 6 of ASME/ANS RA-Sa-2009, as endorsed by the NRC.
- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06.
- Passive Components: ANO-2 passive categorization methodology (Reference 26)

The approaches for using the modeled PRA hazards (i.e., IEPRA (Internal event), and FPRA (fire)) and non-PRA methods by the licensee to assess internal and external hazards and shutdown risk are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. The non-PRA method for the categorization of passive components is consistent with the ANO-2 methodology for passive

[3] Deviation: Methodology proposed for the categorization of passive components not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1, but approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2).

components approved for risk-informed safety classification and treatment for repair/replacement activities in class 2 and 3 moderate and high energy systems.

The NRC staff's review of the information provided by the licensee for the modeled PRA hazards and non-PRA methods for acceptability is provided in the following subsections of this SE: PRA modeled hazards in Sections 3.5.1 through 3.5.2.2 and the non-PRA methods in Sections 3.5.3 through 3.5.4. The NRC-approved ANO-2 methodology for passive components is not included in the NEI 00-04, Revision 0, guidance as endorsed by the NRC staff. A discussion of the ANO-2 methodology for use in the SSC categorization process is provided in Section 3.5.4 of this SE.

3.5.1 Evaluation of PRA Acceptability to Support the Categorization Process

Regulation 10 CFR 50.69(c)(i) requires, in part, a licensee's PRA must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. 10 CFR 50.69(b)(2)(iii) further requires that the results of the peer review process conducted to meet 10 CFR 50.69(c)(1)(i) criteria be submitted as part of the application.

3.5.1.1 Scope of PRA

The NSPM PRAs are comprised of a full power, Level 1, IEPRA (includes internal flood) and a full power, Level 1, FPRA, both of which evaluate the CDF and LERF risk metrics.

Section 3.3 of the LAR states that for the IEPRA and FPRA models described in Section 3.2 of the LAR, the models have been assessed against RG 1.200, Revision 2, and are consistent with NRC Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation" (Reference 13). Furthermore, LAR Section 3.3 states, in part, "finding closure reviews were conducted on the internal events, including internal flooding, and fire PRA models in August and October 2017, respectively."

Attachment 3 of the LAR provided the open findings/open items, along with the dispositions to address impact on the 10 CFR 50.69 program. Subsequently, the NRC staff review of the PRA(s) acceptability considered the Appendix X process performed, information provided in Attachment 3 and Attachment 6 of the LAR, in addition to applicable information provided in the staff issuance of the SE for the NSPM adoption of Technical Specifications Task Force (TSTF) Traveler TSTF-425 (References 11 and 29, respectively). The NRC staff finds that the information provided in the LAR to support the NRC staff review of the IEPRA and FPRA for technical acceptability was sufficient and, therefore, meets the requirements set forth in 10 CFR 50.69(b)(2)(iii).

By letter dated January 31, 2019, the NRC staff issued a RAI to further assess the PRA acceptability of the IEPRA and FPRA models for consistency with RG 1.200, Revision 2, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. A more detailed staff review of the PRA acceptability for the IEPRA and the FPRA models are provided in Sections 3.5.1.3 and 3.5.1.4 of this SE, respectively.

Aspects considered by the NRC staff to evaluate the scope of the IEPRA and FPRA include: (1) peer review history, (2) the Appendix X, independent assessment process, (3) credit for FLEX in the PRA(s), and (4) assessment of assumptions and approximations. The NRC staff review of these aspects are provided in Subsections 3.5.1.2 through 3.5.1.5 of this SE.

3.5.1.2 IEPRA Peer Review History

The LAR states that a full-scope peer review of the NSPM IEPRA was performed in April 2013, and a focused-scope peer review was performed in April 2017, to review the convolution analysis portion of the PRA model. In the NSPM LAR dated December 19, 2017 (Reference 11), the licensee states that the IEPRA was peer reviewed using NEI 05-04, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard" (Reference 14), RG 1.200, Revision 2, and ASME/ANS RA-Sa-2009.

The NRC staff finds that the NSPM IEPRA has been peer reviewed against the currently endorsed PRA standard consistent with RG 1.200, Revision 2.

3.5.1.3 Internal Fire PRA Peer Review History

The LAR states that a full-scope peer review of the NSPM FPRA was performed in March 2015. Section 2.3.2 of the LAR dated December 19, 2017, to support the adoption of TSTF-425, states that, "[t]he FPRA Peer Review was performed [...] applying the NEI 07-12, 'Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines,' [...] process, the ASME PRA Standard (ASME/ANS RA-Sa-2009) and RG 1.200, Revision 2." A focused-scope peer review of the NSPM FPRA was performed in December 2016 to assess the incorporation of new methods in the FPRA. Subsequent to the peer reviews, an independent assessment was performed in October 2017. (Note: Discussion of the independent assessment is provided in Section 3.5.1.2 of this SE.) Attachment 3 of the LAR provided the remaining open F&Os for the FPRA model along with the disposition and resolution for each open finding and/or open item from the self-assessment.

The NRC staff reviewed the summary of the 14 open finding-level F&Os for the FPRA model, and requested additional information in letter dated January 31, 2019. In RAI 01.a through h, the NRC staff requested additional justification to support the dispositions of the F&Os provided in Attachment 3 of the LAR. The NRC staff sought confirmation that there is insignificant or no adverse impact on the 10 CFR 50.69 categorization process or requested a mechanism to ensure the activities and changes associated with the F&Os will be completed, appropriately reviewed, and any issues resolved prior to implementation of 10 CFR 50.69 categorization process.

Attachment 3 of the LAR for F&O 2-5, which is associated with SR IGN-A7, notes that the peer review team identified that the fire influencing factors assigned in the FPRA model were based on engineering judgment and a set of rules documented in the FPRA documentation. The independent assessment team's (i.e., the F&O closure review team) recommendation for the resolution to close F&O 2-5 provided in Table 2-1 of Attachment 2 of the NSPM LAR dated December 19, 2017, states, in part, "better justification of application of a 'very low' factor in two compartments [8 and 33] needs to be provided." In RAI 01.a, the NRC staff noted that FAQ 12-0064, "Close-Out of National Fire Protection Association 805 Frequently Asked Question 12-0064 on Hot Work/Transient Fire Frequency Influence Factors" (Reference 20), provides guidance for consideration for applying influencing factors in an FPRA model. The response to RAI 01.a.iii provided by letter dated March 13, 2019, states, "[t]he MNGP internal fire PRA method for applying transient fire influencing factors follows PRA . . ." The response further states, "[i]t was determined that these compartments are not justified at a "very low" transient storage factor, and therefore they have been updated to a justifiable "low" value of 1." The NRC staff reviewed the licensee's response and concludes the use of the influencing factor of 1.0 for fire compartments 8 and 33 is consistent with the guidance provided in FAQ 12-0064.

Therefore, the NRC staff finds F&O 2-5 to be resolved for the 10 CFR 50.69 SSC categorization process.

In Attachment 3 of the LAR for F&O 3-6, the independent assessment team identified approximately 10 components that should be treated as failed in a fire that were not treated as such in the FPRA model. In Attachment 3 of the LAR, the licensee's disposition to F&O 3-6 states, "[t]he Closure Review Team recommendation will be addressed by including the specified basic events in the fire failed events flag file." In RAI 01.b, the NRC staff noted that the disposition in the LAR did not explain how there will be assurance that the cited basic events will be added to the fire-failed events flag file, prior to implementation of the 10 CFR 50.69 program to address this F&O and its resolution. The response to RAI 01.b.iii provided by letter dated March 13, 2019, states that "[t]he MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update," and that the,"[b]asic events that have been identified as not credited in the internal fire PRA model have been failed via a flag file or an equivalent approach." The NRC staff reviewed the response to RAI 01.b.iii and concludes the licensee appropriately addressed the treatment of basic events in the FPRA for SSCs that were not selected in the Equipment Selection technical element in ASME/ANS RA-Sa-2009 by setting those basic events to a failed state in the PRA model. Therefore, the NRC staff finds F&O 3-6 to be resolved for the 10 CFR 50.69 SSC categorization process.

In Attachment 3 of the LAR for F&O 4-11, the peer review team identified, using an initial ambient air temperature of 20 degrees Celsius in fire models, is not appropriate for fire zones that are not temperature controlled such as the diesel generator building, and areas of the reactor building. The disposition to F&O 4-11 states, in part, "[t]he Closure Review Team recommendation will be addressed by revising the fire models using expected plant ambient temperatures for each fire zone." RAI 01.c, noted that the disposition did not explain how it will be ensure that the FPRA model will be updated using expected plant ambient temperatures that are bounding temperatures to account for days when the outdoor temperature is high prior to implementation of the 10 CFR 50.69 program to address this F&O and its resolution. The response to RAI 01.c.ii provided by letter dated March 13, 2019, states that "[t]he MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update," and the, "fire modeling calculations have been updated to use a bounding ambient temperature." The NRC staff reviewed the response to RAI 01.c.ii and concludes the licensee appropriately addressed the treatment of initial ambient air temperature assumptions for fire zones modeled in the NSPM FPRA that are not temperature controlled and replaced the 20 ^{O}C initial ambient air temperature assumption with a bounding initial ambient temperature assumption. Therefore, the NRC staff finds F&O 4-11 to be resolved for the 10 CFR 50.69 SSC categorization process.

In Attachment 3 of the LAR for the resolution of F&O 4-20, states that additional verification and documentation of the main control board configuration for sensitive electronics was determined to be required to fully resolve this F&O. The LAR states that the disposition of the F&O closure team's recommendation is not expected to have any impact on CDF or LERF since the recommendations are associated with documentation changes to better explain modeling rationale. RAI 01.d noted that verification is not a documentation issue when configurations are potentially identified that result in modelling changes to the PRA that could impact the application. The response to RAI 01.d.iv provided by letter dated March 13, 2019, states, "[t]he MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update," and that, "[a] review of sensitive electronics has been performed to confirm that FAQ 13-0004 guidance was applied to all closed cabinets containing sensitive electronics. Additional fire modeling has been performed to address the possible impacts to sensitive electronics for open cabinet configurations." The NRC staff reviewed the response to RAI 01.d and finds the licensee appropriately addressed the treatment of the main control board configuration for sensitive electronics because additional fire modeling analysis addressing possible impacts to sensitive electronics for open cabinet configurations was performed, which is consistent with the guidance in FAQ 13-0004. The NRC staff finds F&O 4-20 to be resolved for the 10 CFR 50.69 SSC categorization process.

In Attachment 3 of the LAR for the resolution to F&O 4-33, the independent assessment team states, in part, the results may not be bounding for cable trays in wall or wall-corner locations and verification that FLASH-CAT results were not used for such configurations needs to be performed. RAI 01.e, noted that it was unclear how the fire scenarios that need detailed fire modelling in the FPRA model were determined (i.e., considered) and the overall impact on the PRA results used to support risk-informed categorization to address this F&O and its resolution. The response to RAI 01.e provided by letter dated March 13, 2019, states, "[t]he MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update," and "[w]all and corner effects have been applied to applicable Consolidated Fire and Smoke Transport (CFAST) heat release rates." The NRC staff reviewed the response to RAI 01.e.iii and finds the licensee appropriately addressed bounding assumptions for cable trays in wall or wall-corner locations by applying wall and corner effects to relevant CFAST heat release rates. Therefore, the NRC staff finds this F&O 4-33 to be resolved for the 10 CFR 50.69 categorization process.

The independent assessment teams concluded several of the recommended actions for resolution of the F&Os provided in Attachment 3 of the LAR had not been corrected in the PRA(s). The resolutions for the F&Os are associated with F&Os 6-3, 6-9, 6-11, and 7-3. For the dispositions of these F&Os, the LAR states that the corrections are not expected to have a significant impact on total CDF or LERF, and the effect of the individual and the cumulative changes to the PRA or the results to support risk-informed categorization. RAI 01.f requested a proposed mechanism to ensure that all the recommended actions identified by the independent assessment team for these F&Os will be resolved at Capability Category (CC) II for the applicable supporting requirements and incorporated into the FPRA model and/or documentation prior to implementation of the 10 CFR 50.69 program. The response to RAI 01.f provided by letter dated March 13, 2019, states, "[t]he MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update." The licensee confirmed in the RAI response for F&O 6-3 that the ignition frequency assignment for the battery chargers (i.e., D70, D80, and D90) has been corrected in the FPRA model. For F&O 6-9, the response states, "RHR [residual heat removal] Pump B maintenance unavailability and CCF events have been included in alternate shutdown scenarios." For F&O 6-11, the response states that, "failure probabilities have been assigned to non-flag basic events." For F&O 7-3, the response states, "[m]odeling of reactor level instrumentation has been updated to reflect redundancy and procedures for loss of level instrumentation," and notes that this is a change from the expected resolution provided in the original LAR. The NRC staff reviewed the response to RAI 01.f and finds the licensee appropriately addressed the recommended actions associated with each of the four F&Os referenced above and found the resolutions to be consistent with the ASME/ANS RA-Sa 2009 as endorsed by the NRC in RG 1.200, Revision 2, to meet CC II for the applicable surveillance requirements (SRs).

The disposition for F&O 7-4 states, "[t]he Closure Review Team recommended actions will be addressed by performing thermal hydraulic MAAP [Modular Accident Analysis Program] analysis to determine the success criteria for the opening of two or more SRVs [safety relief valves] [and that] the fault tree model will be revised to reflect the determined success criteria."

RAI 01.g.iii requested that a mechanism be proposed that ensures F&O 7-4 will be resolved at CC II for the applicable SRs prior to implementation of the 10 CFR 50.69 categorization process. The response to RAI 01.g.iii provided by letter dated March 13, 2019, states, "[t]he MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update," and that, "[s]purious opening or failure to reclose of two or more SRVs has been updated to more accurately reflect the expected plant response." The NRC staff reviewed the response to RAI 01.g and finds the licensee appropriately addressed the Independent Assessment Team's recommended closure of F&O 7-4 to meet the associated SRs at CC II for the 10 CFR 50.69 risk-informed application because the success criteria used in fire PRA for two or more failed open SRVs has been updated consistent with the MAAP analysis and to reflect the as-built as-operated plant.

The disposition for the F&O FO-1 states, in part, "[t]he Closure Review Team recommendation will be addressed by reviewing the cable heat soak fire modeling that credits ventilation limited burning and credit for ventilation limited burning will be removed." RAI 01.h, noted that it is not clear that removing credit for ventilation-limited burning from the cable heat soak fire models would have an adverse and/or insignificant impact on the PRA results used to support risk-informed categorization (e.g., mask the importance measures for other SSCs). The response to RAI 01.h.iii provided by letter dated March 13, 2019, states, "[t]he MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update," and that, "[t]he fire modeling has been updated such that the baseline (ventilation limited case) and sensitivity cases (non-ventilation limited case) for verification and validation (V&V) have been evaluated with the heat soak model when the damage criteria is exceeded, and the limiting result has been used in the internal fire PRA." The NRC staff finds a heat soak model was performed to evaluate the ventilation configurations when the damage criteria was exceeding and incorporated the limiting results into the FPRA. Therefore, the F&O has been resolved to meet the associated SRs (i.e., FSS-D3 and FSS-D4) at CC-II for the 10 CFR 50.69 risk-informed application.

The NRC staff reviewed the FPRA peer review results, the resolutions for the peer review finding-level F&Os, and the supplemental responses to the RAIs (References 2 and 37), and finds as required by 10 CFR 50.69(b)(2)(ii) that the application contains sufficient information to support the categorization of SSCs . The NRC staff finds that the identified errors and weaknesses in the FPRA have been resolved prior to implementation of the 10 CFR 50.69 categorization process. Therefore, the NRC staff concludes that the FPRA meets the requirements in 10 CFR 50.69(c)(1)(i).

3.5.1.4 Appendix X, Independent Assessment Process for F&O Closure

Section X.1.3 of Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13[4] (Reference 15), provides guidance to perform an independent assessment for the closure of F&Os identified from a full scope or focused scope peer review. By memorandum dated May 3, 2017,[5] the NRC staff stated that Appendix X provides an acceptable process for F&O closure. The independent

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[4] *Errata;* Anderson, V. K., Nuclear Energy Institute, letter to Stacey Rosenberg, U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, 'Close-Out of Facts and Observations,'" dated February 21, 2017 (ADAMS Package Accession No. ML17086A431).

[5] Giitter, J., and Ross-Lee, M. J., U.S. Nuclear Regulatory Commission, letter to Mr. Greg Krueger, Nuclear Energy Institute, "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).

assessment process as described in Appendix X includes criteria for: (1) the qualifications of independent assessment team members, (2) pre-review activities, (3) on-site review activities, and (4) post-review activities. The Appendix X process assures that closure of the F&Os are met at CC II for the applicable SR in the ASME/ANS Ra-SA-2009 PRA Standard as endorsed by RG 1.200, Revision 2.

LAR Section 3.3 states, in part:

> 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" as accepted by the NRC in [Reference 16] ... Note: All internal events PRA model, including internal flooding, findings were closed during the finding closure review.

The NRC staff's memorandum dated May 3, 2017, stated that the NRC intends to periodically conduct audits of licensee's implementation of the Appendix X F&O closure process. Consistent with this statement, the NRC staff performed an electronic audit (Reference 25). The NRC audit did not identify any discrepancies associated with the process for the independent assessment performed for closure of F&Os performed in August 2017, and October 2017, for the NSPM IEPRA and FPRA models (Reference 24), respectively. Furthermore, Section 3.3 of the LAR discussed that all finding-level F&Os from the April 2013, peer review performed for the IEPRA were closed as a result of the October 2017, independent assessment, therefore, no F&Os related to the NSPM IEPRA were submitted in the LAR. Section X.2 of the Appendix X guidance states, in part, "[o]nce an F&O is closed out, the utility is not required to present and explain them in peer reviews, NRC submittals, or other requests excluding NRC audits."

3.5.1.5 Credit for FLEX Equipment

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (Reference 18), provides the NRC's staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decisionmaking in accordance with the guidance in RG 1.200, Revision 2.

The response to RAI 06.a provided by letter dated March 13, 2019, states, ". . . FLEX equipment and strategies will not be credited in the PRA models for the categorization process . . . ." The NRC staff finds that the NSPM IEPRA does not credit FLEX equipment for the SSC categorization process. Therefore, no new methods and/or upgrades with respect to FLEX equipment were inadvertently incorporated into the NSPM IEPRA without a peer review. The licensee treatment of FLEX equipment is consistent with the ASME/ANS RA-Sa-2009 PRA standard as endorsed by the NRC.

3.5.1.6 Assessment of Assumptions and Approximations

Table A-1 "Staff Position on ASME/ANS RA-Sa-2009 Part 1, General Requirements for an At-Power Level 1 and LERF PRA", of RG 1.200, Revision 2, includes the staff position for Section 1-6.1 of ASME/ANS RA-Sa-2009. Table A-1 states, in part, that "the peer review shall also assess the appropriateness of the assumptions."

RG 1.174, Revision 3, cites NUREG-1855, Revision 1, as related guidance that includes changes associated with expanding the discussion of sources of uncertainties. NUREG-1855, Revision 1, explicitly states, in part, "RG 1.200 [NRC 2009] and the PRA consensus standard published by ASME and ANS (ASME/ANS, 2009) each recognize the importance of identifying and understanding uncertainties as part of the process of achieving acceptability in a PRA, and these references provide guidance on this subject."

Section 3.2.7 of the LAR states that guidance in NUREG-1855, Revision 0, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," and Electric Power Research Institute (EPRI) TR [topical report]-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" (Reference 23), was used to identify, characterize, and screen model uncertainties.

Identification of Assumptions and Sources of Uncertainty for Base PRA(s)

To assess the assumptions and sources of uncertainty identified by NSPM for the base IEPRA, RAI 02.a.ii, requested that a description be provided of how the key assumptions and sources of uncertainties provided in Attachment 6 of the LAR were identified from the initial comprehensive list of PRA model(s) (i.e., base model) sources of uncertainty and assumptions. The response to RAI 02.a.ii provided by letter dated March 13, 2019, states, "[s]ince the time of the LAR submittal, the internal events and fire PRAs have been updated to include the EPRI-identified generic sources of uncertainty as documented in EPRI TR-1016737 and EPRI TR-1026511. Both modeling uncertainty and completeness uncertainty sources were examined. Each PRA includes an evaluation of the sources of uncertainty for the base case models using the approach that is consistent with the ASME/ANS RA-Sa-2009 requirements for identification and characterization of uncertainties and assumptions." Section 1.3 of NUREG-1855, Revision 1, states, in part, "[a]lthough assumptions and approximations made on the level of detail in a PRA can influence the decisionmaking process, they are generally not considered to be model uncertainties because the level of detail in the PRA model could be enhanced, if necessary. Therefore, methods for identifying and characterizing issues associated with level of detail are not explicitly included in NUREG-1855; they are, however, addressed in EPRI reports 1016737 and 1026511." The NRC staff finds that the licensee used approved NRC methods identified in EPRI TR-1016737 and EPRI TR-1026511 to identify the assumptions and sources of uncertainties for the base IEPRA and FPRA, which is consistent with the NUREG-1855, Revision 1, and, therefore, meets the intent of RG 1.201, Revision 1.

Key Assumptions and Sources of Uncertainty for the Risk-Informed Application

RG 1.200, Revision 2, Section 3.3.2, provides guidance that states, in part, "[f]or each application that calls upon this RG, the applicant identifies the key assumptions and approximations relevant to that application." This will be used to identify sensitivity studies as input to the decisionmaking associated with the application."

Section 4.2, of RG 1.200, Revision 2, provides guidance that identifies specific information to be included in the licensee submittal to demonstrate the technical adequacy of the PRA used in an application is of sufficient quality. The identified information includes the identification of the key assumptions and approximations relevant to the results used in the decisionmaking process.

Identification of Key Assumptions and Sources of Uncertainty

RAI 02.a requested the licensee provide a detailed summary of the process used to identify the key assumptions and sources of uncertainty provided in Attachment 6 of the LAR, including a discussion of how the process is consistent with NUREG-1855, Revision 1. Specifically, the NRC staff requested the licensee provide a basis to justify the appropriateness of any deviations for use in the 10 CFR 50.69 categorization process (e.g., exclusion/consideration of EPRI TR-1026511). The response to RAI 02.a.ii provided by letter dated May 15, 2019, states, in part:

> "[a]t the time of the original LAR submittal, the identification of those base PRA uncertainties that were important for 10 CFR 50.69 categorization was performed based on expert judgement. To enhance the traceability of this evaluation, an additional review was performed. [...] The updated evaluation process includes a review of the Internal Events and Fire PRA Uncertainty Notebooks to determine which uncertainties could impact the 10 CFR 50.69 categorization process results.

Further response to RAI 02.a.ii also provided a table to replace the existing table in Attachment 6 of the LAR in its entirety. The response also confirmed that for the NSPM uncertainty analysis, some uncertainties and assumptions were screened based on their use in a consensus[6] method. The NRC staff reviewed the response to RAI 02.a.i and finds that the licensee's process used to identify the key assumptions and sources of uncertainty presented in the supplemental response to RAI 02 is consistent with NUREG-1855, Revision 1. The licensee performed updated evaluations that considered the generic sources of model uncertainty in EPRI TR-1016373 and EPRI TR-1026511 and screening based upon the identification of consensus methodologies used in the PRAs. The NRC finds that the identified key assumptions and sources of uncertainty provided in the table in revised Attachment 6 provided by the supplemental letter dated May 15, 2019, are consistent with RG 1.200, Revision 2, and NUREG-1855, Revision 1.

Treatment of Key Assumptions and Sources of Uncertainty

Section 5 of the NEI 00-04 guidance as endorsed by the NRC states, in part:

> An analysis of the impacts of parametric uncertainties on the importance measures used in this categorization process was performed and documented in EPRI TR-1008905, Parametric Uncertainty Impacts on Option 2 Safety Significance Categorization [...]. The conclusion of this analysis was that the importance measures used in combination with identified set of minimum sensitivity studies adequately address parametric uncertainties.

Furthermore, the guidance in NEI 00-04, Revision 0, specifies sensitivity studies should be conducted for each PRA model to ensure that PRA assumptions and sources of uncertainty (e.g., human error, CCF, and maintenance probabilities) do not mask the SSC(s) importance. Tables 5-2 and 5-3 of NEI 00-04 provides several recommended sensitivity studies for the IEPRA and FPRA models.

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[6] Per NUREG-1855, Revision 1, a consensus model is a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group.

RAI 03 requested the licensee identify which of the sensitivity studies outlined in Section 5 of the NEI 00-04 is directly applicable for the following three sources of uncertainty described in Attachment 6 of the LAR:

- ignition counting in the FPRA model,
- fire cable selection for the FPRA model, and
- heat release rates specified in NUREG/CR-6850 for the FPRA model.

The response to RAI 03 provided by letter dated March 13, 2019, states, "[t]he most recent update of the MNGP fire PRA model re-evaluated the sources of uncertainty based on the most recent industry guidance. The current methodologies used for developing the MNGP fire PRA model use industry accepted approaches that meet and have been peer reviewed against the ASME/ANS RA-Sa-2009 Standard [...] at Capability Category II." The NRC staff reviewed the response to RAI 03 and concludes the licensee addressed the key sources of uncertainty by using consensus methodologies. , The key sources of uncertainty related to the FPRA were subsequently removed from the Attachment 6 table provided in the supplemental letter dated May 15, 2019, therefore, eliminating the need to explore an alternative hypothesis and the associated uncertainty is treated in the quantification of the PRA as a parameter uncertainty. The staff finds the removal of the above FPRA sources of uncertainty provided in Attachment 6 of the LAR is consistent with NUREG-1855, Revision 1, and acceptable for the 10 CFR 50.69 categorization process.

The revised table provided in supplemental letter dated May 15, 2019, identified three key sources of model uncertainty and related assumptions related to very small loss-of-coolant accidents (LOCAs), minimum values for individual pre- or post-initiator human error probabilities (HEPs); and assumptions about pipe sizes and lengths for determining the frequency of certain internal flood initiating events. The licensee's disposition of the item related to very small LOCAs states they will perform a sensitivity study, "that addresses very small LOCAs in accordance with NEI 00-04, Table 5-2, to determine if there are any changes in the HSS/LSS determination." The licensee's disposition of the item related to minimum values for individual pre- or post-initiator HEPs states, "[s]ensitivity studies in accordance with NEI 00-04, Table 5-2, will be performed to evaluate the potential impact of variations in HEP values." For the third source of uncertainty provided for the disposition, the licensee states, "this item does not represent a key source of uncertainty for 50.69 calculations." To confirm impact to the 10 CFR 50.69 SSC categorization process for the third source of uncertainty, the licensee reviewed the internal flood events where pipe lengths could not be validated with a walkdown and determined the pipe length estimates used were reasonable based upon room size using the isometric drawings. Consistent with NUREG-1855, Revision 2, treatment for key assumptions and key sources of uncertainty (e.g., sensitivity studies) are considered in the context of the decision, and for this risk-informed application some are dependent upon which SSC is being categorized at the time of performing the risk analysis. The NRC staff finds the licensee will perform additional sensitivity studies for the two remaining sources of uncertainty consistent with Table 5-2 in NEI 00-04 for the IEPRA .

3.5.1.7 <u>Summary of Internal Events PRA and Internal FPRA Acceptability</u>

The NRC staff finds the information provided by the licensee appropriately identified the technical elements of ASME/ANS RA-Sa-2009 that were not met and resolved the remaining open finding-level F&Os for the 10 CFR 50.69 risk-informed application. The NRC staff concludes that the IEPRA and FPRA are consistent with NEI 00-04, as endorsed by RG 1.201, Revision 1. Therefore, the staff concludes that the IEPRA and FPRA meets the requirements set forth in 10 CFR 50.69(c)(1)(i).

3.5.2 Importance Measures and Integrated Importance Measures

Regulation 10 CFR, paragraph 50.69(c)(1)(i), states, in part, that a licensee should "[c]onsider the results and insights from the plant-specific PRA." These requirements are met, in part, by using importance measures and sensitivity studies consistent with the ASME/ANS PRA standard, ASME/ANS RA-Sa-2009, as endorsed in RG 1.200, Revision 2. RG 1.200, Revision 2, states, in part, that:

> Methods such as importance measure calculations (e.g., Fussell-Vesely Importance[7] [F-V], risk achievement worth[8] [RAW], risk reduction worth, and Birnbaum Importance) are used to identify the contributions of various events to the estimation of CDF for both individual sequences and the total CDF [i.e., both contributors to the total CDF, including the contribution from the different hazard groups and different operating modes (i.e., full- and low-power and shutdown) and contributors to each contributing sequence are identified].
>
> The results of the Level 2 PRA are examined to identify the contributors (e.g., containment failure mode, physical phenomena) to the model estimation of LERF or LRF [large release frequency] for both individual sequences and the model as a whole.

NEI 00-04, Revision 0, provides guidance where the F-V and RAW importance measures are obtained for each component and each PRA modeled hazard (i.e., separately for the IEPRA and FPRA) and the values are then compared to specified criterion as follows:

- Components which have importance measures values that exceed the risk criteria (i.e., F-V greater than 0.005, RAW greater than 2, CCF RAW greater than 20) are assigned candidate safety-significant.

Section 5.1 of NEI 00-04, Revision 0, recommends that a truncation level of five orders of magnitude below the baseline CDF (or LERF) value should be used for calculating the F-V risk importance measures. The guidance also recommends that the truncation level used should be sufficient to identify all functions with a RAW value greater than 2.

3.5.2.1 Importance Measures

Section 3.1.1 of the LAR states, in part, that "[t]he IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address . . . the

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[7] *Fussell-Vesely importance measure*: for a specified basic event, Fussell-Vesely importance is the fractional contribution to the total of a selected figure of merit for all accident sequences containing that basic event. For PRA quantification methods that include nonminimal cutsets and success probabilities, the Fussell-Vesely importance measure is calculated by determining the fractional reduction in the total figure of merit brought about by setting the probability of the basic event to zero.

[8] *risk achievement worth importance measure*: for a specified basic event, risk achievement worth importance reflects the increase in a selected figure of merit when an SSC is assumed to be unable to perform its function due to testing, maintenance, or failure. It is the ratio or interval of the figure of merit, evaluated with the SSC's basic event probability set to one, to the base case figure of merit.

interpretation of risk importance measures . . . ." Additionally, Section 3.2.3 of the LAR states that because the seismic risk is assessed via a SMA, which is a screening tool, importance measures are not used to determine safety significance.

A more detailed review of the characterization for seismic risk using a non-PRA method is discussed in Section 3.5.3.1 of this SE. The licensee's SSC categorization process includes the use and treatment of importance measures consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, therefore, the NRC staff finds that the licensee will consider the results and insights from the plant-specific PRAs, thus, meeting the intent of 10 CFR 50.69(c)(1)(i).

3.5.2.2 Integrated Importance Measures

Section 5.6 of NEI 00-04, Revision 0, titled, "Integral Assessment," discusses the need for an integrated computation using the available importance measures. It further states, in part, that the, "integrated importance measure essentially weights the importance from each risk contributor (e.g., internal events, fire, and SPRA models) by the fraction of the total CDF [or LERF] contributed by that contributor." The guidance also provides formulas to compute the integrated F-V, and integrated RAW.

The NRC recognizes there are variations of PRA modeling practices associated with accident sequences and computer software applications (e.g., FRANC, CAFTA, Phoenix) used to support the quantification of both CDF and LERF across the multiple PRA hazards. The NRC also recognizes that, given these variations of PRA modeling practices, there is a potential to inadvertently introduce a deviation from the computations for F-V and RAW provided in the guidance in NEI 00-04, Revision 0.

RAI 08 requested confirmation that the importance measures generated for use in the 10 CFR 50.69 categorization process are consistent with the guidance in NEI 00-04, Revision 0, and that those importance measures do not inadvertently introduce a deviation from the computations for F-V and RAW provided therein, as endorsed in RG 1.201, Revision 1. RAI 08.a specifically requested confirmation whether the PRA model to be used in the 10 CFR 50.69 categorization process is an integrated one-top model that combines all modeled PRA hazards and if the integrated one-top model includes accident sequence(s) modeling to support quantification of both CDF and LERF.

The response to RAI 08.a provided by letter dated March 13, 2019, confirmed, "[t]he MNGP one-top model will be used for the 50.69 categorization process." Furthermore, RAI 08.a.i requested the licensee discuss the process used to validate and confirm the integration of the PRA hazards into a one-top model to ensure that after the PRA model change was performed, SRs QU-F2 and SR FQ-F1 continue to be met (e.g., cut set reviews, identification of non-minimal cut sets, peer review). The response to RAI 08.a.i provided by letter dated March 13, 2019, states, ". . . the MNGP Internal Events and Fire PRA models were developed as separate models with separate documentation that maps each associated supporting requirement (SR) to a section within the documentation. Each model update includes quantification with non-applicable initiators eliminated, including review of quantification results per the applicable SRs. Each model update also includes applicable updates to the affected model documentation including the SR mapping to ensure that the ASME/ANS RA-Sa-2009 standard is still met for all applicable SRs." The NRC staff reviewed the licensee's response to RAI 08.a.i and finds the licensee's process for validating and confirming the results of their PRA models following a change appropriately addresses ASME/ANS RA-Sa-2009, as endorsed in RG 1.200, Revision 2.

RAI 08.a.ii requested the licensee provide a discussion of how the individual importance measures (i.e., F-V and RAW) for the PRA one-top all hazards model are derived from the one-top model, and justify why the importance measures generated do not deviate from the guidance in NEI 00-04, Revision 0, including justification for the appropriateness of importance measures that are computed based on a deviation from the guidance in NEI 00-04, Revision 0. The response to RAI 08.a.ii provided by letter dated March 13, 2019, stated, "[t]he integrated importance measures will be performed manually in accordance with NEI 00-04, Section 5.6, until such time as variable parameters between hazards models are aligned (e.g., truncation, HEP minimums) and the integrated importance measures from the one-top model can be shown to be numerically equivalent. Quantification of a combined one-top model accounts for the overall importance directly because the calculated F-V or RAW is based on the impact on all hazards." The NRC staff reviewed the licensee's response to RAI 08.a.ii and finds that the licensee's process for deriving importance measures is acceptable because NSPM will perform manual quantification of the integrated importance measures in accordance with NEI 00-04, Section 5.6, until such time as variable parameters between hazards models are aligned (e.g., truncation, HEP minimums) and the integrated importance measures from the one-top model can be shown to be numerically equivalent. 10 CFR 50.69(e) and (f) allow for routine maintenance and update of the PRA(s).

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

As required by 10 CFR 50.69(c)(1)(ii), SSC functional importance must be determined using an integrated, systematic process for addressing initiating events, SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents.

As described in Sections 3.2.3, 3.2.4, and 3.2.5, of the LAR, the NSPM categorization process uses the following non-PRA methods, respectively:

- SMA performed for the IPEEE
- Screening analysis performed for the IPEEE for external hazards (e.g., high wind, external flood)
- Screening analysis performed for the IPEEE for other hazards;
- Safe shutdown risk management program consistent with NUMARC 91-06.

3.5.3.1 Seismic Risk

NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, states, in part:

> In the event an SMA is used, the categorization process is, once again, more conservative (i.e., designed to identify more SSCs as safety-significant). This is due to the fact that the SMA analysis is a screening tool. As a screening tool, importance measures are not available to identify safety significance. The NEI 00-04 approach identifies all system functions and associated SSCs that are involved in the seismic margin success paths as safety-significant. This measure

of safety significance assures that the SSCs that were required to maintain low seismic risk are retained as safety-significant.

Section 3.2.3 of the LAR proposes using the SMA performed for the IPEEE in response to Generic Letter (GL) 88-20, Supplement 4. As discussed above, the SMA is a screening method that does not quantify CDF or LERF values. For the NSPM SMA, the EPRI SMA method described in EPRI NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," (Reference 30), for the IPEEE was used. The SMA method includes the development of the seismic Safe Shutdown Equipment List (SSEL) which contains the components that would be needed during and after a seismic event. The SSEL identifies one preferred and one alternate path capable of achieving and maintaining safe shutdown conditions for at least 72 hours following an earthquake.

Section 3.2.3 of the LAR states, "[a]n 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." The LAR further states that "[t]he 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."

Based upon the NRC staff review of the NSPM response to GL 88-20, Supplement 4, and the additional evaluation performed by the licensee to update the SMA to the as-built, as operated configuration, the NRC staff finds the licensee's use of the SMA to assess seismic risk acceptable. The LAR, as supplemented, is consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and, therefore, meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

3.5.3.2 <u>Non-Seismic External Hazards and Other Hazards</u>[9]

Non-seismic external hazards were initially evaluated by the licensee during the IPEEE. This hazard category includes all non-seismic external hazards such as high winds and external floods. Other hazards, such as those hazards listed in Appendix 6-A of ASME/ANS RA-Sa-2009, were also evaluated by the licensee during the IPEEE. Section 3.2.4 of the LAR, states, in part, that "[a]ll 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." Section 5.4 of NEI 00-04, Revision 0, provides guidance on the treatment of such non-seismic external hazards and other hazards in the 10 CFR 50.69 categorization process.

The NRC staff's previous review of the NSPM IPEEE (Reference 10), found that the external hazard analysis used a progressive screening approach consistent with NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" (Reference 31), and concluded that: "(1) the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities from external events, and (2) the NSPM IPEEE has met the intent of Supplement 4 to GL 88-20 and the resolution of specific generic safety issues discussed in this SER [safety evaluation report]."

[9] Other hazards include any internal or external hazards that are not considered as part of the development of an internal events, internal flood, internal fire, seismic, high wind, or external flood PRA model using the applicable parts of the ASME/ANS PRA standard, as endorsed by the NRC.

RAI 05.a requested that the licensee identify the external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6. The response to the RAI provided a list of the hazards that will be evaluated for future SSC categorization consistent with Section 5.4 of the NEI guidance and confirmed that the remaining non-seismic hazards will not be considered during the categorization process. The NRC staff reviewed the licensee's response to RAI 05.a and finds the licensee appropriately identified the hazards that would be evaluated according to the flow chart from Figure 5-6 of Section 5.4 in NEI 00-04, Revision 0, because the non-seismic external hazards not included on the list have an insignificant impact on the categorization of SSCs.

RAI 05.a.i requested the licensee provide justification for the conclusion provided in Attachment 5 of the LAR that the external flood, high winds, and tornados hazard(s) cannot cause a core damage accident. The licensee's criterion is the same as Criterion A in SR EXT-C1 from ASME/ANS RA-Sa-2009 for screening non-seismic external hazards based on the current design-basis-hazard event not causing a core damage accident thereby providing an acceptable basis for bounding analysis or demonstrably conservative analysis.

The response to RAI 05.a.i provided by letter dated March 13, 2019, confirmed that a detailed re-evaluation of other external hazards was conducted and that the evaluation used the screening and evaluation criteria specified in Part 6 of ASME/ANS RA-Sa-2009.

The LAR as supplemented, confirmed that based on the re-evaluation, external flooding events would not cause flooding damage to safety-related SSCs and that external flooding and local intense precipitation hazard events could be screened out following ASME/ANS RA-Sa-2009.

The discussion provided for external flooding in Attachment 4 of the LAR discussed the re-evaluation of the external flood hazard performed in response to the post-Fukushima 10 CFR 50.54(f) request for information. The re-evaluated external flood hazard indicated that river flood was bounded by the current licensing basis and local intense precipitation did not challenge safety systems. The NRC staff reviewed the response to RAI 05.a.i with respect to external flooding and finds that the licensee's re-evaluation of the external flood hazard is applicable for the screening because the licensee's design basis can be considered to be a bounding or a demonstrably conservative analysis.

The NRC staff's previous assessment of NSPM's evaluation of the external flood hazard at NSPM dated April 12, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18081A948), refers to passive and active plant features that are credited to mitigate flood damage. The response to RAI 05.a.ii provided by letter dated March 13, 2019, stated in part,

> [e]xternal flood protection at MNGP for floods above site grade level is provided through construction of berms and a bin wall (levee), as well as sealing various openings upon prediction of flood levels approaching site grade level. The walls that are part of the plant structures generally prevent flooding ingress; however, not all external plant doors are watertight.

The licensee also confirmed that plant SSCs that are credited for external flooding protection will be evaluated in accordance with the guidance of Figure 5-6 of NEI 00-04 to ensure that no unscreened scenarios are created. The NRC staff finds the licensee's SSC categorization process will evaluate the safety significance of SSCs for the external flooding hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

The LAR confirmed that the licensee's re-evaluation of non-seismic external hazards also determined that NSPM has been designed for extreme winds and tornado loadings that meet or exceed the current regulatory guidance. In Attachment 4 of the LAR, the licensee concluded that wind damage is bounded by damage caused by tornados. The tornado wind speed corresponding to an exceedance frequency of 1×10^{-6} per year is less than the wind speed that plant structures were designed. Screening based on CDF less than 1×10^{-6} per year was met, and damage due to the forces associated with extreme wind or tornados could be screened. The criterion cited by the licensee is the same as screening Criterion C in SR EXT-C1 from ASME/ANS RA-Sa-2009, which states that the CDF, calculated using a bounding or demonstrably conservative analysis, has a mean frequency less than 10^{-6}/year.

The rationale for screening high winds and tornados did not take into consideration the possibility of tornado missiles. The NRC staff notes that tornados with frequencies higher than 1×10^{-6} per year (corresponding to lower wind speeds) can generate missiles which can potentially damage plant equipment that supports safe plant shutdown. The response to RAI 05.a.iii provided by letter dated March 13, 2019, stated, "[s]afety-related SSCs at MNGP are protected from high winds and tornados by reinforced concrete slabs and walls, and steel missile barriers." In addition, the response confirmed that penetrations that could not be screened were identified and an analysis was performed using the methodologies and data presented in NUREG/CR-4461, "Tornado Climatology of the Contiguous United States", and EPRI Report NP-2005, "Tornado Missile Simulation and Design Methodology, Volumes 1 and 2." The NRC staff finds that NSPM reviewed the design basis against Part 6 of ASME/ANS Ra-SA-2009 and performed additional analysis using the criteria in the ASME/ANS PRA Standard as well as an NRC-accepted methodology (Reference 38). The licensee confirmed that plant SSCs that are credited for high winds and tornado protection will be evaluated in accordance with the guidance of Figure 5-6 of NEI 00-04 to ensure that no unscreened scenarios are created. Therefore, the NRC staff finds the licensee's SSC categorization process will evaluate the safety significance of SSCs for the high winds and tornados hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

The response to RAIs 05.a, 05.b.i, and 05.b.ii provided by letter dated March 13, 2019, confirmed that the categorization process will not deviate from the guidance presented in NEI 00-04 for the evaluation of non-seismic external hazards and other hazards. The response further clarified that as part of the categorization assessment of non-seismic external hazards and other hazard risk an evaluation would be performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario and, consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, those components would be considered HSS. The NRC finds that the licensee's categorization process will evaluate the safety significance of SSCs for non-seismic external hazards and other hazards consistent with the guidance provided in Figure 5-6 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

3.5.3.3 Shutdown Risk

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

The use of NUMARC 96-01 described in the submittal is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. The approach uses an integrated and systematic process to identify HSS components, consistent with the shutdown evaluation process. Therefore, the NRC staff finds the licensee's use of NUMARC 96-01 is acceptable and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

3.5.4 Component Safety Significance Assessment for Passive Components

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

Section 3.1.2 of the LAR proposes using a categorization method for passive components not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1, for passive component categorization, but was approved by the NRC for ANO-2. The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for Class 2 and 3 pressure retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1" (Reference 27). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

Section 3.1.2 of the LAR states that "[t]he 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." The NRC staff finds, the licensee's proposed approach for passive categorization acceptable for use in the 10 CFR 50.69(b)(2)(iv) categorization process.

3.6 Maintain DID (NEI 00-04, Section 6)

Section 6 of NEI 00-04, Revision 0, provides guidance on assessment of DID. Figure 6-1 in NEI 00-04, Revision 0, provides guidance to assess design basis DID based on the likelihood of the design-basis internal initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. The likelihood of the initiating events are binned. The bins for the different likelihood consider whether HSS is assigned for SSCs that require fewer than the number of mitigating trains nominally available. Section 6 of NEI 00-04, Revision 0, also provides guidance to assess containment DID based on preserving containment isolation and long-term containment integrity and on preventing containment bypass and early hydrogen burns.

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

Section 3.1.1 of the LAR clarifies that an SSC will be categorized as HSS based on the DID assessment performed in accordance with NEI 00-04, Revision 0. The NRC staff finds that the licensee's process is consistent with the NRC-endorsed guidance in NEI 00-04 and, therefore, fulfills the 10 CFR 50.69 (c)(1)(iii) criteria that requires DID to be maintained.

3.7 Preliminary Engineering Categorization of Functions (NEI 00-04, Section 7)

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

Section 3.1.1 of the LAR states, in part, that "if any component is identified HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the DID assessment (Section 6), the associated system function(s) would be identified as HSS." The LAR goes on to state that, "[o]nce a system function is identified as HSS, then all the components that support that function are preliminary HSS." The NRC staff finds that the above description provided by the licensee for the preliminary categorization of functions is consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and, therefore, acceptable to meet the requirements set forth in 10 CFR 50.69(c)(1)(ii).

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

Regulation 10 CFR 50.69(c)(1)(iv) requires, in part, that any potential increases in CDF and LERF resulting from changes to treatment are small. The categorization process described in NEI 00-04, Revision 0, includes an overall risk sensitivity study to verify that any increase in CDF or LERF would remain small with regard to the cumulative impact postulated from simultaneous degradation in the reliability of all the LSS components.

Section 3.1.1 of the LAR states that an unreliability factor of 3 will be used for the sensitivity studies described in Section 8, "Risk Sensitivity Study," of NEI 00-04, Revision 0. Additionally, Section 3.2.7 of the LAR states that a cumulative sensitivity study will be performed where the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of 3. The NRC staff finds the application of a factor of 3 for the sensitivities is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

Section 3.1.1 of the LAR, for the "Overall Categorization Process," 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)." This sensitivity study together with the periodic review process discussed in Section 3.11 of this SE assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee will perform the risk sensitivity study consistent with the guidance in Section 8 of NEI 00-04, Revision 0, and will, therefore, assure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.9 IDP Review and Approval (NEI 00-04, Revision 0, Sections 9 and 10)

As required by 10 CFR 50.69(c)(2), the SSCs must be categorized by an IDP staffed with expert, plant knowledgeable, members whose expertise includes, at a minimum, PRA, safety analysis, plant operations, design engineering, and system engineering. Section 3.1.1 of the LAR states that the IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA. Therefore, the IDP will comprise the required expertise.

NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, provides guidance to ensure that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process as required by 10 CFR 50.69(c)(1)(ii). Section 3.1.1 of the LAR discusses that at least three members of the IDP will have a minimum of 5 years of experience at the plant, and there will be at least one member of the IDP who has a minimum of 3 years of experience in modeling and updating of the plant-specific PRA. The LAR further states that the IDP will be trained in the specific technical aspects and requirements related to the categorization process. This training will address, at a minimum, the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the DID philosophy and requirements to maintain this philosophy.

Attachment 1 of the LAR confirms as a perquisite for categorization that the licensee will establish procedure(s) prior to the use of the categorization process on a plant system. The LAR also confirms the procedure(s) will specifically include an element for the IDP member qualification requirements. The NRC staff finds that the licensee's IDP areas of expertise meet the requirements in 10 CFR 50.69(c)(2) and the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

Section 9.2.2, "Review of Safety Related Low Safety-Significant Functions/SSCs," of NEI 00-04, Revision 0, which is performed by the IDP, states, in part, that "in making their assessment, the IDP should consider the impact of loss of the function/SSC against the remaining capability to perform the basic safety functions . . ." This section also provides seven specific questions that should be considered by the IDP for making the final determination of the safety-significance for each function/SSC.

The IDP's authority to change component categorization from preliminary HSS to LSS is limited. Consistent with the guidance in NEI 00-04, Revision 0, components found to be HSS from the aspects of the process cannot be re-categorized by the IDP. Components not meeting the criteria for categorization, but identified as HSS through a seismic PRA, FPRA, or through the sensitivity studies outlined in Section 5 of NEI 00-04, may be presented to the IDP for categorization as LSS, if this determination is supported by the integrated assessment process and other elements of the categorization process.

RAI 04 requested clarification on how the IDP will collectively assess the seven specific questions to identify a function/SSC as LSS as opposed to HSS. The response to RAI 04.a provided by letter dated March 13, 2019, provided a revised Table 3-1 to replace the Table 3-1 from the LAR. The revised Table 3-1 includes a note to confirm that the assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. The response further confirmed, if the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The NRC staff reviewed the response provided in the supplemental letter dated March 13, 2019, and finds the proposed revision of Table 3-1 to be acceptable because it states that, "[i]f the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS."

The NRC staff finds this acceptable and consistent with the guidance in NEI 00-04, Revision 0, as endorsed in the NRC in RG 1.201, Revision 1.

The IDP may change the categorization of a component from LSS to HSS based on its assessment and decisionmaking. As outlined in Section 10.2, "Detailed SSC Categorization of NEI 00-04, Revision 0," the IDP may re-categorize components supporting an HSS function from HSS to LSS only if a credible failure of the component would not preclude the fulfillment of the HSS function and the component was not categorized as HSS based on the six criteria (i.e., IEPRA, integrated PRA component risk, shutdown, passive categorization, and DID). Components not meeting the criteria for categorization as described in the guidance in NEI 00-04, Revision 0, but identified as HSS through a seismic PRA, internal FPRA, or through the sensitivity studies in the guidance in Section 5 of NEI 00-04, Revision 0, may be presented to the IDP for categorization as LSS, if this determination is supported by the integrated assessment process and other elements of the categorization process.

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

3.10 <u>Programmatic Configuration Control NEI 00-04, Revision 0, Sections 11 and 12)</u>

Regulation 10 CFR 50.69(c)(ii) requires, in part, that all aspects of the integrated, systematic, process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices and applicable plant and industry operating experience. Sections 11 and 12 of NEI 00-04, Revision 0, includes discussion on periodic review; and program documentation and change control. Maintaining change control and periodic review will also maintain confidence, that all aspects of the 10 CFR 50.69 program and risk categorization for SSCs, continually reflect the NSPM as built, as-operated plant. A more detailed NRC staff review is provided as follows:

3.10.1 <u>Periodic Review (NEI 00-04, Revision 0, Section 12)</u>

Regulation 10 CFR 50.69(e), "Feedback and Process Adjustment," requires periodic updates to the licensee's PRA and SSC categorization. Changes over time to the PRA and to the SSC reliabilities are inevitable and such changes are recognized by the 10 CFR 50.69(e) requirement for periodic updates.

Section 11.2 of NEI 00-04, Revision 0, titled, "Following Initial Implementation," states, in part, "[t]he periodic update of the plant PRA may affect the results of the categorization process. If the results are affected, the licensee must make adjustments as necessary to either the categorization or treatment processes to maintain the validity of the processes." Section 3.2.6 of the LAR describes the process for maintaining and updating the NSPM PRA models used for the 10 CFR 50.69 categorization process. The described process provides administrative controls to ensure that the PRA models used to support the categorization reflect the as-built, as-operated plant over time. The process includes provisions for: monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience); assessing the risk impact of unincorporated changes; and controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization.

Routine PRA updates are performed every two refueling cycles at a minimum. The NRC staff finds the administrative controls described by the licensee in the LAR consistent with Section 12 of NEI 00-04, Revision 0, guidance as endorsed in the NRC in RG 1.201, Revision 1.

3.10.2 <u>Program Documentation and Change Control (NEI 00-04, Revision 0, Section 11)</u>

Regulation 10 CFR 50.69(f) requires, in part, program documentation, change control, and records. Section 3.2.6 of the LAR states that a process that addresses the requirements in Section 11 of NEI 00-04, Revision 0, pertaining to program documentation and change control records will be implemented. Section 3.1.1 of the LAR states that the RISC categorization process documentation will include the following 10 elements:

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

Attachment 1 of the LAR provides the following steps/elements that will be include in the 10 CFR 50.69 programmatic procedures prior to the use of the categorization process: (1) IDP member qualification requirements; (2) qualitative assessment of system functions;

(3) component safety significance assessment; (4) assessment of DID and safety margin; (5) review by the IDP; (6) overall risk sensitivity study; (7) periodic review; and (8) documentation requirements identified in Section 3.1.1 of the LAR.

The NRC staff recognizes for facilities licensed under 10 CFR Part 50, Appendix B, Criterion VI, for document control, procedures are considered formal plant documents requiring that, "[m]easures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality." The NRC staff finds that the elements provided in Section 3.1.1 of the LAR, in addition to the list of prerequisites provided in Attachment 1 of the LAR for the NSPM 10 CFR 50.69 categorization process that are to be documented in formal licensee procedures consistent with Section 11 of NEI 00-04, Revision 0, as endorsed in the NRC in RG 1.201, Revision 1, are sufficient for meeting 10 CFR 50.69(f) requirement for program documentation, change control and records.

3.11 Summary of 10 CFR 50.69 Categorization Process

The NRC staff finds the PRAs and the non-PRA methods described by the licensee in the submittal as supplemented in letters dated March 13, 2019, and May 15, 2019 (References 2 and 37), acceptable for use in the SSC categorization process. The NRC staff approves the use of the following PRA approaches and non-PRA methods in the licensee's 10 CFR 50.69 categorization process:

- IEPRA to assess internal events and internal flood risk, respectively
- FPRA to assess internal fire risk
- SMA performed for the NSPM IPEEE to assess seismic risk
- Screening analysis performed for the IPEEE to assess the contribution of risk from high winds, external floods, and external hazards
- Shutdown safety management plan consistent with NUMARC 91-06 to assess shutdown risk
- NRC-approved ANO-2 passive categorization method to assess passive components risk for Class 2 and 3 SSCs and their associated supports

RAI 09 requested assurance that the PRA changes identified to address the open F&Os across all PRA modeled hazards will be completed, and that the resolution of the F&Os are appropriately resolved prior to implementation of the 10 CFR 50.69 process. The response to RAI 09 provided by letter dated March 13, 2019, confirmed that the resolutions for the actions identified in response to RAIs 01.a through 01.h have been completed and there were no implementation items associated with RAI 09. In Section 3.5.1.4 of this SE, the NRC staff reviewed the licensee's responses and finds the licensee completed the actions identified for resolution of the F&Os.

The NRC staff reviewed all of the primary steps outlined in Section 3.2 of this SE used by the licensee in the 10 CFR 50.69 categorization process to assess the safety significance of active and passive components while ensuring the SSC's intended functions are required to remain intact. The NRC staff concludes that the licensee's categorization process adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with NEI 00-04, Revision 0, and RG 1.201, Revision 1, and, therefore, satisfies the requirements of 10 CFR 50.69(c). The NRC staff finds the licensee's proposed categorization process acceptable for categorizing the safety significance of SSCs. Specifically, the NRC staff concludes that the licensee's categorization process:

(1) Considers results and insights from plant-specific IEPRA and internal FPRA that have been subjected to a peer review process against RG 1.200, Revision 2, as reviewed in Section 3.5.1.2 through 3.5.1.6 of this SE is of sufficient quality and level of detail to support the categorization process and, therefore, meets the requirements in 10 CFR 50.69(c)(1)(i);

(2) Determines SSC functional importance using an integrated systematic process that reasonably reflects the current plant configuration, operating practices, and applicable plant and industry operational experience, as reviewed in Sections 3.3 through 3.8 and, therefore, meets the requirements in 10 CFR 50.69(c)(1)(ii);

(3) Maintains DID as reviewed in Section 3.6 of this SE and, therefore, meets the requirements in 10 CFR 50.69(c)(1)(iii);

(4) Includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small, as reviewed in Section 3.8 of this SE and, therefore, meets the requirements in 10 CFR 50.69(c)(1)(iv);

(5) Is performed for entire systems and structures, rather than for selected components within a system or structure, as reviewed in Section 3.3 of this SE and, therefore, the requirements in 10 CFR 50.69(c)(1)(v) will be met upon implementation; and

(6) Includes categorization by IDP, staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering and system engineering, as reviewed in Section 3.9 of this SE and, therefore, meets the requirements in 10 CFR 50.69(c)(2).

Attachment 1 of the LAR provides a list of prerequisite actions that will establish procedure(s) prior to the use of the categorization process on a plant system. The list of the eight prerequisites encompass in its entirety, the steps/elements described in the NEI 00-04, Revision 0. The NRC staff review is provided in Sections 3.3 through 3.10 of this SE. Therefore, the NRC staff concludes that upon the completion of these prerequisite actions, the NSPM SSC categorization will be updated and controlled consistent with 10 CFR 50.69(e) and (f) for ensuring that the categorization of SSCs continue to reflect the as-built-as-operated plant design.

4.0 CHANGES TO THE OPERATING LICENSE

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned upon the license condition provided below:

The LAR proposed to condition the RFOL for MNGP. The proposed condition was revised in response to RAI 07 which was provided in the letter dated May 15, 2019. The proposed MNGP license condition is:

> NSPM is approved to implement 10 CFR 50.69 using the approaches for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and

RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding and internal fire, with the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards (e.g., external flooding and high winds) updated using the external hazard screening significance criteria identified in ASME/ANS PRA Standard RA-Sa-2009, as endorsed in RG 1.200, Revision 2; as specified in MNGP License Amendment No. [XXX] dated [DATE].

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

The NRC staff finds that the proposed license condition and list of prerequisites are acceptable because they adequately implement 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment on July 25, 2019. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (83 FR 23735). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

8.0 REFERENCES

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2. Letter from Church, C. R., Xcel Energy, to U.S. Nuclear Regulatory Commission "Response to Request for Additional Information: Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" (EPID L-2018-LLA-0076)," dated March 13, 2019 (ADAMS Accession No. ML19072A298).

3. NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (ADAMS Accession No. ML052910035).

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

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9. Letter from Hill, W. J., Northern States Power Company, to U.S. Nuclear Regulatory Commission, "Submittal of Monticello Individual Plant Examination of External Events (IPEEE) Report, Revision 1; Seismic Analysis, Revision 0 and Internal Fires Analysis, Revision 1 (TAC M83644)," dated November 20, 1995 (ADAMS Accession No. ML080720565).

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11. Letter from Church, C. R., Xcel Energy, to U.S. Nuclear Regulatory Commission, Subject: "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).

12. TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control—RITSTF Initiative 5b," March 2009 (ADAMS Accession No. ML090850642).

13. RIS-07-06, "NRC Regulatory Issue Summary 2007-06 Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428).

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15. Anderson, V. K., Nuclear Energy Institute, letter to Stacey Rosenberg, U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, 'Close-Out of Facts and Observations,'" dated February 21, 2017 (ADAMS Package Accession No. ML17086A431).

16. Giitter, J., and Ross-Lee, M. J., U.S. Nuclear Regulatory Commission, letter to Mr. Greg Krueger, Nuclear Energy Institute, "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).

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21. U.S. Nuclear Regulatory Commission, "Clarifications on Treatment of Sensitive Electronics," FAQ 13-0004, Revision 1, June 26, 2013 (ADAMS Accession No. ML13182A708).

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