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L-MT-16-044
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

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)

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)

2) NRC (R. Kuntz) to NSPM (R. Loeffler), “DRAFT Request for Additional Information RE: Monticello license amendment request for ILRT extension (CAC MF7359),” dated August 30, 2016

On February 10, 2016, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, submitted a license amendment request 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 August 30, 2016, the U.S. Nuclear Regulatory Commission (NRC) requested additional information pertaining to the primary containment performance and history and a clarification of ILRT test results (Reference 2). The responses to these requests for additional information are provided in the Enclosure.

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 October 10, 2016.

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

Enclosure

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

ENCLOSURE

MONTICELLO NUCLEAR GENERATING PLANT

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

(10 pages follow)

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On February 10, 2016, NSPM submitted a license amendment request proposing a change 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 August 30, 2016, the NRC requested the following additional information. The responses to this request for additional information (RAI) are provided below.

<u>RAI-1</u>

In the LAR, Enclosure 1, Section 4.7.1, Table “NEI 94-01, Revision 2-A, Limitations and Conditions,” the fourth “Limitation/Condition” reads “The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to the NRC staff’s safety evaluation for NEI 94-01, Revision 2, dated June 25, 2008 (SE) Section 3.1.4).”

The “MNGP Response” reads:

> There are no major modifications planned for the MNGP that would affect the containment structure.

SE Section 3.1.4 (Reference 2) reads:

> Section 9.2.4 of NEI TR 94-01, Revision 2, states that: “Repairs and modifications that affect the containment leakage integrity require LLRT or short duration structural tests as appropriate to provide assurance of containment integrity following the modification or repair. This testing shall be performed prior to returning the containment to operation.” Article IWE-5000 of the ASME Code, Section XI, Subsection IWE (up to the 2001 Edition and the 2003 Addenda), would require a Type A test after major repair or modifications to the containment. In general, the NRC staff considers the cutting of a large hole in the containment for replacement of steam generators or reactor vessel heads, replacement of large penetrations, as major repair or modifications to the containment structure. At the request of a number of licensees, the NRC staff has agreed to a relief request from the IWE requirements for performing the Type A test and has accepted a combination of actions consisting of ensuring that: (1) the modified containment meets the pre-service non-destructive evaluation (NDE) test requirements (i.e., as required by the construction code), (2) the locally welded areas are examined

> for essentially zero leakage using a soap bubble, or an equivalent, test, and (3) the entire containment is subjected to the peak calculated containment design basis accident pressure for a minimum of 10 minutes (steel containment) and 1 hour (concrete containment), and (4) the outside surfaces of concrete containments are visually examined as required by the ASME Code, Section XI, Subsection IWL, during the peak pressure, and that the outside and inside surfaces of the steel surfaces are examined as required by the ASME Code, Section XI, Subsection IWE, immediately after the test. This is defined as a short duration structural test of the containment. For minor modifications (e.g., replacement or addition of a small penetration), or modification of attachments to the pressure retaining boundary (i.e., repair/replacement of steel containment stiffeners), leakage integrity of the affected pressure retaining areas should be verified by a LLRT.

The LAR Section 4.1.4, "Non-Risk Based Assessment" (Page 31 of 64) reads:

> <u>Pressure Testing Requirements</u>
>
> If repair/replacement activities of ASME Section XI, Subsection IWA-4000 become necessary on Class MC components, as authorized by the Fifth Interval ISI Plan Relief Request RR-007, post repair/replacement pressure test requirements for components and parts of the pressure retaining boundary shall comply with the requirements of the 2007 Edition including the 2008 Addenda of ASME Section XI, Subsection IWE-5000, as well as all applicable conditions in 10 CFR 50.55a for post-repair/replacement pressure testing of Class MC components.
>
> Personnel performing post repair/replacement pressure testing required by IWE-5000, Appendix J leak rate tests, are qualified in accordance with the MNGP Primary Containment Leakage Rate Testing Program.

The NRC staff notes that the above LAR excerpts are "forward looking" with respect to plans for any future containment modification. In contrast, the MNGP containment has been in service for approximately 45 years. The NRC staff requests that additional historical information be provided (i.e., a synopsis) about any modifications to the MNGP containment and about the subsequent post modification testing. The synopsis should demonstrate consistency with guidance of SE Section 3.1.4.

<u>Response to RAI-1</u>

A synopsis of the significant modifications made prior to the 2007 ILRT are provided below:

- Modifications were performed to the pressure suppression system to address generic Mark I Containment Load Program conclusions.
- Modifications were performed to permit proper Type C testing of valves where the testing did not conform to Appendix J.
- A new penetration was added for installation of the Hard Pipe Vent System in response to Generic Letter 89-16, "Installation of a Hardened Wetwell Vent."
- Upgrades to the Drywell Personnel Airlock equalizing valves and electrical penetration.
- The Reactor Pressure Vessel Head Spray lines were cut and capped as this function was removed.
- The Combustible Gas Control System lines communicating with the primary containment were cut and capped as part of the removal of the system from service with the adoption of 10 CFR 50.44 rule changes allowing removal of the hydrogen recombiners.
- The primary containment double ply bellows assembly for penetration X-16B, "A Core Spray," was replaced with a modified bellows design in 1998.

These modifications were tested in accordance with 10 CFR 50 Appendix J and ASME XI Subsection IWE, as applicable, in accordance with the requirements of the Design Modification Process.

Since the 2007 ILRT the only modification that potentially affected the leakage integrity of containment was installation of an EPA in spare nozzle penetration X-101A for instrumentation cables to monitor the Steam Dryer performance for the Extended Power Uprate. Provisions were made in the design of the modification to ensure that the EPA and associated welds could be locally leak tested. The penetration was leak tested in accordance with 10 CFR 50 Appendix J, examined in accordance with ASME B&PVC Section III, Subsection NE, and visually examined in accordance with ASME B&PVC XI Subsection IWE.

<u>RAI-2</u>

Section 9.2.3 of Nuclear Energy Institute 94-01, Revision 0 (Reference 3) reads in part:

> For purposes of determining an extended test interval, the performance leakage rate is determined by summing the UCL [upper confidence limit] (determined by containment leakage rate testing methodology described in American National Standards Institute and American Nuclear Society (ANS) 56.8-1994) with As-left MNPLR [minimum pathway leakage rate] leakage rates for penetrations in service, isolated or not lined up in their accident position (i.e., drained and vented to containment atmosphere) prior to a Type A test. In addition, any leakage pathways that were isolated during performance

> of the test because of excessive leakage must be factored into the performance determination. If the leakage can be determined by a local leakage rate test, the As-found MNPLR for that leakage path must also be added to the Type A UCL. If the leakage cannot be determined by local leakage rate testing, the performance criteria for the Type A test are not met.

LAR Table, "MNGP Type A ILRT Results" in Enclosure 1 displays the Type A test results since December 1984 (Reference 1 – page 17 of 64). For the Type A test of April 2007, the staff requests that a summary breakdown of the test specific data for the MNGP containment "As-Found" Leakage Rate of 0.7323 percent primary containment air weight per day (wt/day).

Consistent, with the above excerpt from Section 9.2.3, the detailed breakdown of data should include the cumulative "as left" MNPLR leakage rate penetration and Containment Isolation Valve (CIV) test values used to derive the "As-Found" Leakage Rate of 0.7323 % wt./day. Also provide, the minimum containment pressure (P_a) value (in psig) recorded for the duration of the ILRT.

Response to RAI-2

The measured leakage rate for the 2007 ILRT at the 95% UCL, was 0.5135 wt.%/day. The "As-Left" and "As-Found" ILRT Leakage values reported in the LAR for the 2007 ILRT have been revised to account for some discrepancies as discussed below. These discrepancies resulted in non-significant changes to the values determined for the Lineup Penalty Leakage (Table 1) and LLRT Savings (Table 3).

The "As-Left" ILRT leakage rate is calculated from the measured leak rate plus corrections for:

1. Tank and vessel level changes during the test;
2. Valves and penetrations not lined up (i.e., not drained/vented) for the test; (Table 1)
3. For components isolated during the test. (Table 2)

The level changes inside containment during the test can change the measured leakage rate. The level changes were determined to be negative and as such a zero correction was used, 0.00 wt.%/day.

The ILRT lineup leaves some penetrations isolated, but not vented or drained, as long as the As-Left MNPLR local leakage from them is applied to correct the ILRT results (i.e., take a Type B or Type C test "penalty"). Table 1 provides the lineup penalty leakage values associated with the 2007 ILRT.

Table 1 – Lineup Penalty Leakage Values

Penetration	Penetration Description	Leakage (in SCFH)
X-9A	RPV Feedwater	0.706
X-9B	RPV Feedwater	0.370
X-10	RCIC Steam Supply	1.410
X-11	HPCI Steam Supply	1.415
X-12	RHR Supply	0.922
X-13A	B RHR Return	0.047
X-13B	A RHR Return	0.951
X-14	Cleanup Supply	0.482
X-16A	B Core Spray	0.085
X-16B	A Core Spray	0.0308
X-18	Floor Drain Sump Discharge	0.155
X-19	Equipment Drain Sump Discharge	0.102
X-20	Demin Water Supply	0.050
X-21	Service Air (Breathing Air)	0.395
X-22	Instrument Air	0.116
X-23	Cooling Water Supply	4.650
X-24	Cooling Water Return	0.725
X-27D	H2/O2 Analyzer	0.089
X-27E	Oxygen Analyzer	0.456
X-27F	H2/O2 Analyzer	0.091
X-34A	Alternate N2 Supply (B Train)	0.032
X-35A	Traversing In-Core Probe	0.001
X-35B	Traversing In-Core Probe	0.094
X-35C	Traversing In-Core Probe	0.004
X-35E	Traversing In-Core Probe	2.312
X-37B	CRD Insert and Recirc Pump 11 Seal Injection	0.227
X-38B	CRD Withdraw and Recirc Pump 12 Seal Injection	1.088
X-39A	Containment Cooling (RHR)	6.541
X-39B	Containment Cooling (RHR)	0.066
X-41	Recirc Loop Sample	0.009
X-42	Standby Liquid Control	0.060
X-105BG	Alternate N2 Supply (A Train)	0.035
X-211A	RHR to Spray Header	0.004
X-211B	RHR to Spray Header	0.004
X-212	RCIC Turbine Exhaust	0.418
X-214	H2/O2 Analyzer Return	1.36(1)
X-215	H2/O2 Analyzer	1.735(1)
X-217	HPCI Exhaust Vacuum Breaker	0.004
X-219	RCIC Exhaust Vacuum Breaker	3.151
X-220	H2/O2 Analyzer	2.286(1)
X-221	HPCI Turbine Exhaust	0.242

Penetration	Penetration Description	Leakage (in SCFH)
X-229B	Vacuum Breaker Air	0.211
X-240	Hard Pipe Vent	0.253[(1)]
	Total SCFH:	33.385
	Converted Total wt.%/day	0.0895

(1) Transcription errors were identified between the LLRT procedures and ILRT procedure. Corrected values are shown above. This condition has been entered into the Corrective Action Program.

During the conduct of the ILRT two penetrations were isolated. Table 2 shows the additional leakage from the As-Left MNPLR associated with these pathways. No issues were identified with the Isolation Penalty Leakage Values.

Table 2 – Isolation Penalty Leakage Values

Penetration	Penetration Description	Leakage (in SCFH)
X-25 / X-26	DW Vent Penetration / DW Purge Penetration (applicable portions)	1.890
X-209A	Torus Pressure Narrow Range	15.759
	Total SCFH:	17.649
	Converted Total wt.%/day	0.0473

The 2007 ILRT "As-Left" and revised "As-Left" Leakage is:

	Measured Leak Rate 95 % UCL		Lineup Penalty Leakage		Isolation Penalty Leakage		As-Left ILRT Leakage
"As-Left" Leakage in 2007 ILRT:	0.5135	+	0.0876	+	0.0473	=	0.6484 wt.%/day
"As-Left" Leakage 2007 ILRT Revised:	0.5135	+	0.0895	+	0.0473	=	0.6503 wt.%/day

The above result indicates that there is significant margin to the As-Left acceptance criteria of less 0.75 L_a (0.9 wt.%/day) in accordance with Specification 5.5.11.

The "As-Found" ILRT value is calculated from the "As-Left" ILRT value and includes positive differences between the "As-Found" and "As-Left" LLRT leakage rates for each pathway tested and adjusted prior to the performance of the Type A test (leakage savings). Table 3, LLRT Savings shows the penetrations and associated LLRT leakage savings.

Table 3 – LLRT Savings

Penetration	Penetration Description	MNPLR Savings SCFH
X-200B	Torus Manway Northeast	0.001
X-200A	Torus Manway Southwest	0.002
X-16A	B Core Spray	0.000(1)
X-26 / X-218	DW Ventilation Supply / Nitrogen Purge	6.539(1)
X-217	HPCI Exhaust Vacuum Breaker	4.916
X-21	Service Air Supply	21.139(1)
X-38B	12 Recirc Seal Injection	0.273
X-35B	Traversing In-Core Probe	4.370
	Total SCFH:	37.24
	Converted Total wt.%/day	0.0998

(1) Calculational or transcription errors were identified between the LLRT procedures and ILRT procedure. Corrected values are shown above. This condition has been entered into the Corrective Action Program.

The 2007 ILRT "As-Found" and revised "As-Found" Leakage is:

	As-Left ILRT Leakage		Leakage Savings		As-Found ILRT Leakage
"As-Found" Leakage in 2007 ILRT:	0.6484	+	0.0839	=	0.7323 wt.%/day
"As-Found" Leakage 2007 ILRT Revised:	0.6503	+	0.0998	=	0.7501 wt.%/day

The difference between the "As-Found" and "As-Left" ILRT Leakages reported in the LAR and those discussed here are insignificantly small and do not change the conclusion that the 2007 ILRT was valid and determined a leakage which is acceptable by the MNGP plant Technical Specifications.

The minimum containment pressure (P_a) during the 2007 ILRT was 56.557 psia. The pre-pressurization average atmospheric pressure for both monitoring channels was 14.112 psia. The minimum containment pressure in psig for the 2007 ILRT was determined to be 56.557 psia – 14.112 psia = 42.445 psig.(1) This value satisfies the requirement of ANSI/ANS-56.8-1994, paragraph 3.2.11, "Type A Test Pressure," which states:

> The Type A test pressure shall not be less than 0.96 P_{ac} nor exceed P_d. … The test pressure shall be established relative to the external pressure of the primary containment measured at the start of the Type A test.

1. While not required at the date of the 2007 ILRT, this test pressure also met the 0.96 P_{ac} criterion for the later increased P_{ac} of 44.1 psig for EPU.

The peak accident pressure (P_{ac}) in 2007 for the MNGP primary containment was 42 psig and the design pressure (P_d) value was 56 psig.

RAI-3

The concluding sentence of the first paragraph of LAR Enclosure 1, Section 4.4.3 reads:

> In accordance with Specification 5.5.11, the allowable maximum pathway total Type B and Type C leakage is 0.6 L_a [allowable leakage rate] (or 60 percent of L_a) approximately 285 scfh, where L_a equals 475.1 scfh, excluding the Main Steam Pathway, Specification 5.5.11.a.2.

The staff reviewed the local leak rate summaries from the last five refueling outages (i.e., RFO23 through RFO27) contained in LAR Enclosure 1 (i.e., Section 4.4.3) Table "MNGP Type B and C Local Leak-Rate Test (LLRT) Combined As-Found / As-Left Trend Summary" (Page 39 of 64).

The staff requests the following clarification of two issues pertaining to Table "MNGP Type B and C LLRT Combined As-Found / As-Left Trend Summary":

1) The LLRT data contained in the four columns of the Table, for RFO23 through RFO26, consistently yields a L_a approximately equal to 458.6 scfh for all Minimum and Maximum Pathway Leakage values. For example, for RFO23 the "As-Found" Minimum Pathway Leakage (scfh) yields 75.27 ÷ 0.1641 = 458.7 scfh.

   In contrast, the LLRT data contained in the last column of the Table, RFO27, consistently yields a L_a approximately equal to 475.1 scfh for all Minimum and Maximum Pathway Leakage values.

   What caused the value of L_a to increase from 458 scfh to 475 scfh following RFO26 in 2013?

2) License Amendment No. 176 for MNGP extended power uprate (EPU) was approved by the NRC staff on December 9, 2013. (Reference 4). As a result of the EPU, the P_a value in TS 5.5.11 increased from 42 psig to 44.1 psig. Currently, what percentage of the components (i.e., containment penetrations and CIVs) have been tested at the EPU elevated P_a?

Response to RAI-3

The spring 2013 Refueling Outage (RFO) 26 was the last refueling outage completed before receipt in December 2013 of the EPU license amendment (Amendment 176), which increased P_a from 42 psig to 44.1 psig. This increase in pressure results in an increase in allowable leakage that corresponds to L_a. In 2015, the MNGP completed RFO 27, the first post-EPU refueling outage, and the LLRT tests were performed with the higher P_a (44.1 psig.)

During RFO 27 in 2015, approximately 81 % of the Type B components and approximately 79 % of the Type C components were tested to the higher post-EPU P_a[(2)] The remaining components are scheduled for local leak rate testing in RFO 28 in 2017.

RAI-4

MNGP Updated Safety Analysis Report, Appendix K, Section K2.1.31 indicates that the MNGP structures monitoring program is implemented under the MNGP maintenance rule program.

Provide MNGP operating experience, including inspection intervals, relative to the inspection of concrete components, and any corrective action taken to disposition the findings. The response should include the inspection results for the accessible areas of reactor building foundation mat/floor slab, Drywell floor slab, shield walls, and the reactor vessel pedestal. Also, discuss whether existence of or potential for degraded conditions in inaccessible concrete areas were identified and evaluated based on conditions found in accessible areas.

Response to RAI-4

Periodic structural inspections are performed to support implementation of the requirements of the Maintenance Rule, 10 CFR 50.65, and the Renewed License Aging Management Program (AMP). The MNGP Structural Monitoring Program requires visual inspection of plant structural features within the scope of the Maintenance Rule on a 5 year interval. Inspection of the reactor building floor slab, drywell floor slab, shield walls, and reactor vessel pedestal is included in the surveillance. This structural inspection is intended to assess the overall condition of the structures and identify gross defects. Relevant plant and industry Operating Experience, as well as previous inspections of each structure are reviewed to help assess the present condition. Detailed evaluation of structural components is not necessary unless structural degradation or damage is found that justifies performing additional detailed inspections.

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2. In accordance with Operating License Condition 14, added for EPU, leak rate tests performed under the pre-EPU test conditions are not required to be performed at EPU conditions until their next scheduled performance.

Based on the inspection reports of the reactor building floor slab, drywell floor slab, shield walls, and reactor vessel pedestal since 1998, the concrete has experienced normal hairline cracking, and certain cracks have been noted to display evidence of groundwater seepage. These cracks have been evaluated for any structural integrity or degradation issues, and no such issues have been identified. The cracks have been repaired, planned to be repaired, or are being monitored for changes. The site groundwater chemistry is monitored for chemical levels which could be detrimental to the concrete. Because there has been no structural concrete degradation identified in the accessible areas of the plant, no evaluation for possible similar degradation has been performed for inaccessible areas.[3]

3. In response to Information Notice 2011-20, “Concrete Degradation by Alkali- Silica Reaction (ASR),” a petrographic examination of a concrete core sample was performed. The sample chosen was from original construction floor slab in the Turbine Building Condenser Room because it has periodically infiltrated by ground water. No evidence of ASR or ASR microcracking were observed.