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May 15, 2019

L-MT-19-030
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

Supplement to a Response for a 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 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'", (L-MT-18-010) dated March 28, 2018 (ADAMS Accession No. ML18087A323)

2) Email from 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)

3) Letter from NSPM to the NRC, "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)", (L-MT-19-018) dated March 13, 2019 (ADAMS Accession No. ML19072A298)

4) NRC Notice of Public Meeting with NSPM, "Public Teleconference with Xcel Energy to Discuss the Monticello Nuclear Generating Plant License Amendment Request to Adopt Section 50.69 of Title 10 of the Code of Federal Regulations (EPID L-2018-LLA-0076)", dated April 26, 2019 (ADAMS Accession No. ML19116A125)

On March 28, 2018, the Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), requested a license amendment (Reference 1) 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", 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 issued a Request for Additional Information (RAI) on January 31, 2019 (Reference 2). On March 13, 2019, NSPM provided a response to the NRC RAI (Reference 3).

On May 8, 2019, a public teleconference (Reference 4) was held with the NRC to discuss NSPM's response to the RAI. NSPM is providing this supplement to address an NRC request for clarification with respect to RAI 02 discussed during the call. This supplement supersedes and completely replaces the response provided for RAI 02 within the March 13, 2019, letter.

In addition, the enclosure includes a revision to the proposed license condition submitted in response to RAI 07 in the March 13, 2019, letter. This revision supersedes and completely replaces the prior proposed license condition.

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

Should you have questions regarding this letter, please contact Mr. Richard Loeffler at (612) 342-8981.

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 May 15, 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

<u>ENCLOSURE</u>

MONTICELLO NUCLEAR GENERATING PLANT

SUPPLEMENT TO A RESPONSE FOR A 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”

(8 pages follow)

Supplement to a Response for a 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

On March 28, 2018, the Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), requested a license amendment (Reference 1) 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). Section 3.1.1 of the license amendment request (LAR) states that NSPM will implement the risk categorization process in accordance with Nuclear Energy Institute (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 issued a Request for Additional Information (RAI) on January 31, 2019 (Reference 4). On March 13, 2019, NSPM provided a response to the NRC RAI (Reference 5).

On May 8, 2019, a public teleconference (Reference 6) was held with the NRC to discuss NSPM's response to the RAI. NSPM is providing this supplement to address an NRC request for clarification with respect to RAI 02 discussed during the call. This supplement supersedes and completely replaces the response provided for RAI 02 within the March 13, 2019 letter.

Also, this enclosure includes a revision to the proposed license condition submitted in response to RAI 07 in the March 13, 2019, letter. This revision supersedes and completely replaces the prior proposed license condition.

2.0 REVISED RESPONSES TO THE REQUEST FOR ADDITIONAL INFORMATION

RAI 02 – Identified Key Assumptions and Sources of Uncertainties

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 [RG] 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 Electric Power Research Institute (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 (Reference 9) 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 10).

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 11). 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 system, structure, and component (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-1855, Revision 1, 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 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.

Assumption/Uncertainty	Discussion	Disposition
A minimum value for a single pre or post initiator Human Error Probability (HEP) was assumed to be 1.0 E-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.
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.

RAI 07 – Proposed License Condition

NSPM Response

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).

3.0 REFERENCES

1. Letter 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’”, (L-MT-18-010) dated March 28, 2018 (ADAMS Accession No. ML18087A323)

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

3. NRC Regulatory Guide (RG) 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 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. Letter from NSPM to the NRC, “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)”, (L-MT-19-018) dated March 13, 2019 (ADAMS Accession No. ML19072A298)

6. NRC Notice of Public Meeting with NSPM, “Public Teleconference with Xcel Energy to Discuss the Monticello Nuclear Generating Plant License Amendment Request to Adopt Section 50.69 of Title 10 of the Code of Federal Regulations (EPID L-2018-LLA-0076)”, dated April 26, 2019 (ADAMS Accession No. ML19116A125)

7. Electric Power Research Institute (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. American Society of Mechanical Engineering (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

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

11. Letter 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’”, (L-PI-18-012) dated July 20, 2018 (ADAMS Accession No. ML18204A393)

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