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April 29, 2019

L-PI-19-014
10 CFR 50.90
10 CFR 50.69

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

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

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-0196)

References:

1) 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)

2) Email from the NRC to NSPM, "Request for Additional Information RE: Prairie Island 50.69 Amendment Request", dated February 26, 2019 (ADAMS Accession No. ML19057A165)

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 Prairie Island Nuclear Generating Plant, Units 1 and 2. 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. Peter Gohdes at (612) 330-6503 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 April 29, 2019.

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

Enclosure

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

**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 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 Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2. 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 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 1 – Proposed License Condition</u>**

10 CFR 50.69(b)(2)(ii) requires that a LAR to implement 50.69 include a "description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant…(including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques)… are adequate for the categorization of SSCs."

10 CFR 50.69(c)(1)(i) and (ii) 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 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 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 3 of the LAR states that the licensee has submitted a separate LAR dated May 18, 2018 (ADAMS Accession No. ML18138A402), as supplemented on July 10, 2018

(ADAMS Accession No. ML18074A308), requesting revision of the license condition associated with implementation of NFPA 805. Table A-1 of this LAR, “Table A-1 - Risk Significant Modifications Related to Implementation of NFPA 805” lists several risk-significant plant modifications that are credited in the fire PRA model but which are not yet installed in the plant. Because of the NRC staff’s concurrent review of a separate LAR to revise the NFPA 805 modifications, the list provided in Table A-1 may yet change. Therefore, Table A-1 may not contain all modifications that would affect the plant PRA models. The NRC staff notes that the fire PRA model used for SSC categorization shall reflect the as-built, as-operated plant. This can be accomplished by completion of all NFPA 805 required modifications that affect PRA models or, if not all of these modifications are completed, by ensuring the as-built, as-operated plant PRA risk results satisfy all RG 1.174 acceptance guidelines.

Section 2.3 of the LAR Enclosure 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 (SSCs) specified in the license amendment request dated July 20, 2018.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).
>
> NSPM shall complete the modifications listed in Table A-1 of the license amendment request dated July 20, 2018, prior to implementation.

The proposed license condition does not explicitly address the use of PRA and non-PRA approaches or provide assurance that PRA models used for SSC categorization reflect the as-built, as-operated plant. The final paragraph proposed in the license condition in the LAR references the modifications listed in Table A-1 of the LAR. RG 1.174 guidance is that the PRA used to support an application is technically acceptable and reflects the as-built, as-operated plant. The LAR provided insufficient information for the NRC staff to confirm that the sub-set of NFPA-805 modifications in Table A-1 were necessary and sufficient, and notes that additional method or plant changes may be required to yield a technically acceptable PRA that reflects the as-built, as-operated plants.

Therefore, the staff has included a general statement in the final paragraph of the sample licensee condition provided below that is intended to ensure that all changes that are required to complete the transition to NFPA 805 and that are also modelled in the PRA are completed prior to implementation of the 50.69 categorization process.

Provide a license condition that explicitly addresses all the categorization approaches used by the staff and all the NFPA 805 changes that affect the PRA, 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 flooding and fire; 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 updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. [XXX] dated [DATE].

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

NSPM shall [ensure that the fire PRA model used for SSC categorization reflects the as-built, as-operated plant using the fire PRA and plant configuration that will be accepted to support final NFPA-805 implementation for both PINGP units at the time of the 50.69 categorization] prior to implementation.

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

<u>NSPM Response to RAI 1</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 PINGP License Amendment No. [XXX] dated [DATE].

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

NSPM will complete the implementation items listed in Attachment 1 of NSPM's letter to the NRC dated [DATE], prior to implementation of 10 CFR 50.69.

NSPM shall ensure that the fire PRA model used for SSC categorization reflects the as-built, as-operated plant for both PINGP units prior to implementation of 10 CFR 50.69.

**<u>RAI 2 – Open/Partially Open Findings in the Process of Being Resolved</u>**

10 CFR 50.69(c)(i) 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.

10 CFR 50.69(b)(2)(iii) 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.

The licensee used RG 1.200, Rev. 2, which describes one acceptable approach to determine if a PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that it can be used in regulatory decision making for light-water reactors. Section 4.2 of RG 1.200 states that the LAR should include a discussion of the resolution of the peer review facts and observations (F&Os) that are applicable to the parts of the PRA required for the application. This discussion should take the following forms:

- A discussion of how the PRA model has been changed and

- A justification in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same) by the particular issue.

Attachment 3 of the LAR, "Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items," provides finding-level F&Os that are still open or only partially resolved after the F&O closure review.

F&O ES-C1-01 and FSS-D7-01 resolutions involve updates to the fire PRA model of record (MOR). Attachment 2 of the LAR states the fire PRA MOR used for risk categorization is Revision 5.3-APP1. Address the following related to the resolutions of ES-C1-01 and FSS-D7-01

i. ES-C1-01 states that "[c]riteria have been provided for a minimum level of redundancy and diversity to meet the intent of the ASME PRA standard with respect to determining if instrumentation needs to be modeled."

   Identify the source of these criteria or, if developed by the licensee, summarize and justify the criteria.

ii. FSS-D7-01 states that the new fire PRA "incorporated the updated unreliability for the pre-action suppression system. The process used to calculate the fire ignition frequencies for structural steel fire scenarios was re-performed. The non-suppression probability for the deluge systems was revised to correct the identified errors."

Identify the source of the updated unreliability or, if developed by the licensee, summarize and justify the values.

NSPM Response to RAI 2.i

The PINGP internal fire PRA criteria for screening instrumentation was developed by NSPM to meet the intent of NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology", Section 2.5.5.2 (Reference 5). The minimum redundancy and diversity criteria used for screening instrumentation from modeling is as follows:

- Transmitters may be screened when a minimum of three channels/divisions powered by separate channels/divisions of power are present or if at least two diverse methods of sensing plant conditions exist (i.e., pressure switch and pressure transmitter). It is extremely unlikely that three separate transmitters would be affected due to physical separation or two diverse methods of sensing would be failed.

- Similar control room indications may be screened when a minimum of three channels/divisions powered by separate channels/divisions of power are present. In this case, it is extremely unlikely that three channels would be affected.

- Dissimilar control room indication may be screened if a minimum of two diverse indication types (i.e., annunciator and level indication) exist with diverse power supplies.

- When procedures instruct operators to first verify the plant condition by local observation or other very reliable means before taking action, the instrumentation can be screened.

NSPM Response to RAI 2.ii

Pre-Action system detection and suppression unreliability for the systems credited in the structural steel fire scenarios is based on the generic failure probability as given in NUREG/CR-6850, which is 5.00E-02 for detection and suppression.

**RAI 3 – Other External Hazards**

10 CFR 50.69(b)(2)(ii) requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation is adequate for the categorization of SSCs.

Section 3.2.4 of the LAR states that Attachment 4 provides a summary of the other external hazards screening. The Attachment 4 entry for External Flooding hazard states the PINGP analysis was recently updated in response to a 50.54(f) request and this hazard remains bounded by the current licensing basis. The entry continues to state an additional analysis was performed for local intensity precipitation (LIP) and determined that PINGP has effective flood protection based on, "available physical margin and the reliability of protection features."

Provide clarification if the protection features used to screen LIP are included in the risk categorization process. Include in this discussion if the categorization for these SSCs will be in accordance with Figure 5.6 of NEI 00-04.

NSPM Response to RAI 3

Plant SSCs that are credited for local intensity precipitation (LIP) will be evaluated in accordance with the guidance of Figure 5-6 of NEI 00-04 to ensure that no unscreened scenarios are created.

**RAI 4 – SSCs Categorization Based on Other External Hazards**

10 CFR 50.69(b)(2)(ii) requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation is adequate for the categorization of SSCs.

Section 3.2.4 of the LAR states, "[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. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered high safety significance (HSS). All remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process."

The last sentence implies that the assessment has been completed and concludes that all other external hazards will never need evaluation during categorization. The individual plant examination of external events (IPEEE) screening process did not include the additional step illustrated in Figure 5-6 in Section 5.4 of NEI 00-04. Figure 5-6 and its associated text states that an evaluation is performed to determine if there are components being categorized that participate in screened external event scenarios whose failure would result in an unscreened scenario.

Clarify how the screening criteria in LAR Attachment 5, "Progressive Screening Approach for Addressing External Hazards," satisfy the guidelines that HSS will be assigned to SSCs whose failure would cause a screened external event scenario to become unscreened.

<u>NSPM Response to RAI 4</u>

Section 3.2.4 of the LAR has been modified, as indicated below, to identify which specific external hazards will be reviewed for the potential creation of unscreened accident scenarios as part of the NSPM categorization process. Changes are in italics:

**3.2.4 Other External Hazards**

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

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

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

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

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

## RAI 5 – Shutdown Risk

10 CFR 50.69(b)(2)(ii) requires that quality and level of detail of the systematic processes that evaluate the plant for shutdown is adequate for the categorization of SSCs.

LAR Section 3.2.5 states "SSCs that meet the two criteria (i.e., considered part of a primary shutdown safety system or failure would initiate an event during shutdown conditions described in Section 5.5 of NEI 00-04) will be considered preliminary HSS." NEI 00-04 states that SSCs can be identified as HSS for either criterion. Therefore:

i. Confirm that the proper statement in LAR Section 3.2.5 should be "SSCs that meet one-of-two criteria (i.e., considered part of a 'primary shutdown safety system' or failure would initiate an event during shutdown conditions) described in Section 5.5 of NEI 00-04 will be considered preliminary HSS."

ii. If the correction in part (i) above cannot be confirmed (i.e., only SSCs that meet both NUMARC 91-06 criteria will be considered HSS) then justify how this approach ensures the proper identification of SSCs as having HSS.

NSPM Response to RAI 5

The PINGP LAR Section 3.2.5 was intended to reflect the requirements of NEI 00-04. Therefore, the statement reflected in part i of this RAI is correct. The statement in Section 3.2.5 should be: "SSCs that meet one of-two criteria (i.e., considered part of a 'primary shutdown safety system' or whose failure would initiate an event during shutdown conditions) described in Section 5.5 of NEI 00-04 will be considered preliminary HSS".

**RAI 6 – Key Assumptions and Uncertainties that Could Impact the Application**

10 CFR 50.69(c)(1)(i) 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 that sensitivity studies 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.

Section 3.2.7 of the LAR states that the detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2009 (Revision 0) (ADAMS Accession No. ML090970525) and Section 3.1.1 of EPRI Technical Report (TR)-1016737. The NRC staff notes that one of these sources has been superseded. Revision 1 of NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," dated March 2017

(ADAMS Accession No. ML17062A466) references updated EPRI guidance in TR-1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty" (2012).

The NRC staff notes that Stages C, D, E, and F of NUREG-1855 (Revision 1) provides guidance on how to identify key sources of uncertainty relevant to the application.

LAR Section 3.2.7 states that "Only those assumptions or sources of uncertainty that could significantly impact the risk calculations were considered key for this application," and are provided in Attachment 6. Attachment 6 of the LAR contains 10 key assumptions/sources of uncertainties. The LAR does not describe how the key assumptions and sources of uncertainty were identified, and whether the outcome described in the LAR was the result of a comprehensive examination for key assumptions and sources of uncertainty using recent industry guidance.

i. Provide a description of the process used to determine how the candidate key assumptions and sources of uncertainty were identified and evaluated for the internal event (including internal flooding) and fire PRAs. Include in the discussion explanation of how uncertainty issues associated with plant specific features, modeling choices, and generic industry concerns were addressed. Also, include in the description explanation of whether the assumptions and sources of uncertainty documented in the PRA modeling notebooks were reviewed to determine if they could have a possible impact on the application.

ii. Describe how the process described in Part (i) above is consistent with NUREG-1855, Revision 1, or another NRC-accepted method.

iii. If the process of identifying key assumption or sources of uncertainty for these PRA models cannot be justified, provide the results of an updated assessment of key sources of uncertainty or assumptions. Include a description of the specific assumptions and sources of uncertainty key to this application in enough detail so that its impact on the application can be clearly understood and a specific sensitivity study could be defined to examine the impact on 50.69 categorization.

iv. If the response to part (iii) above results in the identification of key assumptions or sources of uncertainty that should be addressed as part of the 50.69 categorization then propose a mechanism to ensure that the identified sensitivity study is performed as part of PINGP's 50.69 categorization process.

<u>NSPM Response to RAI 6.i</u>

The sources of uncertainty evaluation for the internal events PRA considers both plant-specific sources of uncertainty and the generic uncertainties identified in EPRI TR-1016737 (Reference 6). At the time of the LAR submittal, 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 7). However, a subset of these generic uncertainties were considered within the context of the plant-specific uncertainty evaluation. Both modeling uncertainty and completeness uncertainty issues were examined for both PRAs.

The internal events PRA includes an evaluation of the sources of uncertainty. All identified sources of uncertainty were compiled and characterized in the Uncertainty Notebook by reviewing modeling assumptions from all of the PINGP PRA technical element notebooks for the base case models using an approach that is consistent with the ASME/ANS RA-Sa-2009 (Reference 8) 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).

The fire PRA sources of uncertainty evaluation has been similarly updated to compile and characterize plant-specific assumptions and associated sources of model uncertainty as well as the generic sources of uncertainty presented in EPRI TR-1026511 based on the most recent fire model update. This evaluation meets the intent of steps C-1 and E-1 of NUREG-1855, Revision 1.

To assess the impact of sources of uncertainties on 10 CFR 50.69 system categorizations, a review of the base case sources of uncertainty for the internal events and fire PRAs was performed prior to the submittal of the original LAR and has now been updated based on the latest PRA sources of uncertainty evaluations. Each identified uncertainty was evaluated with respect to its potential to significantly impact the risk ranking evaluations that will be performed to support the categorization effort. Previously identified sources of uncertainty for internal events were investigated further and most were determined to have negligible impacts on the 50.69 process. See the responses to RAI 7 concerning the uncertainties that were removed. A single key uncertainty remains that could impact the 50.69 categorization process. Similarly, the updated fire PRA uncertainty review against the current fire PRA model determined that there are no relevant sources of uncertainty that pertain to the use of the fire PRA for the 50.69 categorization process. This evaluation meets the intent of the screening portion of steps C-2 and E-2 of NUREG-1855, Revision 1.

Based on the updated evaluation, Attachment 6 of the LAR is replaced in its entirely with the following:

**Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty**

| Assumption/Uncertainty | Discussion | Disposition |
|---|---|---|
| Internal Events PRA Model Key Assumptions/Sources of Uncertainty | | |
| Basis for Human Error Probabilities (EPRI-identified generic source of modeling uncertainty) | Human Reliability Analysis (HRA) is a continually evolving discipline. The human error probabilities were obtained using the current EPRI HRA calculator consistent with the Fire HRA Methodology described in NUREG-1921 (Reference 10). The internal events human error probabilities were obtained using guidance from NUREG/CR-1278 (Reference 11), and NUREG/CR-4772 (Reference 12).<br><br>Given the methodologies used, the impact of any remaining uncertainties is expected to be small. | The PINGP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of uncertainty.<br><br>As directed by NEI 00-04, human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values as part of the required 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled Human Error Probabilities are accounted for in the 50.69 application. |
| Fire PRA Model Key Assumptions/Sources of Uncertainty | | |
| None Identified | | |

NSPM Response to RAI 6.ii

As discussed in the response to part i of RAI 6 above, the approach used to identify the sources of uncertainty for consideration meets the intent of NUREG-1855, Revision 1 for the steps identified (C-1, C-2, E-1, and E-2).

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 PRA acceptance guidelines are threshold values for Fussell-Vesely (FV) and Risk Achievement Worth (RAW) for each SSC being categorized, above which the SSC is categorized as candidate high safety significant (HSS), and below which the SSC is categorized as candidate 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 FV and RAW importance measures for all components (SSC), every relevant uncertainty/assumption has the potential to challenge the acceptance guidelines. That is, since RAW and FV 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 FV for every SSC. Thus, the only way to evaluate the impact of a sensitivity is to quantify the sensitivity case and compare the FV and RAW values for all SSCs against the base case FV 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 NUREG-1855 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. NUREG-1855, 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 study 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 probabilities of all LSS SSCs are increased at the same time in the sensitivity study, 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 9) are not exceeded. Thus, the LSS sensitivity demonstrates that the potential impact of all uncertainties/assumptions is acceptable.

The performance monitoring program established as part of the 10 CFR 50.69 process (per NEI 00-04, Section 12) 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.

NSPM Response to RAI 6.iii

As documented in the responses to parts i and ii of RAI 6, the approach used to identify significant sources of uncertainty for the 10 CFR 50.69 application and to evaluate their impacts is adequate and appropriate to ensure that these sources of uncertainty will not impact the system categorization evaluations for SSC risk significance.

NSPM Response to RAI 6.iv

This question is not applicable to PINGP based on the information provided in response to parts i, ii, and iii of RAI 6.

**RAI 7 – Dispositions of Possible Key Assumptions and Uncertainties**

10 CFR 50.69(c)(1)(i) and (ii) 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 importance of components. NEI 00-04 guidance states that additional “applicable sensitivity studies” from characterization of PRA adequacy should be considered.

The dispositions to the 10 assumption/uncertainty items identified in Attachment 6 of the LAR can be summarized as follows: (1) four state that sensitivity studies will be performed “as necessary,” (2) four (including all three for the fire PRA model) conclude that it is not a source of uncertainty for the application, (3) one is addressed by the general NEI 04-10 human error probability (HEP) sensitivity analysis, and (4) one will be treated on a “case-by-case basis as needed.”

i. Explain how a sensitivity study will be determined to be necessary and “as needed.”

ii. The sixth uncertainty item in Attachment 6 concerns thermally-induced steam generator tube rupture (TI-SGTR). The LAR states that TI-SGTR is primarily a phenomenological uncertainty for Large Early Release Frequency (LERF) and that its impact on LERF is low enough such that no impact on 50.69 categorization is expected. However, the LAR also states that TI-SGTR can be significant for some non-station black out sequences.

    a. Provide justification, such as a sensitivity study, that the exclusion of a TI-SGTR does not impact 50.69 categorization for any SSCs.

    b. Alternatively, propose a mechanism to ensure this issue is addressed as a sensitivity study during the 50.69 categorization process.

<u>NSPM Response to RAI 7.i</u>

The specific uncertainties dispositioned in Attachment 6 of the LAR as requiring sensitivity studies “as needed” are primarily items involving exclusion of low probability failure modes in the modeling of specific plant systems, such as Residual Heat Removal (RHR), Main Feedwater, Circulating Water, etc. As the excluded items were judged to be low probability contributors to overall core damage frequency (CDF) and large early release frequency (LERF), it is highly unlikely that risk rankings of components in other systems would be significantly impacted by updates to the system models to refine or eliminate these uncertainties. In addition, the sensitivity studies required by NEI 00-04 that include increasing the failure probabilities of all LSS components by an additional factor of 3 to 5 should bound any change in risk rankings due to these specific uncertainties. Lastly, because the HSS/LSS determination is made for each system’s components using multiple evaluation methods (PRA, passive, defense-in-depth, functional, etc.), if the system components that would be impacted by the identified uncertainty were already determined to be HSS due to any of the required evaluation methods, then an additional system-level sensitivity evaluation would be unnecessary.

The treatment of uncertainties pertaining to thermally-induced steam generator tube rupture (SGTR) is addressed in the response to RAI 7 part ii.

The uncertainty associated with the assumption that low pressure RHR piping will always rupture upon exposure to Reactor Coolant System (RCS) pressure, resulting in an interfacing system loss of coolant accident (ISLOCA) event, is a conservative treatment of ISLOCA impacts. ISLOCA is a small contributor to CDF and a key contributor for LERF. Reducing the probability of pipe rupture given an ISLOCA would reduce the importance of the ISLOCA contributors and possibly increase the risk significance of other SSCs, especially for LERF. A sensitivity study was performed that reduced the probability of a low pressure pipe rupture on exposure to RCS pressure from 1.0 to 0.05. The results of the study identified no events across both units that could transition from LSS to HSS on the basis of the risk results that would not already be identified as HSS as a result of the base importances or the NEI 00-04 required sensitivity studies. Therefore, no additional sensitivity evaluations would be necessary.

<u>NSPM Response to RAI 7.ii.a</u>

For the thermally-induced failure of a steam generator tube, the PINGP PRA model follows WCAP-16341-P, “Simplified Level 2 Modeling Guidelines” (Reference 13), which provides a basis for updated Level 2 analyses. This WCAP provides a common, standardized method for PWRs with large dry containments. The guidance particularly addresses the latest understanding for thermally-induced steam generator tube ruptures and other Level 2 issues.

The text included in the “Discussion” column of Attachment 6 of the LAR was derived from the overall assessment of all base PRA sources of uncertainty that is documented in the internal events PRA uncertainty notebook. The text that was developed for that notebook was specific to applications with a focus on SGTR and is not applicable to 10 CFR 50.69. As noted in the “Disposition” column of Attachment 6 of the LAR, this was judged to not represent a key source of uncertainty in the 50.69 application since CDF would not be impacted and the overall impacts on LERF were expected to be small.

Since the PINGP PRA uses a TI-SGTR modeling approach that is consistent with recent industry approaches and is appropriate for determination of LERF, this does not represent a key source of uncertainty in the 10 CFR 50.69 application.

<u>NSPM Response to RAI 7.ii.b</u>

As discussed in the RAI 7.ii.a response, the TI-SGTR modeling treatment is consistent with industry methods, therefore, no additional mechanism is needed to address this issue.

## <u>RAI 8 – Flowserve N9000 RCP and Abeyance Seal Modeling</u>

10 CFR 50.69(c)(1)(i) and (ii) 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 importance of components. NEI 00-04 guidance states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

Section 3.3 of the LAR states a focused-scope peer review was conducted to address the incorporation of Flowserve N9000 Reactor Coolant Pump seals. The disposition to F&O SY-A17-01 in Attachment 3 of the LAR states, "the N-9000 RCP seal model must obtain NRC review and approval." The NRC staff notes that the N-9000 RCP seal model is approved for Combustion Engineering (CE) plants using the guidance of WCAP-16175-P-A with conditions, limitations, and modifications in the NRC staff safety evaluation (SE) (ADAMS Accession No. ML071130383). The staff's SE that "[a]dditional conditions, limitations, and modifications are provided in this SE to address some of the issues that must be addressed by application of TR WCAP-16175-P, Revision 0, RCP seal failure model to non-CE plants." The LAR did not address if the PINGP PRA model implementation used all applicable guidance in this WCAP.

The NRC staff also notes that abeyance seals are sometimes used as a backup to Flowserve RCP seal packages to limit leakage if excessive flow from the mechanical face seals occurs (ADAMS Accession No. ML15222A357). There is currently no NRC accepted methodology to model the abeyance seal in PRAs.

In light of these observations:

i. Confirm that the PRA model implementation of the N-9000 RCP seal was in accordance with WCAP-16175-P-A and addressed all applicable NRC staff conditions, limitations, and modifications as described in the associated safety evaluation. Alternatively, describe the methodology used to model the N-9000 RCP seal and justify that this methodology is acceptable.

ii. If the baseline PRA model of record used for this LAR credits an abeyance RCP seal, provide the PRA methodology to model the abeyance seal and describe how this inclusion impacts the categorization.

iii. Propose a mechanism that ensures an NRC approved abeyance RCP seal model is available before incorporation of an abeyance seal into the PRA MORs.

<u>NSPM Response to RAI 8.i</u>

NSPM has completed a review of the PINGP PRA model implementation of the N-9000 RCP seal and confirmed that it was completed in accordance with WCAP-16175-P-A, "Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants", (see Reference 14 for non-proprietary version) and addressed all applicable NRC conditions, limitations, and modifications as described in the associated safety evaluation.

NSPM Response to RAI 8.ii

The NSPM PRA model used for categorization will only credit the abeyance RCP seal after the NRC has accepted the methodology to model the abeyance seal in PRAs. Prior to methodology acceptance, the PRA model used for categorization will not credit the abeyance seal.

NSPM Response to RAI 8.iii

The proposed mechanism is in the response to RAI 1.

**RAI 9 – Integrated PRA Hazards Model**

10 CFR 50.69(c)(1)(ii) of require that SSC functional importance be determined using an integrated, systematic process. NEI 00-04, Section 5.6, "Integral Assessment," discusses the need for an integrated computation using available importance measures. It further states that the "integrated importance measure essentially weights the importance from each risk contributor (e.g., internal events and fire PRAs) by the fraction of the total core damage frequency [or LERF] contributed by that contributor." The guidance provides formulas to compute the integrated Fussel-Vesely (FV), and integrated Risk Achievement Worth (RAW).

The LAR does not address the integration of importance measures across all hazards (i.e., internal events, internal flooding, and fire). Therefore:

i. Explain how the integration of importance measures across hazards for 50.69 categorization process will be performed and whether it will be performed using an integrated one-top model across multiple PRA hazards.

ii. If an integrated one-top model across multiple PRA hazards will be used, then discuss how the individual importance measures (e.g., FV and RAW) for the PRA one-top model are derived, and justify why the importance measures generated do not deviate from the NEI guidance or Table 3-1 of the LAR. If the practice or method used to generate the integrated importance measures is determined to deviate from the NEI guidance, then justify why the integrated importance measures computed are appropriate for use in the categorization process.

NSPM Response to RAI 9.i

The PINGP internal events and fire PRA models are currently separate models. As such, integration of importance measures across hazards will be performed using the methodology specified in NEI 00-04, Section 5.6. Future PRA model updates may combine the PINGP internal events and fire PRA models into an integrated one-top model and importance measures will be performed as discussed in the response to RAI 9 part ii.

NSPM Response to RAI 9.ii

While an integrated one-top model is not currently used, it could 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) 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 of all hazards.

**RAI 10 – Incorporation of FLEX into the PRA Models**

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. The LAR does not state whether or not the licensee has incorporated FLEX mitigating strategies and associated equipment into the PRA models at PINGP.

Provide the following information separately for internal events PRA, external hazard PRAs, and external hazard screening as appropriate:

i. 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 brief statement to confirm the PRA results are not impacted. If not incorporated no additional response is needed.

ii. 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:

   a. 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.

   b. 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.

c. 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 were 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 not explicit provide a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

d. 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.

Therefore, provide the following:

1. 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. 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 50.69 categorization program. An example mechanism would be a table of listed implementation items referenced in a license condition.

NSPM Response to RAI 10

The PINGP base PRA models do not currently credit FLEX equipment and strategies, therefore, no additional response is required. 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 15) are resolved, FLEX equipment and strategies may be used during the categorization process in the future in accordance with the NRC accepted resolution.

**RAI 12 – Implementation Items**

10 CFR 50.69(b)(2)(ii) requires that a licensee’s application contain a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown are adequate for the categorization of SSCs.

If the responses to RAIs 1 through [10] above require any follow-up actions prior to implementation of the 50.69 categorization process, provide a list of those actions and any PRA modeling changes including any items that will not be completed prior to issuing the amendment but must be completed prior to implementing the 50.69 categorization process. Propose a mechanism that ensures these activities and changes will be completed and appropriately reviewed and any issues resolved prior to implementing the 50.69 categorization process (for example, a license condition that includes all applicable implementation items and a statement that they will be completed prior to implementation of the 50.69 categorization process).

NSPM Response to RAI 12

NSPM proposes the adoption of a license condition as described in the response to RAI 1. This license condition will refer by date to the RAI response letter which will include an attachment detailing the applicable implementation items. Proposed implementation items are listed in Attachment 1 to this Enclosure.

## 3.0 REFERENCES

1. 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)

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: Prairie Island 50.69 Amendment Request”, dated February 26, 2019 (ADAMS Accession No. ML19057A165)

5. 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)

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

7. 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

8. 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)

9. 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-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines”, dated July 2012 (ADAMS Accession No. ML12216A104)

11. NUREG/CR-1278, “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications”, dated August 1983 (ADAMS Accession No. ML071210299)

12. NUREG/CR-4772, “Accident Sequence Evaluation Program Human Reliability Analysis Procedure”, dated February 1987

13. 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)

14. Westinghouse Electric Company WCAP-16175-NP-A, “Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants”, Revision 0, dated March 2007 (ADAMS Accession No. ML071130383)

15. 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)

**ENCLOSURE, ATTACHMENT 1**

**PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2**

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"

**10 CFR 50.69 IMPLEMENTATION ITEMS**

(1 Page Follows)

**Table A.1 – 10 CFR 50.69 Implementation Items**

| No. | Implementation Item |
|---|---|
| 1. | The NSPM PRA model used for categorization will only credit the abeyance RCP seal after the NRC has accepted the methodology to model the abeyance seal in PRAs. Prior to methodology acceptance, the PRA model used for categorization will not credit the abeyance seal. |