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March 13, 2019

L-MT-19-018
10 CFR 50.90
10 CFR 50.69

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

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

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)

References:

1) Letter (L-MT-18-010) from NSPM to the NRC, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors'", dated March 28, 2018 (ADAMS Accession No. ML18087A323)

2) Email from the NRC to NSPM, "Request for Additional Information RE: Monticello License Amendment Request to Adopt 10 CFR 50.69", dated January 31, 2019 (ADAMS Accession No. ML19031A913)

In Reference 1, Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), requested an amendment to adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components [SSCs] for Nuclear Power Reactors", for the Monticello Nuclear Generating Plant. The categorization process being implemented through this change is consistent with Nuclear Energy Institute (NEI) Report NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline", as endorsed by Regulatory Guide 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance". The NRC identified the need for additional information and provided the Request for Additional Information (RAI) in Reference 2. The enclosure to this letter provides NSPM's response to the NRC RAI.

The information provided in this letter does not alter the evaluations performed in accordance with 10 CFR 50.92 in Reference 1.

Please contact Mr. Richard Loeffler at (612) 342-8981 if additional information or clarification is required.

Summary of Commitments

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

I declare under penalty of perjury, that the foregoing is true and correct.
Executed on March 13, 2019.

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

Enclosure

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

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"

1.0 BACKGROUND

In Reference 1, Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), requested a license amendment request (LAR) to adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components [SSCs] for Nuclear Power Reactors", for the Monticello Nuclear Generating Plant (MNGP). The categorization process being implemented through this change is consistent with Nuclear Energy Institute (NEI) Report NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference 2), as endorsed by Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference 3). The NRC identified the need for additional information and provided the Request for Additional Information (RAI) in Reference 4. The NSPM responses to this RAI follow.

2.0 RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

<u>RAI 01 – Internal Fire PRA F&Os</u>

Section 50.69(c)(i) of 10 CFR requires that 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. Section 50.69(b)(2)(iii) of 10 CFR 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.

Attachment 3 of the LAR provides Facts and Observations (F&Os) that remain open following the Independent Assessment performed for the internal fire PRA (FPRA). The dispositions of several of these F&Os state that the open F&O has insignificant or no impact on the application, but do not provide sufficient justification. Also, several of dispositions state that, "[t]he Closure Review Team Recommendations will be addressed," and briefly state how the recommendation will be addressed, but do not propose a mechanism to ensure that the PRA update will be performed prior to implementation of the 10 CFR 50.69 program (for example, propose a licensee condition that includes all applicable implementation items and a statement that they will be completed prior to implementation of the 10 CFR 50.69 categorization program). Provide the following information:

a. F&O 2-5: Use of Transient Fire Influencing Factors

For F&O 2-5, the peer review team identified that the influencing factors assigned in the FPRA model were based on engineering judgement and a set of rules documented in Section 5.6.2 of the Ignition Frequency Notebook. The peer review team further stated that the influencing factors assigned resulted in comparatively low values (i.e., averaging much less than 3). In the NRC staff's parallel review of Monticello's proposed adoption of Technical Specifications Task Force (TSTF) Standard Technical Specification (STS) Change TSTF-425, Table 2-1 of the LAR (ADAMS Accession No. ML 17353A189) for the resolution of the F&O 2-5 the Independent Assessment team (i.e., Closure Review team) stated in part, "better justification of application of a 'very low' factor in two compartments [8 and 33] needs to be provided".

The update for treatment of influencing factors for the two fire compartment areas, 8 and 33, which were assigned very low influencing factors, could have an impact on this risk-informed application. Additionally, Frequently Asked Question (FAQ) 12-0064, "Close-Out of National Fire Protection Association 805 Frequently Asked Question 12-0064 on Hot Work/Transient Fire Frequency Influence Factors" (ADAMS Accession No.ML12346A488), provides related guidance for consideration in the use of influencing factors in an FPRA.

EITHER:

i. Provide discussion to support the justification for why the treatment (use of the influencing factors) used in the Monticello FPRA for fire compartments 8 and 33 is appropriate for this application (e.g., explain how the influencing factors used for fire compartments 8 and 33 are consistent with or bounds the guidance in FAQ 12-0064;

OR

ii. Provide the results of a sensitivity study performed to address the impact, and a description of how the conclusion of the sensitivity study considers changes to the PRA results (e.g., total CDF, total LERF, importance measures) used in the 10 CFR 50.69 categorization process).

OR

iii. Alternatively, propose a mechanism to ensure the activities and changes associated with F&O 2-5 will be completed, appropriately reviewed, and any issues resolved prior to implementation of 10 CFR 50.69 categorization process. Additionally, this mechanism should specify how the F&O 2-5 will be resolved in the PRA at Capability Category (CC) II for the applicable Supporting Requirements (SRs) and include any additional finding-level F&O(s) identified as a result of performing a potential peer review (i.e., resolution of the F&O that may involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

b. F&O 3-6: Fire-Induced Failures

The disposition to F&O 3-6 states that the F&O Independent Assessment team found about ten components that should be treated as failed in a fire that were not treated as such in the FPRA model. The disposition to F&O 3-6 also states, "[t]he Closure Review Team Recommendation will be addressed by including the specified basic events in the fire failed events flag file."

The disposition does not explain how Monticello will ensure 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. The licensee further states in the LAR, a sensitivity study was performed that demonstrates exclusion of the cited fire-induced failures has only a small effect on core damage frequency (CDF) and large early release frequency (LERF). It is not clear to the NRC staff how the sensitivity study performed concluded that the excluded fire-induced failures would have an insignificant impact on the categorization of SSCs associated with specific systems. Considering these observations:

EITHER

i. Provide discussion to support the justification that the exclusion of all applicable basic events from the fire-failed events flag file has no impact on the PRA results used to support risk-informed categorization.

OR

ii. Provide the results of a sensitivity study performed to address the impact, and a description of how the conclusion of the sensitivity study considers changes to the PRA results (e.g., total CDF, total LERF, importance measures) used in the 10 CFR 50.69 categorization process.

OR

iii. Alternatively, propose a mechanism that ensures F&O 3-6 will be resolved prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of the changes that will be made to the PRA model or documentation to resolve this issue and include any additional finding-level F&O(s) identified as a result of performing a potential peer review that may be determined necessary for resolution of the F&O (i.e., involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

c. F&O 4-11: 20 degrees Celsius (C) Ambient Air Assumption

For F&O 4-11 the peer review team identified that using an initial ambient air temperature of 20 degrees C 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."

The disposition does not explain how Monticello will 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. Considering these observations:

EITHER:

i. Provide discussion to support the justification that the initial ambient air temperatures for fire modeling has no adverse impact (does not mask/skew the importance measures of other SSCs) or no impact on the PRA results used to support risk-informed categorization.

OR

ii. Alternatively, propose a mechanism that ensures F&O 4-11 will be resolved at CC II for the applicable SR(s) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model or documentation to resolve this issue and include any additional finding-level F&O(s) identified as a result of performing a potential peer review that may be determined necessary for resolution of the F&O (i.e., involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

d. F&O 4-20: Treatment of Sensitive Electronics

For the resolution of F&O 4-20, the Independent Assessment team states in part, 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 licensee's disposition to the F&O states that addressing 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. Verification is not a documentation issue when configurations are potentially identified that result in modelling changes to the PRA that could impact the application. Considering these observations:

EITHER:

i. Confirm that the guidance in FAQ 13-0004 (ADAMS Accession No. ML13182A708) has been fully implemented for all fire zones addressed in the FPRA model, including for the main control room, (i.e., complete the verification);

OR

ii. If the guidance in FAQ 13-0004 was not fully implemented, provide justification that addresses why this incomplete treatment (deviation) does not impact the 10 CFR 50.69 application. Include in the justification a description of the proposed alternate treatment, applicable fire zones, and the associated impact to the 10 CFR 50.69 categorization process.

OR

iii. Provide the results of a sensitivity study performed to address the impact, and a description of how the conclusion of the sensitivity study considers changes to the PRA results (e.g., total CDF, total LERF, importance measures) used in the 10 CFR 50.69 categorization process).

OR

iv. Alternatively, propose a mechanism that ensures F&O 4-20 will be resolved at CC II for the applicable SRs prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model and/or documentation to resolve this issue and include any additional finding-level F&O(s) identified as a result of performing a potential peer review that may be determined necessary for resolution of the F&O (i.e., involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

e. F&O 4-33: Wall and Corner Effects Using FLASH-CAT

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. It is not clear to the NRC staff how the fire scenarios that need detailed fire modelling in the FPRA model are determined (i.e., considered) and the overall impact on the PRA results used to support risk-informed categorization. Considering these observations:

i. Provide discussion to support justification that the current fire modeling practices that do not consider detailed fire modelling is bounding to the as-built, as-operated plant and has no adverse impact (does not mask/skew the importance measures of other SSCs) or no impact on the PRA results used to support the 10 CFR 50.69 risk-informed categorization.

OR

ii. Provide the results of a sensitivity study performed to address the impact, and a description of how the conclusion of the sensitivity study considers changes to the PRA results (e.g., total CDF, total LERF, importance measures) used in the 10 CFR 50.69 categorization process).

OR

iii. Alternatively, propose a mechanism that ensures F&O 4-33 will be resolved at CC II for the associated SR(s) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model and/or documentation to resolve this issue and include any additional finding-level F&O(s) identified as a result of performing a potential peer review that may be determined necessary for resolution of the F&O (i.e., involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

f. Resolutions of Identified F&Os

A number of recommended actions specified by the Independent Assessment team for F&O closure were identified and provided in Attachment 3 of the LAR that have not been corrected in the PRA model and/or associated documentation proposed to be used in the 10 CFR 50.69 categorization process. The resolutions are associated with F&Os 6-3, 6-9, 6-11, and 7-3. For the disposition of all these F&Os, the licensee 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 on the PRA results to support risk-informed categorization.

Considering these observations, propose a mechanism to ensure that all the corrections related to F&Os 6-3, 6-9, 6-11, and 7-3, will be resolved at CC II for the applicable SR(s) and incorporated into the FPRA model and/or documentation prior to implementation of the 10 CFR 50.69 program. This mechanism should also provide an explicit description of the changes that will be made to the PRA model(s) and/or documentation to resolve this issue and include any additional finding-level F&O(s) identified as a result of performing a potential peer review that may be determined necessary for resolution of the F&O (i.e., involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

g. F&O 7-4: Logic Associated With Fire-induced Openings of Safety Relief Valves (SRVs)

In Attachment 3 of the LAR, the F&O resolution states in part, the F&O finding closure review team identified additional locations in the model where the revised logic model still needs to be added to fully account for fire-induced SRV opening scenarios. The disposition to F&O 7-4 states, "[t]he Closure Review Team recommendation will be addressed by performing thermal hydraulic MAAP analysis to determine the success criteria for the opening of two or more SRVs. The fault tree model will be revised to reflect the determined success criteria." Considering these observations:

EITHER:

i. Provide discussion to support the justification that the success criteria given two or more open SRVs has no adverse impact (does not mask/skew the importance measures of other SSCs) and/or no impact on the PRA results used to support the 10 CFR 50.69 categorization process.

OR

ii. Provide the results of a sensitivity study performed to address the impact, and a description of how the conclusion of the sensitivity study considers changes to the PRA results (e.g., total CDF, total LERF, importance measures) used in the 10 CFR 50.69 categorization process).

OR

iii. Propose a mechanism 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. This mechanism should also provide an explicit description of the changes that will be made to the PRA model(s) and/or documentation to resolve this issue and include any additional finding-level F&O(s) identified as a result of performing a potential peer review that may be determined necessary for resolution of the F&O (i.e., involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

h. F&O FO-1– Exclusion of Credit Associated with Ventilation-Limited Burning

For F&O FO-1 the resolution states in part, the F&O finding closure review team identified issues with the sensitivity study case and its applicability in certain situations. Additional justification concerning the treatment of the ventilation-limited modeling for those areas needs to be developed. The disposition states [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.

It is not clear to the NRC staff 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). Considering these observations:

EITHER

i. Provide justification to support that removal of the credit for ventilation-limited burning in the cable heat soak models has no adverse impact (does not mask/skew the importance measures of other SSCs) and/or no impact on the PRA results used to support risk-informed categorization.

OR

ii. Provide the results of a sensitivity study performed to address the impact, and a description of how the conclusion of the sensitivity study considers changes to the PRA results (e.g., total CDF, total LERF, importance measures) used in the 10 CFR 50.69 categorization process).

OR

iii. Alternatively, propose a mechanism that ensures F&O FO-1 will be resolved prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model or documentation to resolve this issue and include any additional finding-level F&O(s) identified as a result of performing a potential peer review that may be determined necessary for resolution of the F&O (i.e., involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

NSPM Response

Response to RAI 01.a, “F&O 2-5: Use of Transient Fire Influencing Factors”:

The MNGP internal fire PRA method for applying transient fire influencing factors follows PRA FAQ 12-0064, “Hot/Work/Transient Fire Frequency Influence Factors” (Reference 5). However, as was noted in the F&O Closure team’s comments, better justification was required for the application of the “very low” transient storage factor of 0.3 to fire compartments 8 and 33. It 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. This is consistent with an RAI response (Reference 6) for the MNGP application to adopt TSTF-425, “Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b”.

Response to RAI 01.b, “F&O 3-6: Fire-Induced Failures”:

The MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update. Basic 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.

Response to RAI 01.c, “F&O 4-11: 20 degrees Celsius (C) Ambient Air Assumption”:

The MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update. Fire modeling calculations have been updated to use a bounding ambient temperature.

Response to RAI 01.d, "F&O 4-20: Treatment of Sensitive Electronics":

The MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update. 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. It should be noted that this is a change from the expected resolution that was provided in the LAR.

Response to RAI 01.e, "F&O 4-33: Wall and Corner Effects Using FLASH-CAT":

The MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update. Wall and corner effects have been applied to applicable Consolidated Fire and Smoke Transport (CFAST) heat release rates.

Response to RAI 01.f, "Resolutions of Identified F&Os":

The MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update. The following items have been performed for the corresponding findings:

(1) F&O 6-3: Battery Chargers D70, D80, and D90 have been counted in Bin 10. Of note, the LAR incorrectly specified that these battery chargers would be changed to Bin 15. The chargers have been changed from Bin 15 to Bin 10.

(2) F&O 6-9: RHR Pump B maintenance unavailability and common cause failure events have been included in alternate shutdown scenarios.

(3) F&O 6-11: Failure probabilities have been assigned to non-flag basic events.

(4) F&O 7-3: Modeling of reactor level instrumentation has been updated to reflect redundancy and procedures for loss of level instrumentation. Of note, this is a change from the expected resolution provided in the LAR.

Response to RAI 01.g, "F&O 7-4: Logic Associated With Fire-induced Openings of Safety Relief Valves (SRVs)":

The MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update. Spurious opening or failure to reclose of two or more SRVs has been updated to more accurately reflect the expected plant response.

Response to RAI 01.h, "F&O FO-1– Exclusion of Credit Associated with Ventilation-Limited Burning":

The MNGP internal fire PRA has been updated following the NSPM PRA procedure for model maintenance and update. The 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.

<u>RAI 02 – Identified Key Assumptions and Sources of Uncertainties</u>

Paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR require that a licensee's PRA be of sufficient quality and level of detail to support the SSC categorization process, and 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 operational experience. The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model to address uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask the SSC(s) importance. Regulatory Guide 1.174, Revision 3, cites NUREG-1855, Revision 1, as related guidance. In Section B of RG 1.174, Revision 3, the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties.

In Section 4.1 of the LAR, Monticello identifies RG 1.174, Revision 3, as an applicable regulatory requirement/criteria. Contrary to Section 4.1 of the LAR, 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-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," was used to identify, characterize, and screen model uncertainties. Attachment 6 of the LAR identifies five assumptions and sources of uncertainty applicable to either the IEPRA (includes internal flood) or FPRA models.

NUREG-1855 has been updated to Revision 1 as of March 2017 (ADAMS Accession No. ML17062A466). The NRC staff notes that NUREG-1855, Revision 1, provides guidance in stages A through E for how to treat uncertainties associated with PRA models in risk-informed decisionmaking. Revision 1 of NUREG-1855 cites EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty." Considering these observations provide the following:

a. A detailed summary of the process used to identify the key assumptions and sources of uncertainty presented in Attachment 6 of the LAR. The discussion should include:

    i. How the process is consistent with NUREG-1855, Revision 1, or other NRC-accepted methods (e.g., NUREG-1855, Revision 0). If deviating from the current guidance provided in NUREG-1855, Revision 1, 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).

    ii. A brief description 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) uncertainties and assumptions, including those associated with plant-specific features, modeling choices, and generic industry concerns. This can include an identification of the sources of plant-specific and applicable generic modeling uncertainties identified in the uncertainty analyses for the base IEPRA (includes internal flood) and the base FPRA and include a disposition for each of the assumptions and/or uncertainties addressing their impact for the 10 CR 50.69 risk application. For any source of uncertainty or assumption judged not to be key to the application, provide discussion for why it is not pertinent to the application and therefore does not need to be addressed (i.e., sensitivity studies performed).

b. If the process used to identify, characterize, and assess the key assumption(s) and the treatment for the sources of uncertainty provided in Attachment 6 of the LAR cannot be justified for use in the 50.69 categorization process, provide the results of an updated assessment of the key assumptions, sources of uncertainty, and treatment of the sources of uncertainty performed in accordance with NUREG-1855, Revision 1, and NEI 00-04, Revision 0. For the treatment of the sources of uncertainty (e.g., sensitivity studies to be performed) include a detailed description of the sensitivity study and how the sensitivity study is bounding to address the specific key assumption and/or source of uncertainty.

NSPM Response

At the time of the submittal of the LAR, the sources of uncertainty evaluation for the internal events PRA had considered both plant-specific sources of uncertainty and the generic uncertainties identified in EPRI TR-1016737 (Reference 7). The fire PRA considered the plant-specific uncertainty sources, but did not specifically address the EPRI generic sources as provided in EPRI TR-1026511 (Reference 8).

Since 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 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. This evaluation meets the intent of steps C-1 and E-1 of NUREG-1855, Revision 1 (Reference 9).

At 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 approach used for this review is similar to that used at the Prairie Island Nuclear Generating Plant for its 10 CFR 50.69 LAR (Reference 10). 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. This evaluation meets the intent of the screening portion of steps C-2 and E-2 of NUREG-1855, Revision 1.

The ultimate goal in assessing model uncertainty is to determine whether (and the degree to which) the risk metric results challenge or exceed the quantitative acceptance guidelines for the application, due to sources of model uncertainty and related assumptions. For 10 CFR 50.69 categorization, the acceptance guidelines are actually threshold values for Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) for each SSC being categorized, above which the SSC is categorized as high safety significant (HSS), and below which the SSC is categorized as low safety significant (LSS). As described in Step E-2 of the NUREG, each relevant uncertainty/assumption requires some sort of sensitivity analysis, and each sensitivity performed to evaluate an uncertainty/assumption involves some change to the PRA results. Since any change to the PRA results has the potential to change the F-V and RAW importance measures for all components, every relevant uncertainty/assumption has the potential to challenge the acceptance guidelines. That is, since RAW and F-V are relative importance measures, any change to any part of the model will generate a new set of cutsets and potentially impact the RAW and F-V for every SSC. Thus, the only way to evaluate the impact of a sensitivity is to quantify the sensitivity case and compare the F-V and RAW values for all SSCs against the base case F-V and RAW values to determine if any exceed the HSS threshold in the sensitivity case that did not previously do so.

As stated in Stage F, Section 8.1, of NUREG-1855, Revision 1, an appropriate method for dealing with uncertainties and related assumptions that challenge or exceed the acceptance guidelines is to use compensatory measures or performance monitoring requirements. Section 8.5 of the NUREG states that performance monitoring can be used to demonstrate that, “following a change to the design of the plant or operational practices, there has been no degradation in specified aspects of plant performance that are expected to be affected by the change. This monitoring is an effective strategy when no predictive model has been developed for plant performance in response to a change”. Since no predictive model of the increase in unreliability following alternative treatment of LSS SSCs exists, this option is appropriate for 10 CFR 50.69. In fact, the example of a performance monitoring approach to address key uncertainties/assumptions given in Section 8.5 is the factor of increase sensitivity combined with the performance monitoring process described for 10 CFR 50.69 in NEI 00-04. The NUREG, Section 8.5, states:

> One example of such an instance is the impact of the relaxation of special treatment requirements (in accordance with 10 CFR 50.69) on equipment unreliability. No consensus approach to model this cause-effect relationship has been developed. Therefore, the approach adopted in NEI 00-04 as endorsed in Regulatory Guide 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” [NRC, 2006a] is to:
>
> - Assume a multiplicative factor on the SSC unreliability that represents the effect of the relaxation of special treatment requirements.
>
> - Demonstrate that this degradation in unreliability would have a small impact on risk.

> Following acceptance of an application which calls for implementation of a performance monitoring program, such a program would have to be established to demonstrate that the assumed factor of degradation is not exceeded.

The use of the sensitivity study required by Section 8.1 of NEI 00-04 and performance monitoring of LSS SSCs as required by 10 CFR 50.69(e)(3) is appropriate to address key uncertainties and assumptions. The impact of any key uncertainty or assumption sensitivity would be to potentially cause an SSC to be categorized as HSS when the base PRA analysis showed it to be LSS. The potential impact of categorizing an SSC as LSS rather than HSS is that the SSC could have alternative treatments applied to it and as such, the possibility exists that the reliability of SSC could be reduced (i.e., the specified aspect of plant performance that is expected to be affected by the change is the reliability of the SSC). Per Section 8.1 of NEI 00-04, a sensitivity is performed which assumes the unreliability of all LSS components is increased by a factor of 3 to 5. Since, as discussed in NEI 00-04, no significant decrease in reliability is expected, this is very conservative. Additionally, since the failure probability of all LSS SSCs are increased at the same time in the sensitivity, this approach addresses all uncertainties/assumptions which could potentially impact the LSS/HSS categorization. The LSS sensitivity then must be shown to demonstrate that even assuming this factor increase, the quantitative guidelines of RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (Reference 11), are not exceeded. Thus, the LSS sensitivity demonstrates that the potential impact of all uncertainties/assumptions is acceptable.

Additionally, a performance monitoring program must be established as part of the 10 CFR 50.69 process (per NEI 00-04 Section 12) which will monitor the reliability of all LSS SSCs to ensure that the factor of increase assumed in the sensitivity is not exceeded. This ensures the validity of the sensitivity study following implementation.

It is noted that uncertainties/assumptions which are related to SSCs being excluded from the PRA model, may not be adequately addressed by the above sensitivity and performance monitoring program. These SSCs may have been excluded because they are not believed to be required for accident mitigation, because they perform a backup function to other equipment but were conservatively not credited in the model, because their failure probability is negligible, etc. As such non-modeled functions were excluded on the basis of low importance to the PRA results initially, it is unlikely that explicitly modeling these functions would have a significant impact on the risk ranking results.

As a result of the updated evaluation of the uncertainties, Attachment 6 has been revised, replacing in total what was provided in the LAR.

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

Assumption/Uncertainty	Discussion	Disposition
Very small loss of coolant accidents (LOCAs) are defined as those for which flow rates are less than can be made up by normal makeup systems such as Control Rod Drive Hydraulic System (~100 gpm). The mitigating success criteria for this break size is identical to that for a transient initiator in which decay heat makeup rates are required, the only difference in plant response being that a high drywell pressure may occur. This would result in trip of drywell coolers and the Residual Heat Removal Service Water System due to load shed, but would affect no other mitigating systems. Therefore, this event is considered to be encompassed by the Reactor Trip or Turbine Trip initiating event and no new initiating event is created for the purpose of evaluating very small LOCAs.	The impact of this assumption will need to be assessed for specific risk applications, including 10 CFR 50.69. Particularly, any applications where small LOCA initiators could be significant contributors may be affected by this assumption. The drywell coolers are not credited in the PRA model; therefore, there is no risk impact.	A sensitivity study will be performed 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.
A minimum value for a single pre or post initiator Human Error Probability (HEP) was assumed to be 1.0E-5. This value is reserved for operator actions which only take a few minutes but have over ten hours to perform. An independent or dependent HEP combination minimum value was assumed to be 1E-6.	HEP values and their dependence have the ability to significantly impact model results; therefore, this is considered an uncertainty.	Sensitivity studies in accordance with NEI 00-04, Table 5-2, will be performed to evaluate the potential impact of variations in HEP values.

Assumption/Uncertainty	Discussion	Disposition
While the walkdown sheets were used whenever possible to obtain source systems, pipe sizes, and pipe lengths for the various walkdown zones, there are zones in the plant for which no such data exists. Such cases required the estimation of the necessary equipment based on P&ID information and the analysts' experience at other similar plants. The walkdown notebook documents the data that was estimated for this analysis.	Per EPRI report (Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments, Revision 3, EPRI-3002000079 (Reference 12), April 2013) internal event flood frequencies are directly proportional to the pipe lengths. The internal flood events where pipe lengths could not be validated with a walkdown were reviewed and found to have reasonable pipe length estimates based on room size. Furthermore, their risk contribution was insignificant. Isometric drawings were used to estimate pipe lengths for high-risk floods.	This item does not represent a key source of uncertainty for 50.69 calculations.

<u>RAI 03 – Disposition of Key Assumptions and Sources of Uncertainties</u>

Paragraph 50.69(c)(1)(i) of 10 CFR requires the licensee to consider the results and insights from the PRA during categorization. The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model to address uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask importance of components.
NEI 00-04 guidance states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model to address sources of uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask importance of components. NEI 00-04 guidance states that applicable sensitivity studies from characterization of PRA adequacy should be considered. For the sources of uncertainty provided in Attachment 6 of the LAR the dispositions do not discuss the specific treatment (e.g., sensitivity study) that will be performed to address the source of uncertainty and/or provide in the disposition a conclusion for why the impact of the source of uncertainty is not adverse and/or insignificant to the risk application.

- Ignition counting in the FPRA model

- Fire cable selection for the FPRA model

- Heat release rates specified in NUREG/CR-6850 for the FPRA model

Considering the NRC staff observations, for each of the above sources of uncertainty identified, provide the following:

EITHER:

a. Provide discussion to justify why the source of uncertainty is not adverse and or insignificant to the risk application, and therefore does not need to be addressed (i.e., sensitivity study performed) for the application.

OR

b. Provide the quantitative results of a sensitivity study and/or justification that supports the conclusion that the source of uncertainty has no adverse impact (i.e., mask/skew the importance measures for other SSCs) and/or insignificant impact on the 10 CFR 50.69 categorization process. Include in the justification (1) a description of the sensitivity study that was performed for the FPRA, (2) how it considered the potential to mask/ skew the importance of certain SSCs, and (3) how the sensitivity study performed bounds the source of uncertainty being addressed.

c. Describe which of the sensitivity studies outlined in Section 5 of NEI 00-04 is directly applicable for this key assumption. Describe how the sensitivity study will be performed and include justification that addresses (1) why the sensitivity study bounds the source of uncertainty being addressed and (2) how the potential to mask/skew the importance measures of other SSCs is considered.

OR

d. If justification and/or a sensitivity study cannot be provided in parts (a), (b), or (c) to confirm that the source of uncertainty is not adverse and/or insignificant to the 10 CFR 50.69 risk application, then propose a mechanism to address (e.g., eliminate) the source of modelling uncertainty in the FPRA model prior to implementation of the 10 CFR 50.69 risk application. This mechanism should also provide an explicit description of changes that will be made to the PRA model(s) and/or documentation to resolve this issue and include any additional finding-level F&O(s) identified as a result of performing a potential peer review that may be determined to be necessary for resolution of the F&O (i.e., involve an upgrade). An example mechanism would be a table of listed implementation items referenced in a license condition.

<u>NSPM Response</u>

The 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 (Reference 13) at Capability Category II. In particular, an additional review of the previously identified sources of uncertainty determined that each of the previously identified sources are not actually significant uncertainty sources since they follow current NRC fire PRA guidance. These sources of uncertainty are:

- Ignition sources were counted using guidance from NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology” (Reference 14), and NUREG/CR-6850 Supplement 1, “Fire Probabilistic Risk Assessment Methods Enhancements” (Reference 15).

- Cable selection was performed using guidance from NUREG/CR-6850, Volume 2, and NUREG/CR-7150, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)”, Volume 1 (Reference 16).

- Heat release rates of ignition sources were taken from NUREG/CR-6850, Volume 2, NUREG-7010, Volume 1, “Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE)” (Reference 17), and NUREG-2178, Volume 1, “Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)”, (Reference 18).

RAI 04 – Qualitative Function Categorization

NEI 00-04, Section 9.2.2, "Review of Safety Related Low Safety-Significant Functions/SSCs, “states in part, 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 questions that should be considered for making the final determination of the safety-significance for each system function/SSC. In Table 3-1 of the LAR, the intersection of the column labeled “IDP Changes from Preliminary HSS to LSS” and the row labeled “Qualitative Criteria” states that the IDP can change HSS to LSS. It is unclear from the LAR how the IDP will collectively assess these seven specific questions.

a. Clarify the IDP will collectively assess the seven specific questions to identify a function/SSC as LSS as opposed to HSS. For example, a function/SSC is considered HSS when the answer to any one question is false.

b. If the criteria provided in part (a) considers more than one question is false for the IDP to assign a category of HSS to an SSC, provide justification to support rationale for why this is appropriate to use in the 10 CFR 50.69 risk-informed application.

NSPM Response

Table 3-1 has been modified to further clarify how the guidance in NEI 00-04, Section 9.2, “IDP Process”, is applied to the NSPM categorization process. The following revision of Table 3-1 replaces what was provided in the LAR:

Table 3-1 – Categorization Evaluation Summary

Element	Categorization Step (NEI 00-04 Section)	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case (Section 5.1)	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case (Sections 5.2, 5.3 and 5.4)		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment (Section 5.6)		Not Allowed	Yes
Risk (Non-modeled)	Fire, Seismic and Other External Hazards (Sections 5.2, 5.3 and 5.4)	Component	Not Allowed	No
	Shutdown (Section 5.5)	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage (Section 6.1)	Function/Component	Not Allowed	Yes
	Containment (Section 6.2)	Component	Not Allowed	Yes
Qualitative Criteria	Considerations (Section 9.2)	Function	Allowable[1]	N/A
Passive	Passive (Section 4)	Segment/Component	Not Allowed	No

Notes:

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

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

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 50.69 team (i.e. all considerations for all functions are reviewed). The IDP may confirm the preliminary

Element	Categorization Step (NEI 00-04 Section)	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.				

<u>RAI 05 – SSCs Categorization Based on Other External Hazards</u>

Sections 50.69(c)(1)(ii) of 10 CFR require that the licensee 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.

LAR Section 3.2.4 states in part, "[a]ll other hazards (i.e., not seismic or fire hazards) were screened from applicability to Monticello per a plant-specific evaluation in accordance with GL-88-20, supplement 4, and updated to use the criteria in ASME PRA Standard RA-Sa-2009." This statement appears to indicate that Monticello proposes to treat all SSCs as LSS with respect to other external events. However, the LAR also states that "[a]s part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario." The two cited statements from the LAR seem to be in conflict. Attachments 4 and 5 of the LAR provide a summary of the other external hazards screening results, but does not appear to address any considerations related to applying Figure 5-6 of NEI 00-04 guidance to those hazards. Considering these observations:

a. Identify the external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6. Provide detailed justification for screening external hazards (i.e., external flood, high winds, and tornados) using the criteria in Part 6 of ASME/ANS RA-Sa-2009. As applicable, the justification should include consideration of uncertainties in the determination of demonstrably conservative mean values as discussed in Section 6.2-3 of ASME/ANS RA-Sa-2009.

    i. Provide justification for the conclusion provided in Attachment 5 of the LAR for criterion PS1, that the external flood, high winds, and tornados hazard(s) cannot cause a core damage accident.

    ii. Attachment 4, External Hazards Screening, of the LAR states that recent evaluation of the external flood hazard performed in response to the post-Fukushima 50.54(f), request for information indicated that risk from river flood is bounded by the current licensing basis and local intense precipitation does not challenge safety systems. Section 3.1.1 of the LAR also states that for these reasons the external flood hazard was screened out. An NRC staff assessment of Monticello's evaluation of the external flood hazard at Monticello dated April 12, 2016 (ADAMS Accession No. ML18081A948) refers to passive and active plant features that are credited to mitigate flood damage. For the external flood hazard, provide detailed justification for concluding that the current licensing basis and local intense precipitation is bounding (i.e., external flood hazard CDF is less than 1×10-6 per reactor-year).

iii. Attachment 4 of the LAR states that wind damage is bounded by damage caused by tornadoes. Attachment 4 of the LAR also states that 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 to and therefore screening category PS4 (CDF less than 1×10-6 per year) is met, and damage due to the forces associated with extreme wind or tornadoes can be screened. However, this rational for screening tornadoes does not take into consideration the possibility of tornado missiles. The NRC staff notes that tornadoes with higher exceedance frequencies 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. Also, the LAR does not provide a basis or justification for the CDF associated with tornadoes missiles is 1.1×10-7 per year. Provide detailed justification for concluding that for the high winds and tornados hazard, the mean frequency is less than 1×10-5 per reactor-year and the mean conditional core damage probability is less than 0.1.

b. Figure 5-6 of NEI 00-04 shows that if an SSC is included in a screened scenario(s), then for that SSC to be considered a candidate LSS, the licensee has to show that if the component was removed, the screened scenario(s) would not become unscreened.

i. Identify and justify what type of SSCs, if any, are credited in the screening of the external hazard(s), including both passive, active, and temporary features.

ii. If there are any SSCs credited for screening of the external hazard(s), then explain and justify how the guidance in Figure 5-6 of NEI 00-04 will be applied for each of the external hazard(s).

c. If the external hazards (i.e., external flood, high winds and tornados) cannot be screened out in item (a), discuss, using quantitative or qualitative assessments, how the risk from those hazards will be considered in the categorization program. The discussion should include consideration of and, as applicable, the basis for the following factors:

- The frequency of the external hazard(s),
- The impact of the external hazard(s) on plant SSCs and plant's operation including the ability to respond to the external hazard initiating event,

- The operating experience associated with reliability of the external hazard(s) protection measures (e.g., flood seals), and

- The reliability of operator actions.

NSPM Response

Response to RAI 05.a:

Section 3.2.4 of the LAR has been modified, as indicated below, to better clarify how the guidance in NEI 00-04, Section 5.4, is applied to the NSPM categorization process. Changes are represented in italics:

3.2.4 Other External Hazards

All other external hazards (i.e., not seismic or fire hazards) were screened from applicability to MNGP per a plant-specific evaluation in accordance with GL 88-20, Supplement 4, and updated to use the criteria in the ASME PRA Standard RA-Sa-2009 (Reference 11). Attachment 4 to this enclosure provides a summary of the other external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

As part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS.

The following hazards will be evaluated for the potential for unscreened scenarios as a result of the SSC categorization process:

- *External Flooding*
- *Extreme Wind or Tornado*
- *Lightning*
- *Low River and Drought*
- *Low Winter Temperature, Snow, and Ice Cover*
- *Transportation and Pipeline Accidents*
- *Toxic Gases*
- *Forest or Range Fire*
- *Sand or Dust Storm*

All remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

Response to RAI 05.a.i:

A detailed re-evaluation of other external hazards was conducted. This evaluation utilized the screening and evaluation criteria specified in Part 6 of ASME/ANS RA-Sa-2009.

In that evaluation, the updated examination of external flood risk, including the updated plant data, flood history and new measures for risk management validate the current flood mitigation strategy of the current design basis. External flooding events will cause no flooding damage to MNGP safety-related SSCs. External flooding and intense precipitation hazard events can be screened out from the MNGP PRA following the ASME/ANS RA-Sa-2009 supporting requirements EXT-B1 (Criterion 3) and EXT-C1 (Criterion A).

The evaluation also determined that MNGP has been designed for extreme winds and tornado loadings that meet or exceed of the current regulatory guidance. The safety related SSCs are protected from tornado missiles using barriers with thicknesses exceeding the current requirements based on recent tornado hazard analysis. Therefore, the MNGP design meets the supporting requirement EXT-C1 Criterion C. It is concluded that the hazard events of extreme winds and tornadoes can be screened out for MNGP.

Response to RAI 05.a.ii:

The probable maximum flood (PMF) at MNGP was determined to be 939.2 feet mean sea level (MSL). The site grade level is 930 feet MSL. External 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.

As noted in the response to RAI 05.a, 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.

Response to RAI 05.a.iii:

Safety-related SSCs at MNGP are protected from high winds and tornados by reinforced concrete slabs and walls, and steel missile barriers. As described in the MNGP Individual Plant Examination of External Events (IPEEE) (References 19 and 20), MNGP structures that include the design basis tornado missiles in their design are considered to provide adequate protection from these missiles since the building wall thicknesses exceed the RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants", Revision 1 (Reference 21), required thicknesses for a Region I tornado. Therefore, these walls can be screened from further consideration. Further, 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" (Reference 22), and EPRI Report NP-2005, "Tornado Missile Simulation and Design Methodology, Volumes 1 and 2" (References 23 and 24), to determine, for each of the identified openings, the corresponding missile strike probability. A conditional core damage frequency (CCDF) was developed assuming damage to equipment in any areas that a missile may penetrate. The product of the missile strike probability and the CCDF gives the CDF associated with a particular missile penetration. The sum of the individual missile-induced core damage frequencies yielded the core damage frequency for tornado missiles. The overall result for the collective set of credible missile strikes produced a CDF of 1.1x10-7/yr, which provides justification for screening tornado missiles from further evaluation.

As noted in the response to RAI 05.a, 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.

Response to RAI 05.b:

As noted in the clarifications to Section 3.2.4 of the LAR, as described above, the NSPM categorization process will not deviate from the guidance presented in NEI 00-04 for the evaluation of other external events hazards.

Response to RAI 05.c:

As noted in the responses to parts a and b of RAI 05, all external hazards (other than seismic) were shown to be screened out on the basis of low risk. Using the guidance of Figure 5-6 of NEI 00-04, categorization evaluations will consider the potential for creation of unscreened scenarios.

RAI 06 – Incorporation of FLEX Into the PRA Model(s)

There are several challenges to incorporating FLEX strategies into PRA models that need to be addressed. 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” (ADAMS Accession No. ML17031A269), provides the NRC’s staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision making in accordance with the guidance of RG 1.200.

a. State whether FLEX equipment and strategies have been credited in the PRA. If their inclusion is not expected to impact the PRA results used in the categorization process provide [a] brief statement to confirm the PRA results are not impacted. If not incorporated no additional response is requested.

b. If the equipment or strategies have been credited, and their inclusion is expected to impact the PRA results used in the categorization process please provide the following information separately for the IEPRA, FPRA, external hazards PRA(s), and external hazards screening as appropriate:

i. A discussion detailing the extent of incorporation, i.e., summarize the supplemental equipment and compensatory actions, including FLEX strategies that have been quantitatively credited for each of the PRA models used to support this application.

ii. A discussion detailing the methodology used to assess the failure probabilities of any modeled equipment credited in the licensee's mitigating strategies (i.e., FLEX). The discussion should include a justification explaining the rational for parameter values, and whether the uncertainties associated with the parameter values are considered in accordance with ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2.

iii. A discussion detailing the methodology used to assess operator actions related to FLEX equipment and the licensee personnel that perform these actions. The discussion should include:

- A summary of how the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of ASME/ANS RA-Sa-2009 are evaluated.

- Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and if the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of ASME/ANS RA-Sa-2009.

- If the procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

iv. ASME/ANS RA-Sa-2009 defines PRA upgrade 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. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of ASME/ANS RA-Sa-2009.

1. Provide an evaluation of the model changes associated with incorporating mitigating strategies, which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences, OR

2. Propose a mechanism to ensure that a focused-scope peer review is performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to Capability Category II prior to implementation of the 10 CFR 50.69 categorization program. An example mechanism would be a table of listed implementation items referenced in a license condition.

NSPM Response

The MNGP PRA models will not credit FLEX equipment and strategies during the categorization process. Since FLEX equipment and strategies will not be credited in the PRA models for the categorization process, no additional response is required for part b of RAI 06. In the future, once the issues identified in 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 25) are resolved, FLEX equipment and strategies may be used during the categorization process in accordance with the NRC accepted resolution.

RAI 07 – Proposed License Condition

The guidance in NEI 00-04 allows licensees to implement different approaches, depending on the scope of their PRA (e.g., the approach if a seismic margins analyses is relied upon is different and more limiting than the approach if a seismic PRA is used). RG 1.201, Revision 1, states that "as part of the NRC's review and approval of a licensee's or applicant's application requesting to implement 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."

Section 2.3 of the LAR proposed the following License Condition:

> *NSPM is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3 and RISC-4 structures, systems and components specified in the license amendment request dated March 28, 2018.*
>
> *Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).*

The proposed license condition does not explicitly address the PRA and non-PRA approaches that were used. Provide a license condition that explicitly address the approaches, e.g.:

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

to evaluate risk associated with internal events, including internal flood, internal fire, external flood, and high winds; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (AN0-2) 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 other external hazards using the IPEEE Screening Assessment for External Hazards, and seismic margin analysis (SMA) used to evaluate seismic risk; as specified in License Amendment No. [XXX] dated [XXXX].

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

Note that if implementation items are identified, the license condition will need to be expanded to address them.

<u>NSPM Response</u>

NSPM proposes the following license condition:

NSPM is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (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 process 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 process specified within the license amendment request dated March 28, 2018, and discussed within a response to a request for additional information dated [DATE].

RAI 08 – Integrated One-Top PRA Hazards Model

NEI 00-04, Section 5.6, "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 seismic PRAs) by the fraction of the total core damage frequency [or large early release frequency] contributed by that contributor." The guidance provides formulas to compute the integrated Fussell-Vesely (FV), and integrated Risk Achievement Worth (RAW).

To address the integration of importance measures, some licensees have updated their PRA model to a one-top model that integrates the PRA model(s) across all hazards (i.e., internal events, internal flood, internal fire, seismic, high winds, external flood).

To confirm that the importance measures generated for use in the 10 CFR 50.69 process is consistent with the NEI guidance and does not inadvertently introduce a deviation from the computations for FV and RAW provided in the NEI 00-04 guidance, as endorsed by RG 1.201, Revision 1:

a. Explain whether the PRA model that will be used in the 10 CFR 50.69 categorization process is an integrated one-top model across multiple PRA hazards and if the integrated one-top model includes accident sequence(s) modeling to support quantification of both CDF and LERF. If using an integrated one-top model across multiple PRA hazards for the 50.69 categorization process, provide the following:

    i. 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).

    ii. Discuss how the individual importance measures (i.e., FV 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 NEI guidance. If the practice or method used to generate the integrated importance measures is determined to deviate from the NEI guidance, justify why the integrated importance measures computed are appropriate for use in the categorization process.

<u>NSPM Response</u>

NSPM's use of a one-top model for MNGP provides the advantages of having a seamless single model that computes overall risk to the plant with less chance of inconsistencies that could occur if two models were maintained separately. The MNGP one-top model will be used for the 50.69 categorization process.

i. Although assembled into a one-top model, 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.

ii. An integrated one-top model can be used while setting non-applicable initiating events to FALSE (e.g., set Fire initiators to false in the Internal Events Base Case). This process ensures that importance measures for each categorization step are quantified accurately and not skewed by initiators not intended for that categorization step. The 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 FV or RAW is based on the impact on all hazards.

RAI 09 – Implementation Items

Attachment 3, “Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process,” of the LAR provides dispositions for the self-assessment open items and the remaining open F&Os from the peer reviews of the IEPRA (includes internal flood) and FPRA that were not closed by the August and October 2018 Independent Assessments performed for F&O closure. Several of the dispositions for the F&Os and/or open items (i.e., 2-1, 3-6, 4-11, 4-33, 6-3, 6-9, 6-11, 7-3, 7-4, and FO-1) state in part, “[t]he closure review team recommendations will be addressed.”

Propose a mechanism that ensures these activities and changes will be resolved prior to implementing the categorization process. This mechanism should also include additional actions identified in response to RAIs 01.a through 01.h and specify, how the F&Os and open items will be resolved in the PRA. An example would be a table of listed implementation items referenced in a license condition.

As an alternative to providing an implementation item for an F&O or open item, please demonstrate that the F&O will have no adverse impact or insignificant impact on the 10 CFR 50.69 categorization process.

NSPM Response

NSPM has resolved the additional actions identified in response to RAIs 01.a through 01.h as noted in the response to RAI 01. Therefore, there are no implementation items to be identified and resolved in response to RAI 09.

3.0 REFERENCES

1. Letter (L-MT-18-010) from NSPM to the NRC, "Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors", dated March 28, 2018 (ADAMS Accession No. ML18087A323)

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

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

4. Email from the NRC to NSPM, "Request for Additional Information RE: Monticello License Amendment Request to Adopt 10 CFR 50.69", dated January 31, 2019 (ADAMS Accession No. ML19031A913)

5. NRC Frequently Asked Question (FAQ) 12-0064, "Hot/Work/Transient Fire Frequency: Influence Factors", Revision 1, dated September 5, 2012 (ADAMS Accession No. ML122550050)

6. Letter (L-MT-18-058) from NSPM to the NRC, "Response to Request for Additional Information: Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (EPID L-2017-LLA-0434)", dated October 23, 2018 (ADAMS Accession No. ML18296A653)

7. EPRI Technical Report TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments", dated December 2008

8. EPRI Technical Report TR-1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty", dated December 2012

9. NRC NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making", Revision 1, dated March 2017 (ADAMS Accession No. ML17062A466)

10. Letter (L-PI-18-012) from NSPM to the NRC, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors'", dated July 20, 2018 (ADAMS Accession No. ML18204A393)

11. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Revision 3, dated January 2018 (ADAMS Accession No. ML17317A256)

12. EPRI Technical Report 3002000079, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments", Revision 3, dated April 2013

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

14. NRC NUREG/CR-6850 Volume 2, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology", dated September 2005 (ADAMS Accession No. ML15167A411)

15. NRC NUREG/CR-6850 Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements", dated September 2010 (ADAMS Accession No. ML15167A550)

16. NRC NUREG/CR 7150 Volume 1, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure", dated October 2012 (ADAMS Accession No. ML12313A105)

17. NRC NUREG/CR 7010 Volume 1, "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), Phase 1: Horizontal Trays", dated July 2012 (ADAMS Accession No. ML12213A056)

18. NRC NUREG-2178 Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume", dated April 2016 (ADAMS Accession No. ML16110A140)

19. Letter from Northern States Power Company (NSP) to the NRC, "Submittal of Monticello Individual Plant Examination of External Events (IPEEE) Report", dated March 1, 1995 (ADAMS Legacy Accession No. 9503090231)

20. Letter (L-MT-10-080) from NSPM to the NRC, "Correction to Appendix C of the Monticello Individual Plant Examination of External Events", dated December 20, 2010 (ADAMS Accession No. ML103610324)

21. NRC Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants", Revision 1, dated March 2007 (ADAMS Accession No. ML070360253)

22. NRC NUREG/CR-4461, “Tornado Climatology of the Contiguous United States”, Revision 1, dated February 2007 (ADAMS Accession No. ML070810400)

23. EPRI NP-2005, “Tornado Missile Simulation and Design Methodology, Volume 1: Simulation Methodology, Design Applications, and TORMIS Computer Code”, dated August 1981

24. EPRI NP-2005, “Tornado Missile Simulation and Design Methodology, Volume 2: Model Verification and Data Base Updates”, dated August 1981

25. NRC Memorandum, “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”, dated May 30, 2017 (ADAMS Accession No. ML17031A269)