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January 31, 2017

L-MT-17-002
10 CFR 50.90

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

Part 2 Response to Probabilistic Risk Assessment (PRA) Related Requests for Additional Information: License Amendment Request for a Permanent Extension of the 10 CFR 50 Appendix J Containment Type A Test Interval (CAC No. MF7359)

References:

1) NSPM (P. Gardner) to NRC, "License Amendment Request: Revise Technical Specification 5.5.11 to Provide a Permanent Extension of the Integrated Leakage Rate (Type A) Test Frequency from Ten to Fifteen Years," (L-MT-16-001), dated February 10, 2016 (ADAMS Accession No. ML16047A272 and ML16047A273)

2) NRC (R. Kuntz) to NSPM (R. Loeffler), "Request for Additional Information RE: Monticello license amendment request for ILRT extension (CAC MF7359)," dated September 9, 2016 (ADAMS Accession No. ML16256A004)

3) NSPM (P. Gardner) to NRC, "Response to Request for Additional Information: License Amendment Request for a Permanent Extension of the 10 CFR 50 Appendix J Containment Type A Test Interval (CAC No. MF7359)," (L-MT-16-044), dated October 10, 2016 (ADAMS Accession No. ML16284A015)

4) NRC (R. Kuntz) to NSPM (R. Loeffler), "Monticello ILRT extension amendment Request for additional information (CAC No. MF7359)," dated November 18, 2016 (ADAMS Accession No. ML16323A242)

5) NSPM (P. Gardner) to NRC, "Part 1 Response to Probabilistic Risk Assessment (PRA) Related Requests for Additional Information: License Amendment Request for a Permanent Extension of the 10 CFR 50 Appendix J Containment Type A Test Interval (CAC No. MF7359)" (L-MT-16-062), dated December 16, 2016 (ADAMS Accession No. ML16355A183)

On February 10, 2016, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, submitted a license amendment request (LAR) proposing a change the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP). The proposed change is to permanently revise the frequency specified in Specification 5.5.11 "Primary Containment Leakage Rate Testing Program", to increase the containment integrated leakage rate test (ILRT) program Type A test interval from 10 years to 15 years (Reference 1).

On September 9, 2016, the U.S. Nuclear Regulatory Commission (NRC) requested additional information pertaining to the primary containment performance history and a clarification of ILRT test results (Reference 2). The responses to these requests for additional information were provided on October 10, 2016, in Reference 3.

From October 13 through 14, 2016, the NRC conducted a regulatory audit to gain a better understanding of the containment accident pressure risk assessment in the MNGP LAR. On November 18, 2016, the NRC requested additional information (RAI) pertaining to probabilistic risk assessment related considerations (Reference 4). The responses to RAIs 2 and 3 were provided by letter on December 16, 2016 (Reference 5). The responses to RAIs 1.b, 1.c, 5.c, 5.d, and 5.e are provided in Enclosure 1. The responses to the remainder of the RAIs will be provided in a future submittal.

<u>Summary of Commitments</u>

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 January 31, 2017.

Peter A. Gardner
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure 1: Part 2 Response to PRA Related Requests for Additional Information
Appendix 1: Application of FLEX for Defense-in-Depth
Appendix 2: Acceptability of Throttling ECCS Pumps to Ensure Adequate NPSH

cc: Administrator, Region III, US NRC
Project Manager, Monticello Nuclear Generating Plant, US NRC
Resident Inspector, Monticello Nuclear Generating Plant, US NRC
State of Minnesota

ENCLOSURE 1

MONTICELLO NUCLEAR GENERATING PLANT

PART 2 RESPONSE TO PROBABILISTIC RISK ASSESSMENT (PRA) RELATED REQUESTS FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST FOR A PERMANENT EXTENSION OF THE 10 CFR 50 APPENDIX J CONTAINMENT TYPE A TEST INTERVAL

(11 pages follow)

PART 2 RESPONSE TO PROBABILISTIC RISK ASSESSMENT (PRA) RELATED REQUESTS FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST FOR A PERMANENT EXTENSION OF THE 10 CFR 50 APPENDIX J CONTAINMENT TYPE A TEST INTERVAL

On February 10, 2016, NSPM submitted a license amendment request (LAR) proposing a change to the Technical Specifications for the Monticello Nuclear Generating Plant (MNGP). The proposed change is to permanently revise the frequency specified in Specification 5.5.11 "Primary Containment Leakage Rate Testing Program", to increase the containment integrated leakage rate test (ILRT) program Type A test interval from 10 years to 15 years. On September 9, 2016, the U.S. Nuclear Regulatory Commission (NRC) requested information pertaining to the primary containment performance and history and a clarification of ILRT test results. The responses to these requests for additional information were provided on October 10, 2016.

From October 13 through 14, 2016, the NRC conducted a regulatory audit to gather information on the probabilistic risk assessment (PRA) related portions of the MNGP LAR to increase the containment ILRT interval from 10 to 15 years. On November 18, 2016, the NRC requested additional information (RAI) pertaining to PRA related considerations (Reference 1). The responses to RAIs 2 and 3 were provided by letter on December 16, 2016 (Reference 2). The responses to RAIs 1.b, 1.c, 5.c, 5.d, and 5.e are provided in this enclosure. The responses to the remainder of the RAIs will be provided in a future submittal.

A discussion is provided in Appendix 2 to this enclosure concerning the acceptability of crediting the operator's ability to throttle the low pressure emergency core cooling system (ECCS) pumps to preserve Net Positive Suction Head (NPSH). This discussion is based in-part on additional guidance in a proposed revision to Regulatory Guide (RG) 1.174 (Reference 3) provided in draft Regulatory Guide (DG)-1285[(1)] (Reference 4).

Also, although the FLEX[(2)] strategies were originally designed to prevent core damage following an extended loss-of-AC-power (ELAP), the strategies may also be beneficial to address other types of initiating events. While no quantitative analysis was performed, a discussion is provided in Appendix 1 to this enclosure considering the application of FLEX to provide defense-in-depth for a situations where there is a loss of containment accident pressure (CAP).

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1. Draft RG DG-1285 (Proposed Revision 3 to Regulatory Guide 1.174, dated May 2011), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"
2. FLEX stands for "Diverse and Flexible Coping Strategies"

RAI 1

The license amendment request (LAR) Section 5.3.1 of Enclosure 2 provides the evaluation of external event contribution.

b. The LAR references the results from the Individual Plant Examination of External Events (IPEEE) for high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards which resulted in screening these events from further consideration. Since the IPEEE was completed in November 1995, assess the current plant for the external event contribution and discuss your assessment performed for the ILRT extension application.

Response to RAI 1b

The MNGP IPEEE (Reference 5) conclusions for high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards were presented as, "previously submitted and approved analyses," and were used to assess an, "order of magnitude estimate for the contribution of the external event to the impact of the changed interval," This approach is consistent with the guidance for PRA applications provided within Section 4.2.7, "External Events," of Electric Power Research Institute (EPRI) Topical Report (TR) 1018243[(3)] (Reference 6), as endorsed by the NRC Safety Evaluation Report (SER) for NEI 94-01 Revision 2-A. An assessment of changes to these external hazards and plant upgrades, made subsequent to the IPEEE, that impact the coping capability/strategy related to these hazards, is presented below.

High Winds and Tornadoes

The High Winds and Tornadoes section of the IPEEE was assessed to determine that the contribution to risk discussed remains applicable to the current plant/environment conditions. Two notable changes have occurred subsequent to the initial IPEEE submittal. First, Appendix C (Other External Events) of the Monticello IPEEE was revised in November 2010, to correct numerical translation errors from a source used to derive tornado missile strike probabilities. Correction of these translation errors did not change the conclusion that, "tornado missiles can be screened from further evaluation." Second, in 2016 plant modifications were completed which added tornado missile protection to the Emergency Diesel Generator (EDG) exhaust/intake, ventilation fans/louvers, and Diesel Fuel Oil System tank vents virtually eliminating these components contribution to risk resulting from tornado missiles.

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3. EPRI Topical Report 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (This report is Revision 2-A of EPRI TR-1009325.)

Additionally, subsequent to performance of the IPEEE, the FLEX strategies have been developed, which incorporate extensive pre-staged equipment that provides additional capability to cope with Beyond Design Basis External Events (BDBEEs); including seismic events, external flooding, high winds, tornadoes, snow and ice storms, extreme cold and extreme heat. FLEX provides independent injection capability by pumps that are external to the primary containment and provides the required supporting equipment (e.g., portable diesel generators) for their operation, providing defense-in-depth. There is no reliance on CAP for the operation of any of the FLEX pumps since they draw upon sources of water that are external to the primary containment.

<u>External Flooding</u>

The Monticello site natural grade level is 930 feet mean sea level (msl).(4) As credited in the Monticello IPEEE at the time, flood protection was provided by the design of the Class I and Class II structures and by installation of flood protection features (e.g., pumps and steel plates, grout, or sandbags to close openings up to elevation 930 feet msl) to provide protection for Class I structures and Class II structures housing Class I equipment. As discussed in USAR Appendix G, "Probable Maximum Flood [PMF] Mississippi at Monticello Minnesota," approximately 12 days are available from the onset of conditions leading to the PMF to the time the peak stage is reached at the site, i.e., 939.2 feet msl.(5) The flood protection program provides time for installation of flood measures and for construction of an earthen levee portion of the flood barrier. Suitable construction materials are maintained at the plant site.

Subsequent to the NRC SER for the MNGP IPEEE submittal (Reference 7), dated April 14, 2000, extensive physical plant upgrades and procedural/process improvements related to external flooding hazards were completed. A permanent section of Bin-Wall and Intake Structure flood barrier (steel plates barrier on top of the structure) has been constructed. This permanent flood barrier will tie into a horseshoe shaped earthen levee and (together with additional Bin-Wall segments) to be constructed around Class I structures, Class II structures housing Class I equipment (e.g., the Turbine Building) and Radwaste Building in the event of a projected flood in exceedance of 930 feet msl, to protect them from the effects of a flood. Additionally, some flood penetrations which previously were intended to be sealed upon imminent threat of flooding have been permanently sealed, the procurement and pre-staging of material and equipment for use in flood protection has occurred, and pre-approved work orders have been developed to be activated upon the threat of flooding.

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4. The maximum recorded high water level was 916 feet msl, and the 1000-year projected high water level elevation is 920 feet msl.
5. The Flood Hazard Reevaluation Report, based on the U.S. Army Corps of Engineers PMF analysis concludes that the combined-effect PMF stage does not reach plant grade of approximately 930 feet msl.

An External Flooding Program has been developed to ensure that credited flood protection features are maintained and that the conditions necessary for accomplishment of credited procedures are met. The program ensures that the permanent and temporary flood protection feature structures, systems, components (SSCs) and procedures needed to install and or operate them during a flood are acceptable and capable of performing their design function, including providing assurance that plant modifications and changes to site topography, do not adversely affect the plant flood protection features. This program also ensures that industry best practices are benchmarked and incorporated, and that evolving regulatory expectations are addressed, including Post-Fukushima Near Term Task Force (NTTF) Recommendation 2.3 for flooding and the related NRC 10 CFR 50.54(f) information request. Additionally, as previously discussed, FLEX strategies have been developed, which incorporate extensive pre-staged equipment and strategies to cope with BDBEEs, including external flooding events.

Also, the recent Flood Hazard Reevaluation Report(6) prepared in response to Post-Fukushima NTTF recommendations, indicated that based on the U.S. Army Corps of Engineers (USACE) PMF analysis, the combined-effect PMF stage does not reach the plant grade of approximately 930 feet msl. Hence, the MNGP is well prepared to mitigate external flooding events.

Transportation and Nearby Facility Accidents

A review of the potential for aircraft accidents was performed to determine if significant changes have occurred to air traffic hazards since performance of the IPEEE. The only item of note, was that the flight frequency for the largest aviation threat, the Pilot Cove Airport due to its proximity (3.14 miles from Monticello), has dropped significantly over the approximately twenty years since IPEEE performance. The airport now only accommodates ultralight aircraft on a short non-paved runway.

A review of the IPEEE assessment of other transportation related accidents including: those associated with railroad; marine (ship/barge); and truck transportation was performed and it was determined that the prior conclusions for these threats remain appropriate. Discussions with Department of Transportation (DOT) and the Department of Homeland Security and with Emergency Management (HSEM) personnel were conducted to identify possible new hazards that were not present when the IPEEE evaluation was developed. Minor differences in highway traffic levels and rail cargo were noted, but no changes were identified that impacted the conclusion of the traffic hazards analysis presented in the IPEEE.

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6. The Flood Hazard Reevaluation Report was prepared in response to a 10 CFR 50.54(f) information request regarding insights from the Fukushima Dai-Ichi accident. The specific value is Security Related Information Withheld under 10 CFR 2.390(d)(1). A redacted version of this report can be found at ADAMS Accession No. ML16145A180.)

Nearby facility development that could potentially pose a threat to the MNGP was evaluated, in-part, through interviews with administrators from Wright County and the City of Monticello. For example, commercial/industrial development in the Otter Creek business park, located several miles away from the plant, was considered, but no cases were identified that would pose a potential hazard to the nuclear plant. Also, it was noted that the Highway 75 railroad overpass near the plant was eliminated, which reduces the potential risk from traffic related accidents.

Other External Hazards

A review of tornado-induced pressure transients and probable maximum precipitation (PMP) calculations revealed that no significant variations have occurred in the threat frequency or the plant design/operation, since the IPEEE submittal date, which could impact the conclusions of the IPEEE associated with these events.

Further, a review of the IPEEE disposition for other external hazards including: External Fires, Extraterrestrial Activity, Volcanic Activity, Earth Movement, Lightning, Ice Storms, Hail Storms, Dust/Sand Storms, and Severe Temperature Transients, indicates no changes have occurred that would require updated analysis to address these phenomena. Furthermore, the recent addition of FLEX capabilities provides for an additional decrease in risk associated with these other external phenomena.

In summary, the external event contribution from external floods, high winds and tornadoes, transportation and nearby facility accidents, or other external hazards, remains small and falls well within the bounding assessment for external events impact used in the LAR such that there is no impact on the ILRT extension application.

c. **Confirm that the fire probabilistic risk assessment (PRA) represents the as built, as-operated plant. If there are plant modifications or procedures that are credited in the fire PRA but not completed, then remove them from the model so that the fire PRA represents the as-built, as-operated plant, and provide the updated core damage frequency (CDF) and LERF for the application.**

Response to RAI 1c

The Fire PRA represents the as-built, as-operated plant. There are no plant modifications that are credited in the Fire PRA that have not been installed in the plant (i.e., no pending credited changes). For the procedure changes that are credited; the procedures were updated prior to the issuance of the Fire PRA used in the ILRT extension PRA analysis.

RAI 5

The NRC staff requests additional information for the following CAP-related key assumptions and uncertainties:

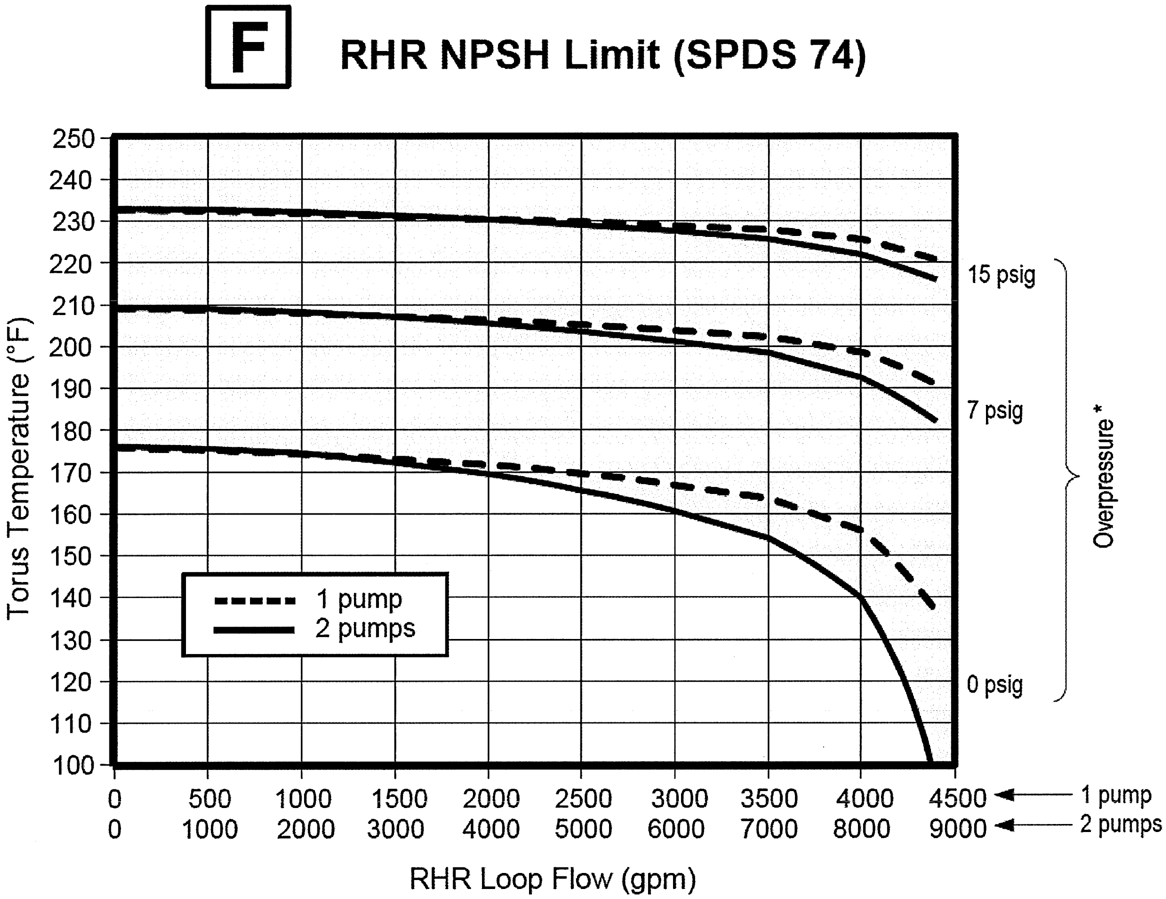
1. **Existing plant conditions impacting NPSH;**
2. **Operator action to throttle low pressure ECCS flow;**
3. **High Pressure Injection (HPI) lube oil cooling;**
4. **Systems available following a loss of instrument air system initiating event;**
5. **Loss of Offsite Power (LOOP)/Station Blackout (SBO) PRA modeling; and,**
6. **Containment leakage rate that would impact CAP**

(The answer to the above portion of this RAI is provided in the response to RAI 5.a, which is being provided in a following submittal.)

c. **The LAR applies PRA credit for operator action to throttle low pressure ECCS flow. Describe the technical basis supporting ECCS throttling as a success criteria for containment conditions which defeat CAP for the PRA mission time. Describe which PRA sequences are credited and not credited.**

Response to RAI 5c

The use of CAP was reviewed extensively under the Extended Power Uprate (EPU) amendment and approved by the NRC. The MNGP Emergency Operating Procedures (EOPs) provide the primary and overriding direction for the use of ECCS pumps taking suction from the torus under emergency conditions. The EOP's include detailed guidance to operate these pumps in such a manner as to allow continued operation and yet preclude failure/damage to the pumps from inadequate NPSH. This guidance explicitly accounts for CAP or the lack thereof, as a function of pump flow and suppression pool temperature. The following figure is an example for the Residual Heat Removal (RHR) system taken directly from the EOP flowchart for Reactor Pressure Vessel (RPV) Control (C.5-1100), and is utilized by the operators to control pump flow such that NPSH limits are not exceeded.



Operators routinely and repetitively practice accommodating NPSH restrictions in the simulator control room under a wide array of accident conditions using the above figure.[7] This figure is duplicated in automated Safety Parameter Display System (SPDS) displays which plot the current plant conditions, thus allowing the reactor operators to easily assess the status of NPSH for the ECCS pumps that are in service and taking suction from the suppression pool.

For the purposes of responding to this RAI, as requested by the NRC, the PRA analysis was adjusted to remove credit for operator throttling of the low pressure ECCS pumps for any of the sequences.

7. A similar figure and guidance exists for the Core Spray, High Pressure Coolant Injection, and Reactor Core Isolation Cooling systems.

d. Explain how the PRA model was adjusted to account for CAP-related impact on high pressure injection (HPI) lube oil cooling, and discuss the justification for PRA credit taken for HPI. If credit is taken for injection from other sources (e.g., condensate storage tank) confirm that it reflects plant operating procedures and is a PRA success criteria for the PRA mission time.

<u>Response to RAI 5d</u>

High Pressure Injection systems credited in the MNGP PRA include High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Feedwater and Control Rod Drive Hydraulics (CRDH). Oil cooling for the Feedwater and CRDH pumps is ultimately provided by Service Water and is not impacted in any way by containment parameters including CAP. HPCI and RCIC utilize their suction source of water as their cooling medium to provide system oil cooling.

Although HPCI and RCIC have the capability to take suction from the Suppression Pool (also referred to as the torus), they are normally aligned to take suction from the Condensate Storage Tanks (CSTs). The CSTs contain water at a low enough temperature so as to not challenge lube oil cooling (or NPSH limits) for the HPCI and RCIC systems. The EOPs direct the operators to maintain suction from the CSTs, if they are available. HPCI has an automatic suction transfer from the CSTs to the Suppression Pool, for elevated Suppression Pool levels. This automatic transfer can be defeated by the operators, using procedural direction from the EOPs. This is accomplished by taking a simple action that is performed entirely within the control room. The operators are trained to assure suction for HPCI and RCIC is from the CSTs when the CSTs are available.

The following figure is taken from the EOP flowchart for RPV Control (C.5-1100) and provides instructions for the use of the Preferred Injection Systems, which includes the operation of the HPCI and RCIC systems.

The Suppression Pool water supply is available to the HPCI and RCIC systems in the unlikely event that the CSTs are unavailable. For this case, EOPs provide guidance to assure NPSH concerns are addressed and potential damage to the respective system from high torus temperature is considered. The PRA model allows credit for HPCI and RCIC systems taking suction from the torus, if the CSTs are not available.

The MNGP PRA model does not require any adjustments to account for any CAP-related impacts on the lube oil cooling for high pressure injection (HPI) systems (i.e., HPCI and RCIC), or due to NPSH concerns, because the MNGP PRA already fails the HPCI and RCIC systems for those scenarios where HPI suction is from the torus, and torus cooling is not successful.

P Preferred Injection Systems

⚠ **Exceeding NPSH or Vortex Limits may damage systems (Figs E-I; SPDS 73-77).**	**Capacity**
• Condensate/Feedwater	9000 gpm
• HPCI ⚠ **Exceeding 180°F suction temperature may damage system.** ☛ Use CST suction if available. ☛ OK to defeat: .... **C.5-3202** • High RPV water level trip • High temperature trip • High torus level suction transfer	3000 gpm
• RCIC ⚠ **Exceeding 180°F suction temperature may damage system.** ☛ Use CST suction if available. ☛ OK to defeat: .... **C.5-3201** • Low RPV pressure trip • High RPV water level trip • High exhaust pressure trip • High temperature trip	400 gpm
• CRD **C.5-3204**	74 gpm
• Core Spray	3150 gpm
• LPCI ☛ Use HXs as soon as you can.	4000 gpm

e. For the loss of instrument air system initiating event explain which systems the PRA credits as available for CAP-related mitigation. Include a discussion on how the Residual Heat Removal (RHR) Service Water system is credited in the PRA model for CAP-related CDF and LERF. Explain whether or not the RHR Service Water system is a risk-significant system for CAP-related risk and provide justification for your conclusion.

Response to RAI 5e

Although the plant cannot continue power operation with a complete loss of Instrument Air, post-transient accident mitigation capability is not severely affected, as components with dependencies on Instrument Air are designed to fail in a safe position or have an alternate pneumatic supply available (e.g., the Residual Heat Removal Service Water (RHRSW) system) to back up the Instrument Air supply.

Upon a loss of all Instrument Air; the Core Spray, Low Pressure Coolant Injection (LPCI), HPCI, RCIC, Fire Water, RHRSW, and CRDH systems are available as injection systems, the RHR system and the Hard Pipe Vent are available for decay heat removal, and the Safety Relief Valves (SRVs) are available for reactor pressure control to serve as the CAP-related mitigation systems.

The RHRSW system relies on pneumatic sources solely to regulate the cooling water flow through the RHR heat exchangers by throttling heat exchanger outlet valves. Pneumatic pressure is automatically retained to the heat exchanger outlet valves by dedicated, safety-related, local air compressors in the event of a loss of all instrument air.

The RHRSW system can function as a decay heat removal or can function as an alternate low-pressure injection system. As the preferred decay heat removal system when the main condenser is not available as a heat sink, the RHRSW system is risk significant for CAP-related events. As an alternate low-pressure injection system, the RHRSW system is not risk significant for CAP-related events, or for any transient events, as there are a number of other low-pressure injection systems available to provide this function.

REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC) (R. Kuntz) to NSPM (R. Loeffler), "Monticello ILRT extension amendment request for additional information (CAC No. MF7359)," dated November 18, 2016

2. NSPM (P. Gardner) to NRC, "Part 1 Response to Probabilistic Risk Assessment (PRA) Related Requests for Additional Information: License Amendment Request for a Permanent Extension of the 10 CFR 50 Appendix J Containment Type A Test Interval (CAC No. MF7359)" (L-MT-16-062), dated December 16, 2016

3. U.S. NRC Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011

4. U.S. NRC Draft Regulatory Guide DG-1285 (Proposed Revision 3 to Regulatory Guide 1.174, dated May 2011), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2012

5. Northern States Power (NSP) to NRC, "Monticello IPEEE Submittal," dated March 1, 1995

6. Electric Power Research Institute (EPRI) Topical Report 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated October 2008

7. U.S. NRC to NSP (M. Hammer), "Review of the Monticello Individual Plant Examination of External Events (IPEEE) Submittal (TAC No. M83644)," dated April 14, 2000

Application of FLEX for Defense-in-Depth

Discussion

NRC Order EA-12-049 in response to the Fukushima events required licensees to develop mitigating strategies FLEX[1] for beyond-design-basis external events (BDBEE). The FLEX strategies were developed in accordance with the guidance of Nuclear Energy Institute (NEI) 12-06 "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (Reference 1). Although the FLEX strategies were originally designed to prevent core damage following an extended loss-of-alternating current (AC)-power (ELAP) event, they are beneficial for other types of initiating events. As indicated in an NRC Office of Nuclear Regulatory Research (RES) study concerning the Monticello ILRT license amendment request (Reference 2), application of the FLEX methodology was one of several items that could potentially be applied to address a loss of containment accident pressure (CAP), potentially resulting in a loss of Net Positive Suction Head (NPSH) for the low pressure Emergency Core Cooling System (ECCS) pumps. FLEX provides independent injection capability by portable diesel generator powered pumps that are external to the primary containment, and also provides required support equipment, providing a diverse, independent, source for injection. There is no reliance on CAP for operation of any of the FLEX pumps since they draw upon sources of water that are external to the primary containment. While the risk reduction achieved from use of FLEX was not quantified, the flex mitigation strategies provide additional defense-in-depth in situations involving a potential loss of NPSH due to a loss of CAP.

In August 2016, NEI 16-06, "Crediting Mitigating Strategies in Risk-informed Decision Making" (Reference 3), was issued. On January 13, 2017, amendments were issued for the Watts Bar Nuclear Plant, Units 1 and 2, crediting use of FLEX equipment in a support of a licensing action (Reference 4). Specification 3.8.1, "AC Sources – Operating," was revised to extend the Completion Time for one inoperable diesel generator (DG) from 72 hours to 10 days based on the availability of a supplemental AC power source (FLEX DG). While the specifics of the licensing action taken for the Watts Bar units are not the same as that proposed for the MNGP (i.e., request for approval to extend the ILRT frequency from 10 to 15 years), the development of industry guidance and issuance of amendments indicates that FLEX equipment is acceptable to consider in licensing actions.

High Level Summary of FLEX

The FLEX strategies, procedures, equipment, training, and guidance are in place at the MNGP to maintain or restore core cooling, containment integrity, and Spent Fuel Pool (SFP) cooling capabilities following a BDBEE. With these strategies, the operators are capable of mitigating a simultaneous loss of all AC power and loss of normal access to the ultimate heat sink (UHS),

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1. FLEX stands for “Diverse and Flexible Coping Strategies”

and have adequate capacity to address challenges to core cooling, containment integrity, and SFP cooling capabilities in all modes of operation.

The FLEX strategies are implemented in three (3) phases. In Phase 1 the installed plant equipment, i.e., RCIC or HPCI, is used for high pressure injection. In Phase 2 the plant transitions from using the installed plant equipment to utilizing on-site portable diesel powered pumps and support equipment. In Phase 3, the equipment that is transported from one of the two National Strategic Alliance for FLEX Emergency Response (SAFER) Response Centers (NSRCs) (equipment delivery begins to arrive in 24 hours) to the plant site, can be utilized.

The overall plant response to an ELAP and loss of the UHS is accomplished through the normal plant command and control procedures and practices. The plant emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) govern the operational response. The FLEX strategies are deployed in support of the AOPs/EOPs using separate FLEX Support Guidelines (FSGs), which provide direction for using FLEX equipment in maintaining or restoring key safety functions. The FSGs provide available pre-planned FLEX strategies for accomplishing specific tasks. The FSGs are used to supplement (not replace) the existing procedure structure that establishes command and control for the event (e.g., AOP, EOP, Extreme Damage Mitigation Guidelines (EDMG), and Severe Accident Management Guidelines (SAMG)). The FLEX strategies were evaluated for integration with the existing procedures. The FLEX strategies have been implemented in such a way as to not violate the basis of existing procedures. NSPM has incorporated interfaces with the FSGs into AOPs and EOPs, as necessary to enable implementation of FLEX strategies in response to BDBEEs at MNGP.

At the MNGP one (1) portable diesel generator and one (1) portable pump are required to implement the FLEX strategies. There are two FLEX equipment storage buildings, each containing a full set of FLEX equipment to meet the N+1 requirement of NEI 12-06. The FLEX equipment storage buildings are separated by sufficient distance to provide protection from external (e.g., tornado) generated hazards. The credited mechanical/electrical tie-ins to plant systems for the portable equipment are permanent connections designed specifically for use with the portable pump and electrical equipment. The portable pump connections for the core, containment and SFP cooling functions have a primary and an alternate connection or delivery point. Phase 2 and Phase 3 deployment strategies rely upon trucks, a front end loader, and a fork lift, to (a) transport FLEX equipment and site personnel, and (b) clear the deployment routes, staging areas and paths for running hoses and cable. The FLEX support guidelines are listed below.

<u>FLEX Support Guidelines</u>

- Station Blackout Guideline
- FLEX Site Assessment
- FLEX Debris Removal

- FLEX Response During External Flooding
- FLEX Portable Diesel Pump Staging and Hose Connection
- FLEX Portable Diesel Pump Operation
- RCIC and HPCI Operation with High Level Trip Bypassed
- Spent Fuel Pool Makeup with FLEX Portable Diesel Pump
- FLEX DC Load Shed
- Stage and Connect FLEX Portable DG (480V operation of battery chargers)
- FLEX Portable Generator Operation
- Operate Essential Battery Chargers from FLEX Portable Diesel Generator
- Backfeed MCCs from FLEX Portable Diesel Generator
- Stage 120V Generator
- FLEX Phase 3 4kV Generator Installation
- Reactor Building Ventilation During FLEX Conditions (RCIC room forced ventilation)
- Control Room Ventilation During FLEX Conditions (and PAB - forced)
- EFT Ventilation During FLEX Conditions
- Lighting During FLEX Conditions
- Spent Fuel Pool (SFP) Strategies
- Repower PAB PBX Phone System with Portable Generator
- Refueling Emergency Portable Diesel Powered Equipment
- Alternate Methods for Monitoring RX Vessel and Containment Parameters

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REFERENCES

1. Nuclear Energy Institute (NEI) 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," dated August 2012

2. U.S. Nuclear Regulatory Commission (NRC), "RES Study in Support of the Review of the Monticello License Amendment Request to Extend the Type A Integrated Leak Rate Test Frequency," dated June 22, 2016

3. NEI 16-06, "Crediting Mitigating Strategies in Risk-informed Decision Making," dated August 2016

4. U.S. NRC to TVA (J. Shea), "Watts Bar Nuclear Plant, Units 1 and 2 – Issuance of Amendments Regarding Extension of Completion Time for Inoperable Diesel Generator (CAC Nos. MF7147 and MF7148)," dated January 13, 2017

Acceptability of Throttling ECCS Pumps to Ensure Adequate NPSH

Regulatory Guide (RG) 1.174 provides guidance for assessing the impact on risk of proposed changes to the licensing basis. As part of the actions to align with SECY-11-0014 (Reference 1), the regulatory guide is being revised to address the current NRC guidance for utilizing containment accident pressure (CAP) to ensure Net Positive Suction Head (NPSH). Draft Regulatory Guide DG-1285 (Reference 2) provides insight on what to consider regarding the acceptability of a reliance upon operator actions, in situations in risk informed submittals.[9] While it is recognized that draft Regulatory Guide DG-1285 is under development, an included example provides valuable insight on considerations that also pertain to the Monticello situation where the operators perform a manual action to throttle the low pressure Emergency Core Cooling System (ECCS) pumps post-accident. The draft RG example involves a power uprate that results in higher temperatures of the water available at the suction of the containment heat removal pumps. The DG-1285 example states:

> The licensee proposes to credit containment accident pressure in its analysis of the net positive suction head (NPSH) available to these pumps. However, it is determined that this pressure may not be available if nonsafety-related containment fan coolers continue to run after an accident. The licensee proposes to write a new procedure to direct operators to secure these containment fan coolers in the event of a reactor accident instead of designing a trip circuit to perform this function automatically. This procedure could be considered overreliance on a programmatic activity.

Section 2.1.1.3 of the draft RG, entitled, “Factors To Consider When Evaluating the Impact of a Change on Defense-in-Depth,” discusses avoiding an overreliance on programmatic activities as compensatory measures for changes to the licensing basis. As indicated above, the draft regulatory guide DG-1285 states:

> However, if it can be demonstrated, for example, that this additional operator action is reliable and feasible, does not overburden the operators or adversely affect their ability to respond to an accident, does not otherwise affect plant safety, the licensee may be able to justify such a change.

While the specifics of the draft RG example are not the same as those for Monticello, performance of an operator action (e.g., throttling the low pressure ECCS pumps) has been determined to be acceptable in a risk informed application, provided the applicable criteria are satisfied. NSPM has demonstrated to the NRC’s satisfaction, as indicated by the approval of the Extended Power Uprate (EPU) license amendment for Monticello – Amendment No. 176 (Reference 3), that the operator action to throttle the low pressure ECCS pumps is reliable and feasible, does not overburden the operators, nor adversely affect their ability to respond to an accident. Each of the considerations in SECY-11-0014 was addressed as reflected in the

9. Draft Regulatory Guide DG-1285 is the proposed Revision 3 to Regulatory Guide 1.174.

NRC safety evaluation for the EPU license amendment (including the operator action to throttle the low pressure ECCS pumps). The SECY specifically takes into account the considerations associated with utilization of CAP to provide NPSH, and the issues to address to appropriately mitigate the associated risk. Therefore, while the revision to RG 1.174 is not yet complete, the considerations SECY-11-0014, which in-part, serves as the basis for the proposed revisions, were reflected in the approval of the EPU licensee amendment, and are applicable here. The risk reduction achieved from crediting the acceptability of throttling the low pressure ECCS pumps was not quantified. However, it is a reliable operator action that was agreed to by the NRC under the EPU license amendment, and hence could be, and should be, able to be credited to some degree to provide additional defense-in-depth in situations involving a potential loss of NPSH due to a loss of CAP.

REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC), SECY-11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," dated January 31, 2011

2. U.S. NRC Draft RG DG-1285 (Proposed Revision 3 to Regulatory Guide 1.174, dated May 2011), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

3. U.S. NRC to NSPM (K. Fili), "Monticello Nuclear Generating Plant – Issuance of Amendment No, 176 to Renewed Facility Operating License Regarding Extended Power Uprate (TAC No. MD9990)," dated December 9, 2013