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March 29, 2017

L-MT-17-022
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 Second Round PRA Related RAIs for the 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)

6) NSPM (P. Gardner) to NRC, "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)" (L-MT-17-002), dated January 31, 2017

7) NSPM (P. Gardner) to NRC, "Part 3 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-17-007), dated February 7, 2017

8) NRC (R. Kuntz) to NSPM (R. Loeffler), "Request for Additional Information RE: Monticello Nuclear Generating Plant Integrated Leak Rate Test Interval Extension Request (CAC NO. MF7359)," dated March 27, 2017

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 (RAIs) 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 pertaining to probabilistic risk assessment (PRA) 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 were provided by letter on January 31, 2017 (Reference 6). The responses to the remaining PRA related RAIs were provided by letter on February 7, 2017 (Reference 7).

On March 27, 2017, the NRC requested further clarification of the PRA analyses and provided a second round of PRA related RAIs (Reference 8). The response to this request for additional information is provided herein.

Summary of Commitments

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

I declare under penalty of perjury, that the foregoing is true and correct.
Executed on March 29, 2017.

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

Enclosure: Response to Second Round PRA Related RAIs for the LAR for a Permanent
              Extension of the 10 CFR 50 Appendix J Containment Type A Test Interval

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

MONTICELLO NUCLEAR GENERATING PLANT

RESPONSE TO SECOND ROUND PRA RELATED RAIS FOR THE LICENSE AMENDMENT REQUEST FOR A PERMANENT EXTENSION OF THE 10 CFR 50 APPENDIX J CONTAINMENT TYPE A TEST INTERVAL

(10 pages follow)

**RESPONSE TO SECOND ROUND PRA RELATED RAIS FOR THE
LICENSE AMENDMENT REQUEST FOR A PERMANENT EXTENSION
OF THE 10 CFR 50 APPENDIX J CONTAINMENT TYPE A TEST INTERVAL**

RAI 1

As provided in the response to request for additional information (RAI) 6.a, the change in population dose of 1.69 person-rem/year or 2% of the total population dose is larger than the acceptance criteria from Section 3.2.4.6 of the Safety Evaluation for the Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2, of 1.0 person-rem/year or 1% of the total population dose, whichever is less restrictive. The table provided in response to RAI 6.a indicates that EPRI Class 7 and 7a (severe accident phenomena-induced containment failures) are significant contributors to the change in population dose results.

In addition, according to the response to RAI 6.a [in Reference 7 in the cover letter], the change in the conditional containment failure probability (CCFP) of 1.66% is above the acceptance criteria of 1.5% from Section 3.2.4.6 of the Safety Evaluation for EPRI TR-1009325, Revision 2.

Given that the reported results are above the criteria for small change, the NRC staff requests that additional quantitative evaluation be performed to determine if the acceptance criteria can be met. Provide additional evaluation of these criteria, a description of the technical evaluation with justification for the approach, and the updated results.

RAI 1 Response

To reduce calculated values for Δdose and ΔCCFP, the Internal Events (IE) and Fire probabilistic risk assessment (PRA) containment accident pressure (CAP) fault tree models were refined to increase the realism of the calculated additional risk from CAP. The following response describes these refinements. For an explanation of the assignment of risk values to Class 7 and 7a, see the response to RAI 3i.

Internal Events (IE) PRA Model Changes

Modular Accident Analysis Program (MAAP) cases were run to demonstrate the impact of a large loss of coolant accident (LOCA) with a loss of CAP on the net positive suction head (NPSH) available to the low pressure emergency core cooling systems (ECCS). These MAAP cases demonstrated greater than three hours are available to establish containment cooling. This prevents the loss of low pressure ECCS pumps from inadequate NPSH for the most limiting Large LOCA case. The conservative assumption that low pressure ECCS fails for all Large LOCAs when CAP is not available, made previously in the response to RAI 4b (in Reference 7 in the cover letter), was modified to reflect actual plant conditions; this was done by retaining and crediting adequate NPSH by establishing containment

cooling so that a Large LOCA would not fail low pressure ECCS outright due to lack of NPSH. Re-quantifying the IE PRA CAP model with this more realistic assumption resulted in improvement to the change in core damage frequency (ΔCDF), change in large early release frequency (ΔLERF), ΔCCFP, and Δdose metrics.

Fire PRA Model Changes

It was determined the change in Fire CDF due to a loss of CAP was artificially high due to unrealistic conservatisms in the Fire PRA model leading to unrealistic increases in dose and CCFP. These conservatisms relate to the time available to establish torus cooling and to establish alternate injection sources that do not depend on the torus, given an initial injection success and using operator actions. In certain fire-induced accident sequences, the Fire PRA model did not include credit for specific operator actions relating to torus cooling and establishing alternate long-term injection when initial injection succeeds. The results indicate that adequate time is available to perform the actions required to establish torus cooling or establish alternate injection with pumps that do not depend on the torus for suction water.

Thus, the Fire PRA model was adjusted to apply the existing operator actions to scenarios where credit could be taken for establishing torus cooling and alternate injection. The impact on overall Fire CDF and LERF was not significant, but the impact to the importance of CAP credit was reduced because the reliability of establishing torus cooling and establishing alternate injection increased, which produced more realistic estimates of the LAR risk metrics ΔCDF, ΔLERF, ΔCCFP, and Δdose.

Refined IE and Fire Risk Metrics

The overall ΔCDF due to loss of CAP attributed to the integrated leak rate test (ILRT) surveillance frequency change was refined. The resulting decrease in total ΔCDF reduces the impact to both population dose and CCFP by decreasing the total change in frequency for the application. The table on the following page shows the refined IE and Fire PRA results for the risk metrics.

ILRT Extension Summary

Class	Dose (person-rem)	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year
1	7.29E+03	1.25E-05	9.14E-02	1.15E-05	8.42E-02	1.08E-05	7.90E-02
2	2.34E+06	1.77E-06	4.13E+00	1.77E-06	4.13E+00	1.77E-06	4.13E+00
3a	7.29E+04	3.31E-07	2.41E-02	1.12E-06	8.17E-02	1.69E-06	1.23E-01
3b	7.29E+05	8.23E-08	6.00E-02	2.79E-07	2.03E-01	4.20E-07	3.06E-01
4-6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	1.88E+06	4.22E-05	7.93E+01	4.23E-05	7.95E+01	4.23E-05	7.96E+01
7a	3.87E+05	3.47E-07	1.34E-01	5.10E-07	1.97E-01	6.28E-07	2.43E-01
8	1.53E+06	8.33E-07	1.28E+00	8.33E-07	1.28E+00	8.33E-07	1.28E+00
Total		5.81E-05	8.50E+01	5.83E-05	8.55E+01	5.85E-05	8.58E+01

ILRT Dose Rate

ΔTotal Dose Rate	From 3 Years	N/A	0.443	0.763
	From 10 Years	N/A	N/A	0.320
%ΔDose Rate	From 3 Years	N/A	0.521%	0.898%
	From 10 Years	N/A	N/A	0.375%

LERF

Total LERF		7.00E-06	7.20E-06	7.34E-06
ΔLERF	From 3 Years	N/A	1.94E-07	3.34E-07
	From 10 Years	N/A	N/A	1.39E-07

CDF

ΔCDF	From 3 Years	N/A	2.62E-07	4.52E-07
	From 10 Years	N/A	N/A	1.90E-07

CCFP %

ΔCCFP%	From 3 Years	N/A	0.437%	0.749%
	From 10 Years	N/A	N/A	0.312%

As shown in this table, the change in population dose rate is 0.763 person-rem/yr, and the percent change in dose rate is 0.898%. As a result of additional quantitative evaluation performed, the criteria of population dose being less than 1.0 person-rem/year or 1% of the total population dose, whichever is less restrictive, is met. As shown in the table, the

ΔCCFP is 0.749%. As a result of the additional quantitative evaluation performed, the criteria of ΔCCFP being less than 1.5% is met.

With the refinements made as described in this RAI response, the ΔCDF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated as 4.52E-7/year. As such, the estimated ΔCDF is determined to be "very small" using the acceptance guidelines of Regulatory Guide (RG) 1.174 (Reference C.5).

When totaling the seismic analysis from RAI 1a (in Reference 7 in the cover letter) (assuming guaranteed crack growth), and the IE and fire CAP analysis from this RAI response, the most conservative estimate (the 3 in 10 to 1 in 15 case) of ΔLERF comes from adding the seismic ΔLERF with the combined IE and Fire ΔLERF:

ΔLERF = seismic ΔLERF + combined IE and Fire ΔLERF = 4.42E-7 + 3.34E-7 = 7.76E-7

Total LERF comes from adding the seismic baseline LERF, seismic ΔLERF (assuming guaranteed crack growth), combined IE and Fire baseline LERF, and combined IE and Fire ΔLERF:

Total LERF = seismic baseline LERF + seismic ΔLERF + combined IE and Fire baseline LERF + combined IE and Fire ΔLERF

= 1.25E-6 + 4.42E-7 + 7.00E-6 + 3.34E-7 = 9.03E-6

Based on the results of this evaluation, it is reasonable to conclude that the change in CDF and LERF are "small" per the RG 1.174 risk acceptance guidelines (Reference C.5).

RAI 2

The response to RAI 4.e [in Reference 7 in the cover letter] indicates that change in large early release frequency (ΔLERF) from containment accident pressure (CAP) is estimated using the probabilistic risk assessment (PRA) models for internal and fire events. This is a change in methodology from that presented in the license amendment request (LAR) in which ΔLERF was assumed to be equal to the change in core damage frequency (ΔCDF) for CAP.

The RAI responses did not provide sufficient information on the evaluation of ΔLERF due to CAP. Comparing the reported ΔLERF of 4.65E-7/yr (which includes non-CAP related contributors) to the CAP-related ΔCDF of 2.47E-6/yr provided in the response to RAI 6.a [in Reference 7 in the cover letter] indicates that over 81% of CAP related change in CDF is classified as non-LERF. Given that the methodology for this application assumes a large pathway to the environment exists through a non-detected large pre-existing leak in the containment:

i.   Explain how the PRA model distinguishes between LERF and non-LERF for loss of CAP;

Response to RAI 2i

The PRA model distinguishes between LERF and non-LERF for loss of CAP via the success or failure of release mitigation. Given core damage due to loss of CAP attributed to a large pre-existing leak (i.e., the Class 3b failure) and failure of torus cooling (which precludes the need for CAP to ensure adequate ECCS NPSH), the PRA modeled that a large-early release would occur unless release mitigation succeeds. The LERF PRA model includes credit for release mitigation via drywell spray; successful release mitigation via drywell spray would scrub the fission product release and reduce the dose such that it is no longer large in magnitude. Failure of release mitigation via drywell spray would result in a large-early release. The LERF PRA model includes logic to model the loss of systems and components that can provide drywell spray via sources other than the low pressure ECCS pumps that depend on the torus as a suction source. This includes the Residual Heat Removal Service Water (RHRSW) System, which takes suction from the ultimate heat sink and does not depend on CAP to ensure adequate NPSH.

ii.   Explain and justify why there is a significant ΔLERF reduction from the ΔCDF;

Response to RAI 2ii

The results in the response to RAI 6a (in Reference 7 in the cover letter) indicate a significant portion of CAP risk was classified as non-LERF. As detailed in the response to RAI 1 in this submittal, the IE and Fire PRA CAP fault tree models were refined to increase the realism of the calculated additional risk from CAP. This resulted in a significantly lower ΔCDF and slightly lower ΔLERF; therefore, the difference between ΔLERF and ΔCDF is smaller. As stated in the response to RAI 1, the ΔCDF due to the ILRT frequency change was reduced by applying credit for torus cooling and injection using sources that do not depend on the torus as a suction source; these model refinements were made to portions of the model where credit was conservatively not applied in the model that supported the results in RAI 6a (in Reference 7 in the cover letter).

There is a ΔLERF reduction from the ΔCDF because the LERF PRA model includes credit for release mitigation via drywell spray, which scrubs the fission products using pumps that do not depend on the torus as a suction source; this includes RHRSW System, which takes suction from the ultimate heat sink, as described in the response to RAI 2i. The ΔLERF reduction is justified based on the MNGP plant-specific LERF analysis of the release magnitude given an un-isolated containment with a large pathway to the environment but successful drywell spray operation.

iii.   Discuss the mechanisms and considerations credited for reducing LERF to non-LERF;

Response to RAI 2iii

As described in the responses to RAIs 2i and 2ii, the LERF PRA model includes credit for release mitigation via drywell spray, with flow provided by the RHRSW systems, which takes suction from the ultimate heat sink. Drywell spray scrubs the fission products, which reduces the source term. As stated in the LERF analysis (Reference C.4), drywell spray is capable of scrubbing fission products from the containment atmosphere, which is especially important when containment fails, and has a significant effect on preventing the volatile and non-volatile fission products from being released to the environment. Reference C.4 documents crediting scrubbing in the LERF analysis.

iv.   Provide technical justification for those credited for ΔLERF reduction and their assigned likelihoods in the PRA model.

Response to RAI 2iv

The technical justification for those methods credited for ΔLERF reduction and their assigned likelihoods in the PRA model is based on the MNGP plant-specific LERF PRA analysis. Reference C.4 credits scrubbing in the LERF analysis. The assigned likelihoods for the systems that support the drywell spray release mitigation function are from the MNGP IE and Fire LERF fault trees that are part of the peer reviewed PRA models. The MNGP IE and Fire LERF PRA models represent the as-built, as-operated plant, including credit for drywell spray for release mitigation. The LERF PRA model has been peer reviewed to ASME/ANS-Ra-Sa-2009 with clarification from RG 1.200 Revision 2, as discussed in the PRA quality statement portion of the LAR (Appendix A of Reference C.6), and includes failure modes of the systems that support the drywell spray function and estimates of probabilities in accordance with the standard.

RAI 3

The NRC staff's review found further explanation is necessary regarding results reported in the responses to the RAIs. Address the following:

i.   In the LAR, Class 7 does not show a dependency on the integrated leakage rate test (ILRT) frequency; however, in the response to RAI 6.a [in Reference 7 in the cover letter], Class 7 and 7a have a dependency on the ILRT frequency which results in significant contribution to the ILRT risk results. Explain why this dependency has been introduced.

Response to RAI 3i

Regarding the increase in Class 7 risk from extending the ILRT interval; in some cases the resulting release due to a pre-existing leak is scrubbed by drywell spray, and prevented from becoming a large early release. This portion of the change in risk was assigned to Class 7, as Class 3b is intended to reflect large and early releases due to the pre-existing leak. Class 7 was used for this portion of risk because the mitigated release is small and early, and Class 7 is used to represent releases that are not part of the other EPRI classes. Class 7 was subdivided into Class 7 and 7a, to show the small early risk associated with a mitigated release from a pre-existing leak, as further described in the response to RAI 3ii.

ii.   Class 3a represents small releases. If Class 3a and Class 7a both represent small releases with dependencies on the ILRT frequency, explain the difference between these categories in the evaluation of the change in CDF over the ILRT frequency.

Response to RAI 3ii

Class 3a and Class 7a both reflect small releases, but the difference between the two classes depends on the physical containment failure and success or failure of release mitigation. Class 3a is a small release reflective of a $10L_a$ leak in containment, given core damage, with associated dose rate, as calculated per the EPRI methodology of scaling the Peach Bottom dose to determine the dose for Classes 1, 3a, and 3b. Class 7a is a small release reflective of a $100L_a$ pre-existing leak in containment with release mitigation success following core damage. The dose for this case is determined using MNGP-specific analysis for small-early dose (Reference C.4).

As stated in the response to RAI 3i, initially all non-LERF ΔCDF was assigned to Class 7 because Class 7 was used to capture all ΔCDF that was not reflective of the other EPRI classes. In order to provide a more refined dose estimate, the Class 7 risk was redefined to not include the small-early release state, because the small-early release state is representative of a core damage event with a large pre-existing leak in containment (i.e., the Class 3b failure) with successful release mitigation. The frequency and dose associated with a Class 3b pre-existing leak with successful release mitigation (quantified as a small-early release in the Level 2 PRA model) was assigned to Class 7a. Once the release category frequencies were appropriately correlated with population doses, the change in population dose was estimated and compared against the criteria.

Although the EPRI guidance does not provide specific guidance for evaluating CAP scenarios, the EPRI guidance provides precedent for sub-dividing Class 7 based on release characterization. Section 5.2 of Revision 2-A of EPRI 1009325 (Reference C.2)

divides Class 7 into multiple sub-categories based on release timing and whether or not the release is scrubbed. Therefore, the EPRI guidance provides precedent for dividing Class 7 accident sequences into sub-categories for the purpose of appropriately correlating release category frequencies with population doses.

iii.   EPRI TR 1009325, Revision 2, guidance stipulates that external events, including fire, should be included in the risk assessment. The February 7, 2017 letter indicates that Fire PRA model changes and enhancements were made, but did not provide the new values for fire CDF and LERF. Please provide these updated values for the Fire PRA baseline risk.

Response to RAI 3iii

The base Fire CDF and LERF with CAP failure values are 5.01E-5 and 6.02E-6, respectively. These values are results from the refined Fire PRA model with CAP failure, as detailed in the response to RAI 1.

iv.   EPRI TR 1009325, Revision 2, uses the Regulatory Guide 1.174 change in LERF (ΔLERF) acceptance guidelines for the ILRT extension. The response to RAI 6.a [in Reference 7 in the cover letter] did not provide a clear presentation of the various contributors to the risk estimates to the Class 3b ΔLERF. Summarize in a table format the numerical contributions to the Class 3b ΔLERF from each hazard/contributor (e.g., internal events, seismic, fire, other external hazards, steel liner corrosion, and containment accident pressure).

Response to RAI 3iv

The table below presents contributions to Class 3b ΔLERF from each hazard/contributor.

Class 3b LERF and ΔLERF by Hazard/Contributor			
Hazard/Contributor	Class 3b: Base Case 3 in 10 Years	Class 3b: Extend to 1 in 15 Years	Class 3b ΔLERF
Internal Events	4.12E-09	2.16E-08	1.74E-08
Internal Events CAP[1]	5.42E-09	2.76E-08	2.22E-08
Fire	5.92E-08	2.96E-07	2.37E-07
Fire CAP[1]	1.36E-08	7.12E-08	5.76E-08
Seismic[2]	1.10E-07	5.52E-07	4.42E-07
Seismic CAP[3]	ε	ε	ε
Other External Events[4]	ε	ε	ε
Steel Liner Corrosion	1.10E-11	5.63E-11	4.47E-11
Total	1.92E-07	9.69E-07	7.76E-07

Note 1: These values include the refinements detailed in the RAI 1 response.

Note 2: This seismic ΔLERF from RAI 1a (in Reference 7 in the cover letter) response conservatively assumes guaranteed crack growth.

Note 3: As discussed in the response to RAI 4b (in Reference 7 in the cover letter) for seismic risk, the ΔCDF due to CAP can be considered insignificant because seismic risk is driven by key structure and correlated system failures in which containment credit for CAP has no impact. For the less significant portion of seismic risk where key structures and systems survive, torus cooling can be initiated to preclude the need for an intact containment. If torus cooling is not available with all power failed (thus failing the Residual Heat Removal and Core Spray Systems), the FLEX strategy can be used to provide reactor inventory makeup and decay heat removal.

Note 4: The ΔLERF from other external events is negligible. As stated in response to RAI 1b (in Reference 6 in the cover letter) (Reference C.3), 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.

REFERENCES

C.1   Calculation PRA-CALC-16-006, "MAAP Analysis for ILRT Extension," Revision 1.

C.2   *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, Revision 2-A of 1009325, EPRI, Palo Alto, CA. 1018243, October 2008.

C.3   "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)," January 31, 2017, ML17032A038.

C.4   Monticello Nuclear Generating Plant, Application for Renewed Operating License, Appendix E – Environmental Report, 2005.

C.5   An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, May 2011.

C.6   Calculation 54005-CALC-01, "Monticello Nuclear Generating Station: Evaluation of Risk Significance of Permanent ILRT Extension," February 2016, ML16047A273. Enclosure 2 of L-MT-16-001, "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," ML16047A272.