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February 7, 2017

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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 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)

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

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 were provided by letter on January 31, 2017 (Reference 6). The responses to the remaining RAIs are provided herein. Submittal of this letter completes the response to all the RAIs.

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 February 7, 2017.

Kent Scott for P. Gardner

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

Enclosure 1: Part 3 Response to PRA Related Requests for Additional Information
Appendix 1: Findings and Observations (F&Os)

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

<u>ENCLOSURE 1</u>

MONTICELLO NUCLEAR GENERATING PLANT

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

(26 pages follow)

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

On February 10, 2016, Northern States Power Minnesota (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 were provided by letter on January 31, 2017 (Reference 3). Submittal of this letter completes the response to the remaining RAIs.

Analysis Approach Revised

During the October 13 through 14, 2016, regulatory audit, concern was expressed with crediting throttling of the low pressure Emergency Core Cooling System (ECCS) pumps under a risk-informed application, such as this one, involving increasing the containment ILRT interval. Also, concern was expressed with the degree of applicability of BWROG Licensing Topical Report (LTR) NEDC-33347P[1] (Reference 4) for demonstrating the preservation of Net Positive Suction Head (NPSH) for the low pressure ECCS pumps. The revised containment accident pressure (CAP) analysis performed for the RAI responses does NOT credit: 1) throttling of the low pressure ECCS pumps,[2] and 2) plant conditions that may preclude the need for CAP credit, as described in BWROG LTR NEDC-33347P, hereafter referred to as the "BWROG CAP topical report." Instead, a new analysis has been performed that relies on MNGP specific thermal hydraulic analysis using the Electric Power Research Institute (EPRI) Modular Accident Analysis Program (MAAP) computer program to determine which PRA

1. Boiling Water Reactor (BWR) Owners Group (BWROG) LTR NEDC-33347P, Revision 1), "Containment Overpressure Credit for Net Positive Suction Head (NPSH)"
2. The Emergency Operating Procedures contain guidance for the throttling of the low pressure ECCS pumps, which the operators routinely train on in various scenarios.

sequences would require CAP to ensure the low pressure ECCS pumps have adequate NPSH.

The CAP analysis is discussed in detail in the responses to RAIs 4b and 6a. The responses to these two RAIs summarize the revised analysis process and results of the reanalysis when credit for throttling the low pressure ECCS pumps (and application of the BWROG CAP topical report) is removed. The purpose of the new analysis was to reduce the uncertainty of the change in risk due to the increased probability of containment leakage when the ILRT surveillance interval is increased.

During refinement of the PRA models to remove credit for throttling the low pressure ECCS pumps and application of the BWROG CAP topical report, a discrepancy was identified in the modeling of spurious Safety Relief Valve (SRV) openings in the Fire PRA model.(3). Several modeling changes and enhancements were made as a result. One involved implementation of a new method for determining target damage in the Fire PRA for which a Focused-Scope Peer Review was performed. All Findings and Observations (F&Os) stemming from this Peer Review have been resolved as not affecting the ILRT extension analysis. Appendix 1 provides the associated F&Os and their resolution.

RAI 1

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

a. **The method used in the LAR to calculate the seismic change in large early release frequency (LERF) does not consider that pre-existing flaws in Class 3a (i.e., smaller flaws not considered LERF contributors) may grow due to a seismic initiating event and may not remain a Class 3a flaw type for some seismic initiators. Potential flaw growth may be a result of the initiating event rather than as a result of the accident sequence of events.**

Consider the potential for Class 3a flaw growth, given a seismic event, for the seismic risk contribution to the integrated leak rate test (ILRT) extension. Describe your method, technical justification, and results for the application. Alternatively, perform an appropriate sensitivity study to determine the risk significance of Class 3a flaw growth potential due to seismic events and provide the sensitivity analysis results impact on the risk metrics for the application.

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3. This issue was entered into and corrected under the MNGP Corrective Action Program.

Response to RAI 1a

NSPM has elected to perform a sensitivity study to estimate the risk significance of Class 3a flaw growth due to seismic events. The sensitivity analysis was performed by assuming the Class 3a probability represents and is added to the chance of a Class 3b large leak. The sensitivity analysis utilized the "simple average" approach to estimate the seismic Core Damage Frequency (CDF) to be consistent with the approach recommended in an Nuclear Energy Institute (NEI) Letter to the NRC[(4)] (Reference 5). The seismic CDF used for this sensitivity analysis is 1.08E-05, which was derived from the Generic Issue (GI) 199 report entitled, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment" (Reference 6).

The seismic LERF is assumed to be equal to 1.24E-06, which is the seismic CDF multiplied by the ratio of internal events LERF to CDF, which is 1.08E-05*9.20E-07/8.01E-06. The pre-existing leak failure probability is given as the sum of the Class 3a and 3b failure probabilities (2/217 and 0.5/218, respectively), which is 9.2E-03+2.3E-03=1.15E-02. The sensitivity analysis is performed using the same methodology of multiplying the pre-existing leak failure probability by 10/3 and 15/3 for the 1 in 10 and 1 in 15 year cases, respectively, as described in the PRA ILRT extension analysis calculation. The results of the sensitivity analysis are shown in the below table.

Subtracting seismic LERF from CDF, the Class 3a and 3b frequency can be calculated by the following formulas:

$$Freq_{class3ab} = P_{class3ab} * (CDF - LERF) = (0.0115) * (1.08\text{E-}5 - 1.24\text{E-}6) = 1.10\text{E-}7$$

$$Freq_{class3ab10yr} = \frac{10}{3} * P_{class3ab} * (CDF - LERF) = \frac{10}{3}(0.0115) * (1.08\text{E-}5 - 1.24\text{E-}6) = 3.68\text{E-}7$$

$$Freq_{class3ab15yr} = \frac{15}{3} * P_{class3ab} * (CDF - LERF) = \frac{15}{3}(0.0115) * (1.08\text{E-}5 - 1.24\text{E-}6) = 5.52\text{E-}7$$

ILRT Extension Delta LERF with Guaranteed Crack Growth			
	Base Case 3 in 10 Years	**Extend to 1 in 10 Years**	**Extend to 1 in 15 Years**
LERF	1.10E-07	3.68E-07	5.52E-07
Δ LERF (3/10 baseline)	N/A	2.57E-07	4.42E-07

4. NEI Letter, "Seismic Risk Evaluations for Plants in the Central and Eastern United States," dated March 12, 2014.

ILRT Extension Delta LERF with Guaranteed Crack Growth			
	Base Case 3 in 10 Years	**Extend to 1 in 10 Years**	**Extend to 1 in 15 Years**
Δ LERF (1/10 baseline)	N/A	N/A	1.84E-07

The results show a change in LERF due to a seismic event in which all small pre-existing leaks would grow to large leaks. The results of this sensitivity indicate that seismic risk may be important if all small pre-existing leaks were to grow to large leaks during a seismic event. However, if only 50% of small leaks would grow to a large leak, the delta LERF would be as shown in the following table:

ILRT Extension Delta LERF with 50% Chance of Crack Growth			
	Base Case 3 in 10 Years	**Extend to 1 in 10 Years**	**Extend to 1 in 15 Years**
LERF	6.62E-08	2.21E-07	3.31E-07
Δ LERF (3/10 baseline)	N/A	1.54E-07	2.65E-07
Δ LERF (1/10 baseline)	N/A	N/A	1.11E-07

When comparing the guaranteed crack growth and 50% chance of crack growth cases, in the most limiting delta LERF case, comparing a 1 in 15 to 3 in 10 interval, delta LERF decreases by 40%. The conclusion of this sensitivity analysis is that if one conservatively assumes all small leaks grow to a large leak during a seismic event, it is conservative to say that the change in risk of the ILRT extension is “small” per the Regulatory Guide 1.174 acceptance guidelines (Reference 7).

RAI 4

According to the LAR containment over-pressure is required for Emergency Core Cooling System (ECCS) performance. Section 5.2.4 of Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2 includes guidance for plants that rely on containment over-pressure for net positive suction head (NPSH) for ECCS injection (Reference B.5). It is not clear how the analysis for containment accident pressure (CAP) provided in Section 5.3.4 of Enclosure 2 to the LAR is consistent with this guidance. The EPRI report provides the following examples of accident scenarios to be considered:

- **Loss of coolant accident (LOCA) scenarios where the initial containment pressurization helps to satisfy the NPSH requirements for early injection, and**

- **Total loss of containment heat removal scenarios where gradual containment pressurization helps to satisfy the NPSH requirements for long-term use of an injection system from a source inside containment.**

The LAR refers to "LOCA initiators," which is not necessarily the same as a LOCA scenario (when the LOCA is not the initiating event). LOCA initiators, as shown in Table 5-1 of Enclosure 2 to the LAR, are a relatively small contributor to the internal events CDF. However, other internal events initiators (e.g., transients) or conditions (e.g., station blackout) could result in a consequential LOCA. In addition, it is not clear whether the scenarios involve early or long-term injection.

a. The EPRI guidance suggests a broader set of scenarios than those provided in the LAR. In fact, the EPRI guidance is to adjust the PRA model to account for CAP if accident scenarios could be impacted by a large containment failure that eliminates the necessary CAP. Explain how the two LOCA scenarios described in Section 5.3.4 of the LAR are sufficient for the CAP sensitivity analysis given the EPRI guidance, or perform an updated analysis using the PRA model. In any case, confirm that the PRA model used for the CAP risk assessment included all initiating events or conditions which correspond to the two EPRI guidance general event categories.

Response to RAI 4a

In lieu of providing justification for the existing analysis, a revised analysis was performed. The new analysis included the initiating events or conditions corresponding to the general event categories in the EPRI guidance. The results of the new analysis are described in the response to Part b of this RAI.

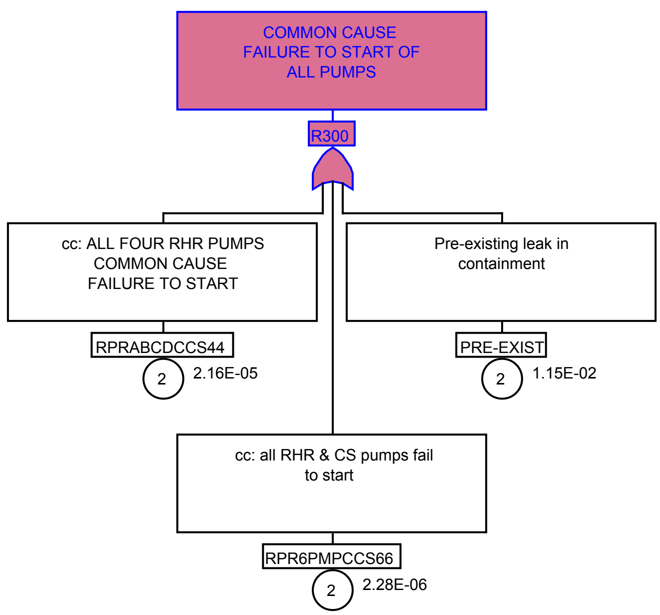
b. If an updated analysis is performed, describe this analysis and justify why it is sufficient to evaluate the risk due to the containment accident pressure for the application.

Response to RAI 4b

A new analysis has been performed that simplifies the risk estimate modeling and relies on MNGP specific thermal hydraulic analysis using the EPRI MAAP computer program. MAAP was used to determine which PRA sequences would require CAP to ensure the low pressure ECCS pumps have adequate NPSH (Reference 8). Performing MNGP specific thermal hydraulic analysis is a best-estimate approach for establishing the sequences in which CAP would be required to ensure the low pressure ECCS pumps have adequate NPSH. An iterative approach was taken to determine which PRA sequences would be significant, in terms of the risk acceptance guidelines in Regulatory Guide 1.174, if CAP is always required for adequate NPSH with a containment failure detectable only by the ILRT.

The first step in the analysis is to estimate the probability that a leak of sufficient size to defeat CAP occurs that could only have been detectable by an ILRT. The probability is calculated in accordance with the methodology established by EPRI Topical Report 1018243[5] (Reference 9), hereafter referred to as the "EPRI topical report," for a Class 3b containment failure.

The second step in the analysis is to add a new basic event to the MNGP PRA model to logically OR the pre-existing leak basic event with the low pressure ECCS pumps, such that the pumps are failed when the leak occurs. The basic event is logically ORed with the basic events that model a common cause failure of all Residual Heat Removal (RHR) and Core Spray pumps. An example is shown below:



5. EPRI Topical Report 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," is Revision 2-A of EPRI Topical Report 1009325.

The next step is to quantify the PRA model and determine which PRA sequences would be risk significant if CAP is required. For the internal events model, the top sequences are:

- medium Loss of Coolant Accident (LOCA) with successful high pressure injection but failure of low pressure injection,
- large LOCA and failure of low pressure injection, and
- several high pressure transients.

The medium LOCA contributes approximately 70% to the CAP risk and the large LOCA contributes 14.5% to the increase in CAP risk. The remaining approximately 15% of the CAP risk is comprised of various transients, most of which are transients where the Reactor Pressure Vessel (RPV) initially remains at high pressure.

For the significant sequence of a medium LOCA with successful high pressure injection, MAAP analysis was performed to determine if CAP is required when the containment is failed due to a large pre-existing leak. The MAAP analysis found that CAP was required if containment cooling was not available. The MAAP analyses also indicated that if torus cooling was established within 3 hours with one RHR pump, one RHR heat exchanger, and one Residual Heat Removal Service Water (RHRSW) pump that the resulting suppression pool temperature was adequate to preclude the need for CAP for the low pressure ECCS pumps (Reference 8).

Several other MAAP internal event cases were developed which bound the set of initiating events modeled in the MNGP internal events PRA. These cases show that a range of possible torus cooling initiation timelines could be modeled in the PRA, depending on the characteristics of the specific accident sequence. These MAAP analyses show that in nearly all transient cases, CAP is not required as long as torus cooling is established prior to 2.5 hours into the accident. Due to limitations of the MAAP computer code, MAAP analyses were not performed for the large LOCA accident. Therefore, torus cooling initiation prior to losing ECCS NPSH with a loss of CAP is not credited for a large LOCA in this PRA analysis (Reference 8).

For the large LOCA, an assessment in the BWROG CAP topical report [(6)] on the application of CAP provides insights regarding the plant response to a large LOCA in particular, the design basis double ended guillotine break of the largest pipe in the Reactor Coolant System (RCS) and it shows that adequate NPSH can be assured if either:

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6. On December 21, 2016, the final Safety Evaluation for the BWROG TR NEDC-33347P, Revision 1, "Containment Overpressure Credit for Net Positive Suction Head (NPSH)," was issued by the NRC.

1) plant conditions (initial suppression pool temperature, river water temperature, etc.), favorable to ECCS NPSH exist, or
2) operators throttle the low pressure ECCS pumps which reduces the amount of NPSH required for pump operation.(7)

Given the low frequency of a large LOCA in the MNGP internal events PRA, a large LOCA concurrent with a loss of CAP due to containment leakage has been assumed to result directly in core damage. This assumption over-estimates the total risk for loss of CAP due to containment leakage but does not affect the conclusion of the risk analysis based on the Regulatory Guide 1.174 risk acceptance guidelines.

Fire Assessment

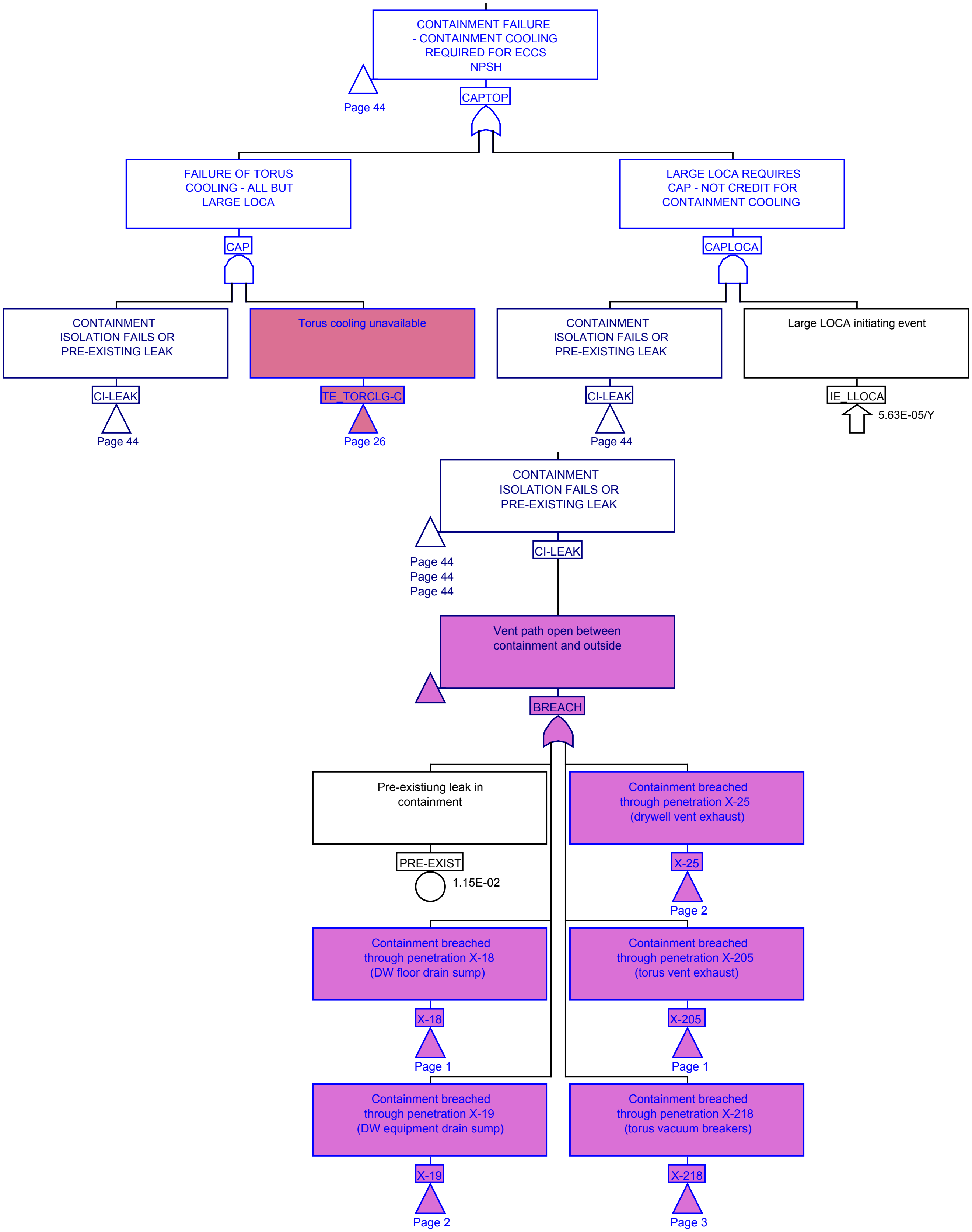
To address fire-induced accident sequences in the Fire PRA not bounded by the plant response to internal events, additional MAAP analyses were developed. These MAAP analyses were performed to determine if torus cooling can be initiated in time to preclude the need for CAP to maintain adequate NPSH for the low pressure ECCS.

In particular, the Fire PRA model introduces multiple stuck-open relief valve (SORV) accidents which can occur due to fire induced hot shorts or fire induced damage to cables associated with the automatic depressurization signal. The most limiting fire induced SORV accident is one in which all 8 SRVs open due to fire damage, which could occur in a large Cable Spreading Room fires, or in specific electrical cabinet fires in the Control Room, such as a fire in the Control Board (Section C03), where cables associated with the control switches for all of the SRVs could be impacted by a fire. In such a fire induced accident, operators would use the Control Room abandonment procedure to establish plant control at the Alternate Shutdown Panel. The Control Room abandonment procedure instructs the operators to depressurize the reactor and establish Core Spray flow and then establish torus cooling. MAAP cases were analyzed that included fire-induced spurious opening of all 8 SRVs at the initiation of a fire induced accident to verify that the establishment of torus cooling precludes the need for CAP credit given an unisolated containment. The MAAP analysis shows that torus cooling needs to be initiated within 2 hours to preclude the need for CAP to ensure adequate low pressure ECCS NPSH (Reference 8).

7. While the BWROG CAP topical report indicates adequate NPSH can be assured if operators throttle the low pressure ECCS pumps, NSPM has NOT credited throttling the pumps in the analyses supporting these RAI responses.

Internal Events and Fire Model Logic

MAAP analyses were performed that form the basis for the success criteria in the internal events and Fire PRA models and represent the best-estimate of the timing required to initiate torus cooling to preclude the need for CAP to ensure adequate low pressure ECCS NPSH. Given the results of the MAAP analyses, the MNGP internal events and Fire PRA models were adjusted to credit torus cooling initiation to prevent loss of low pressure ECCS NPSH if the containment had a large leak or were unisolated for all initiating events. For two types of events, MAAP analyses were not performed. These are the large LOCA and the Anticipated Transient Without Scram (ATWS) events. As described previously, MAAP analyses were not performed for the large LOCA, and hence no credit could be applied for torus cooling. The ATWS event was screened out based on the low contribution to risk. The CAP logic is provided on the following page.


CONTAINMENT FAILURE - CONTAINMENT COOLING REQUIRED FOR ECCS NPSH
Page 44
CAPTOP
FAILURE OF TORUS COOLING - ALL BUT LARGE LOCA
LARGE LOCA REQUIRES CAP - NOT CREDIT FOR CONTAINMENT COOLING
CAP
CAPLOCA
CONTAINMENT ISOLATION FAILS OR PRE-EXISTING LEAK
Torus cooling unavailable
CONTAINMENT ISOLATION FAILS OR PRE-EXISTING LEAK
Large LOCA initiating event
CI-LEAK
TE_TORCLG-C
CI-LEAK
IE_LLOCA
5.63E-05/Y
Page 44
Page 26
Page 44
CONTAINMENT ISOLATION FAILS OR PRE-EXISTING LEAK
CI-LEAK
Page 44
Page 44
Page 44
Vent path open between containment and outside
BREACH
Pre-existiung leak in containment
Containment breached through penetration X-25 (drywell vent exhaust)
PRE-EXIST
1.15E-02
X-25
Page 2
Containment breached through penetration X-18 (DW floor drain sump)
Containment breached through penetration X-205 (torus vent exhaust)
X-18
Page 1
X-205
Page 1
Containment breached through penetration X-19 (DW equipment drain sump)
Containment breached through penetration X-218 (torus vacuum breakers)
X-19
Page 2
X-218
Page 3

ATWS Assessment

For ATWS sequences, a loss of CAP would have a negligible impact on CDF and LERF. The probability of Reactor Protection System (RPS) failure in the MNGP model is 5.9E-06. The probability of a pre-existing leak for an ILRT test interval of once in 15 years is 1.15E-02. Multiplying the two probabilities yields 6.78E-08. If power control fails, the sequence progresses to core damage and CAP has no impact. If power control succeeds, high pressure injection from the Feedwater or High Pressure Coolant Injection (HPCI) systems would have to fail in order to require depressurization and low pressure injection. If low pressure injection is required early due to a failure of the HPCI and Feedwater systems; the Condensate, Core Spray, and Low Pressure Coolant Injection (LPCI) systems can provide low pressure makeup. Note that the Condensate system does not depend on CAP because it does not depend on the torus as a suction source. Using a 0.1 failure probability for all high pressure and low pressure injection and multiplying by the chance of RPS failure and the chance of a pre-existing leak yields a probability of 6.78E-09. This is considered negligible for CDF and LERF. For LERF, in vessel recovery using the Feedwater and Condensate systems can be credited to further reduce LERF. Including an initiating event frequency with these estimates would further decrease the CDF and LERF due to an ATWS concurrent with a loss of CAP due to a pre-existing leak.

Seismic Risk Assessment

The baseline seismic risk is estimated using the Generic Issue (GI)-199 report, which developed a CDF estimate calculated by using seismic hazard curves and using an assumed plant capacity to produce a CDF estimate. The assumed plant capacity was developed by presuming the high confidence of low probability of failure (HCLPF) is equal to the safe shutdown earthquake (SSE) and review level earthquake as described in Appendix C of GI-199 and using a generic lognormal composite uncertainty factor of 0.4 to find the median seismic capacity of the plant. The GI-199 report states that this method is a lower bound on the plant HCLPF and thus seismic risk could be overestimated (Reference 6).

Using the available seismic risk estimate and the lessons learned from previous seismic PRAs, the following discusses how a loss of CAP could impact a change in CDF and LERF due to the ILRT extension given a seismic event. Generally, seismic risk is dominated by failure of key plant structures and key plant systems due to seismic motion which exceeds the capacity of the key structures and systems. A general practice is to treat equipment of the same type at the same elevation as being failed due to the seismic event, so seismic events that fail all RHR pumps are much more likely than seismic events that only fail some RHR pumps while others survive. Failing all RHR pumps reduces the importance of CAP credit for the low pressure ECCS because with the RHR system failed and containment intact, the likely method for decay heat removal is to vent the containment, which precludes credit for the Core Spray system in the MNGP PRA. Thus, there should be little change in CDF due to credit for CAP on seismic risk studies because risk is dominated where CAP credit would not reduce risk because the key structures and systems are already failed and these sequences are significant to seismic risk.

Considering this, it is assumed that seismic events that do not fail the RHR system or fail key structures and/or cause a large LOCA will be similar to the internal events Loss of Offsite Power (LOOP)/Station Blackout (SBO) sequences with an intact RCS. The MAAP analyses demonstrate that establishing torus cooling prior to 6 hours precludes the need for CAP credit for low pressure ECCS NPSH (Reference 8). If a seismic LOOP occurs but an Emergency Diesel Generator (EDG) is functional, it is likely that torus cooling will be established prior to challenging low pressure ECCS NPSH. If a SBO occurs due to a seismic event and EDG failures (thus failing RHR and Core Spray), the FLEX(8) strategy can be implemented to provide a means of inventory control and containment heat removal that does not depend on the torus as a suction source.

Thus, for seismic risk, the change in CDF 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 (failing RHR and Core Spray), the FLEX strategy can be used to provide reactor inventory makeup and decay heat removal.

Conclusions

The PRA analysis described in this response inherently captures the total change in CDF due to the ILRT extension and credit for CAP. It also captures the change in LERF due to the ILRT extension for both sequences that credit CAP, and sequences where either loss of CAP does not fail the low pressure ECCS, core damage occurs due to other failures unrelated to CAP, or where the containment is not isolated due to the pre-existing leak. This is because the pre-existing leak basic event (BE) is assumed to result in a loss of containment isolation in the MNGP PRA model used for this risk analysis.

8. FLEX stands for "Diverse and Flexible Coping Strategies."

The results of the CDF and LERF quantification are propagated through the EPRI methodology to produce the LAR risk metrics. The LAR risk metrics (including the change in risk due to steel corrosion, credit for CAP in the internal events model, and credit for CAP in the Fire PRA model) are shown in the response to RAI 6a.

c. Confirm that PRA model changes did not result in CAP-related non-minimal cutsets (e.g., inappropriate combination of random and CAP-related failures).

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

The PRA model changes were reviewed and do not result in CAP related non-minimal cutsets.

d. The external events risk contribution described in the LAR does not include CAP-related risk. Include CAP-related risk in the evaluation of external events risk, and describe your approach.

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

The external events risk contribution described in the LAR did include analysis of the risk from fires using the MNGP Fire PRA. The seismic events risk contribution is based on the GI-199 seismic risk study. The Fire PRA analysis within the LAR has been superseded by the analysis described in Part 4b of this RAI. The Fire PRA explicitly modeled a loss of CAP. Seismic events were assessed as insignificant to the change in CDF from loss of CAP as discussed in Part 4b of this RAI, but the contribution to LERF is included based on the EPRI methodology of taking the total seismic CDF and applying the Class 3b failure probability to the different ILRT surveillance test intervals. An additional seismic risk sensitivity analysis is provided in the response to RAI 1 Part a, and a total risk sensitivity is included in the response to RAI 6a.

e. The LAR uses a method of assuming that the CAP-related change in CDF is equal to change in LERF. Describe the method used for estimating the updated change in LERF if different from the LAR.

Response to RAI 4e

The change in CDF and change in LERF are explicitly estimated using the MNGP PRA models for internal events and for fire events, as described in the response to RAI Part 4b. For seismic events, the change in CDF and LERF is estimated as described in the response to RAI Part 4b and Part 4d.

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**

a. Confirm the list above is a complete list of CAP-related key assumptions and sources of uncertainty.

Response to RAI 5a

A summary is provided in this response. Each specific item is addressed in the sub-parts to this RAI question.

	Potential CAP-Related Key Assumptions and Uncertainties:	KA/ KU	See RAI Response	Notes
1.	Existing plant conditions impacting NPSH	KA	RAI 5b.i	Not credited in the RAI responses.
2.	Operator action to throttle low pressure ECCS flow	KU	RAI 5c (A)	Proceduralized plant response to challenging ECCS NPSH. Not credited in the RAI responses.
3.	High Pressure Injection (HPI) lube oil cooling	----	RAI 5d (A)	PRA includes failure of HPI systems if temperature limits exceeded.
4.	Systems available following a loss of Instrument Air (IA) system initiating event	----	RAI 5e (A)	PRA model includes loss of IA initiating event. Model represents as-built, as-operated condition of plant.

	Potential CAP-Related Key Assumptions and Uncertainties:	KA/ KU	See RAI Response	Notes
5.	Loss of Offsite Power (LOOP)/ SBO PRA modeling	----	RAI 5f	PRA models LOOP and SBO risk and is reflective of the as-built, as-operated plant.
6.	Containment leakage rate that would impact CAP	KA	----	This is plant dependent. Studies provide conservative estimates.
7.	ECCS pumps will fail given a loss of NPSH	KA	----	This is a conservative key assumption because the pumps may survive under marginal NPSH conditions which, if credited, would reduce the CDF and LERF.
8.	Containment leak probability is based on assumptions in EPRI topical report, which are stated to be conservative.	----	----	No credit taken for alternate means of containment leakage detection. Several methods are available at MNGP to detect containment leaks.
9.	No credit given to other sources of reactor makeup not currently in PRA model. While the FLEX pumps are mentioned as available in several of the RAI responses, herein, they are NOT credited in the current PRA model.	----	----	The FLEX pumps do not depend on the suppression pool as a suction source.

KA – Key Assumption
KU – Key Uncertainty
(A) – Previously provided by letter L-MT-17-002, dated January 31, 2017.

b. The LAR applies PRA credit for existing plant conditions impacting NPSH. The probabilities are given to be 0.1 and 0.5 (Section 5.3.4 of the LAR). Address the following:

i. Clarify what the assumed probabilities of 0.1 and 0.5 represent and explain their technical basis.

<u>Response to RAI 5b.i</u>

As described herein, credit for plant conditions was not taken using a judged probability estimate for these RAI responses, but was assessed using the MAAP analysis as described in the response to Part b of RAI 4.

For the LAR, the assumed probabilities represent the chance that plant conditions preclude the need for CAP credit as described in the BWROG CAP topical report. The technical basis is the results of the analyses presented in Appendices A and B, both entitled, “DBA-LOCA Containment Response Evaluation for Use in NPSH Evaluation for Monticello Nuclear Generating Plant,” to that report. Uncertainty exists because the conditions and scenarios assumed in Appendix A and Appendix

B may not be applicable to all modeled PRA accidents. The 0.1 probability is based on the analysis in Appendix A which included, among other relevant parameters, the probability that a containment leak occurs of sufficient size to defeat CAP given a Design Basis Accident (DBA) LOCA (which is a large LOCA in the MNGP PRA model). The analysis in Appendix A shows that for the short term injection scenario, adequate NPSH is assured because the calculated available NPSH is greater than the required NPSH with 95% confidence. Therefore, there is a 5% chance that NPSH available is lower than the calculated value. The true failure probability is somewhere between 0 and 5%, because the calculated NPSH available is greater than NPSH required. The 0.1 probability represents conservative judgment based on doubling the already conservative 5% chance that conditions are inadequate to preclude the need for CAP for early injection scenarios.

The 0.5 probability is based on the analysis in Appendix B of the BWROG CAP topical report. The analysis in Appendix B postulated that the containment could not develop accident pressure (rather than using the probability estimate included in the Appendix A analysis). Given that NPSH available is below the NPSH required value with 95% confidence for some amount of time in the Appendix B accident scenario and the conditions assumed may not reflect the conditions in all PRA accidents, it is judged that a 0.5 chance of failure is appropriate. The 0.5 probability can be used to represent an unknown failure probability where the chance is not a guaranteed failure or guaranteed success.

ii. **The ILRT frequency extension CAP risk assessment postulates a pre-existing flaw sufficient to defeat CAP. Explain why the NPSH justification provided above for these two probabilities is applicable given that CAP is defeated. If the justification is not applicable remove this PRA credit.**

Response to RAI 5b.ii

The analyses performed for these RAI responses did not take PRA credit for the BWROG CAP topical report considerations.

For the LAR, the analysis conditions in Appendix A of the BWROG CAP topical report assessment were considered applicable since the analysis applied directly to the MNGP and included the chance that a containment leak sufficient to defeat CAP could occur based upon a deterministic and statistical approach. Therefore, it was considered applicable because the probability of containment failure was included in the analysis in Appendix A of the topical report. The analysis conditions in Appendix B of the BWROG CAP topical report assumed for the MNGP that there was pre-existing containment leakage sufficient to result in a loss of containment overpressure combined with a DBA-LOCA (with no additional safety system failures), could still demonstrate acceptable results.

iii. **EPRI guidance includes two general categories of events involving early injection or gradual containment heat up with long term injection. If PRA credit is determined to be applicable in part ii for plant conditions impacting NPSH, describe how the PRA model is adjusted to account for plant conditions challenging NPSH. Explain how it was determined to use the probabilities of 0.1 or 0.5 on a sequence-by-sequence basis for PRA sequences belonging to the two EPRI general event categories. Account for sequence timing with respect to NPSH conditions (e.g., LOCAs, SBO-recovery, etc). If sequences have not been analyzed in sufficient detail to distinguish between which probability applies to the PRA sequences, then describe how the PRA model is adjusted to account for plant conditions challenging NPSH. Explain which sequences are credited. If the PRA credit does not have technical justification for plant conditions which challenge NPSH, remove this PRA credit.**

<u>Response to RAI 5b.iii</u>

In lieu of using the probability of inadequate plant conditions to preclude requiring CAP, which is based on judgment for some of the PRA scenarios, a revised CAP analysis is presented in the response to RAI Part 4b.

f. **Discuss how the PRA modeled LOOP and SBO-related loss of CAP for the CAP-related CDF and LERF. Explain whether or not LOOP/SBO are risk-significant contributors for CAP-related risk and provide justification for your conclusion.**

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

LOOP and SBO are not risk significant contributors to CAP-related risk. The PRA modeled LOOP and SBO-related loss of CAP because, as the analysis discussed in the response to RAI Part 4b describes, MAAP analyses demonstrate that CAP credit can be precluded if torus cooling is initiated within 6 hours. In LOOP sequences with an EDG available, torus cooling is modeled as discussed in the response to RAI Part 4b.

The MNGP model contains SBO sequences that credit EDG recovery or offsite power recovery. The probability of recovery is a function of time. The probability of not recovering AC power by 6 hours in the internal events model is approximately 30%. For sequences where AC power is recovered after the point in which torus cooling can mitigate a loss of ECCS NPSH with CAP unavailable, systems with suction sources other than the torus can be used to provide inventory while decay heat removal can be accomplished using the hardened wetwell vent. The additional inventory methods are Condensate, Control Rod Drive Hydraulics (CRDH), and RHRSW systems and Fire Water injection using the RHR LPCI cross-tie valves. With offsite power recovered, any of these sources can be used for inventory makeup. With only EDG recovery, the RHRSW, CRD, and Firewater systems can be used but the Condensate system would be unavailable.

g. Discuss how the containment leakage rate that would fail CAP is considered in the CAP risk assessment. Discuss the basis for this assumption and explain how it impacts the CAP risk assessment.

Response to RAI 5g

The CAP risk assessment described in the response to RAI Part 4b assumes that any large pre-existing leak that occurs is sufficient to defeat CAP. The leak probability is calculated as discussed in the response to RAI Part 4b and is consistent with the EPRI topical report methodology for calculating a pre-existing leak large enough to cause a LERF if core damage were to occur. The same probability is used and is assumed to defeat CAP. The probability uses the Jeffreys non-informed prior to estimate the probability of a leak. The Jeffreys non-informed prior is used because no events sufficient to cause a large leak have occurred, as discussed in the EPRI topical report. Assuming different failure probabilities calculated by taking different failure rates and exposing them to the same time would result in a different set of risk results. The analysis used for MNGP is considered reasonable since it relies on the EPRI topical report method for calculating a large pre-existing leak. The discussion in Section 4.2.6, “Containment Overpressure,” of the EPRI topical report describes how the Class 3b contribution can be used to model a loss of CAP to the systems that depend on CAP. This discussion implies the method of calculating the Class 3b contribution is valid for determining the probability that a pre-existing leak is of sufficient size to defeat CAP.

h. If the PRA was adjusted to account for key assumptions or sources of uncertainties, or otherwise changed for the CAP risk assessment, describe these PRA changes. Provide updated results (change in CDF and change in LERF) for the CAP risk assessment.

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

The PRA model CAP analysis and results are described in the response to RAI Part 4b.

i. Provide sufficient technical justification for any PRA credits which are significant in addressing any of the key assumptions/sources of uncertainties for the CAP risk assessment. Include in the discussion how the key assumptions/ sources of uncertainty are addressed, if necessary, for CAP to help ensure that the overall plant risk metrics for the ILRT frequency extension request would be within acceptance guidelines.

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

The CAP risk assessment discussed in the response to RAI Part 4b does not include credit for all of the key assumption or uncertainty items listed in RAI 5. As discussed in the response to RAI 5, only the probability of plant conditions that precludes requiring CAP for low pressure ECCS NPSH (which was an assumption and uncertainty) and the estimate of containment leakage that defeats CAP (which is an uncertainty) are assumptions or uncertainties. The CAP risk assessment in response to RAI Part 4b does not use a probability of plant conditions that would preclude CAP credit. The CAP risk assessment in the response to RAI Part 4b uses the Jeffreys non-informed prior to estimate the probability of a containment leak and assumes any such leak is sufficient to defeat CAP. Since the probability of plant conditions that would preclude requiring CAP to defeat low pressure ECCS NPSH is not used in the risk assessment, there is no impact on the results of the risk assessment. For the containment leakage that would defeat CAP, the discussion in the EPRI topical report supports the use of the Class 3b contribution as the method to estimate the change in CDF from a loss of CAP, and no additional justification is necessary.

RAI 6

The results of the ILRT extension risk assessment are given in Table 6-1 of the LAR for the change in population dose, change in percent dose, change in LERF, and change in conditional containment failure probability (CCFP). These results appear to be based on internal events only.

a. In addition to the internal events risk, the risk contribution from external events, steel liner corrosion, and CAP, should also be included in the overall plant results for the risk metrics (change in population dose, change in percent dose, change in LERF, and change in CCFP), as appropriate. Accordingly, provide updated results for the LAR which reflect changes made for the ILRT frequency extension risk assessment.

Response to RAI 6a

The risk contribution from external events, steel liner corrosion, and CAP, has been included in the overall plant results for the risk metrics (change in population dose, change in percent dose, change in LERF, and change in containment failure probability (CCFP), as appropriate. Note that seismic risk is not included in the change in population dose, change in percent dose, and change in CCFP, because only a seismic CDF point estimate is available. For the purpose of providing a more realistic change in dose estimate, Class 7 was divided into two classes. Class 7a represents a small early release and the Class 7a frequency is removed from the Class 7 frequency; the remaining Class 7 frequency represents various release categories and remains in Class 7. The Class 7a dose is taken from Reference 10; the new Class 7 dose excludes the small early release. The MNGP level 2 model was used to estimate the small early release frequency for the ILRT extension intervals in order to characterize the dose increase from the CDF increase due to a loss of low pressure ECCS NPSH.

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	9.08E-02	1.11E-05	8.08E-02	1.01E-05	7.36E-02
2	2.34E+06	1.50E-06	3.50E+00	1.50E-06	3.50E+00	1.50E-06	3.50E+00
3a	7.29E+04	4.71E-07	3.43E-02	1.58E-06	1.15E-01	2.36E-06	1.72E-01
3b	7.29E+05	1.17E-07	8.54E-02	3.92E-07	2.86E-01	5.88E-07	4.28E-01
4-6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Class	**Dose (person-rem)**	**ILRT Extension Summary Base Case 3 in 10 Years CDF/Year**	**Person-Rem/Year**	**Extend to 1 in 10 Years CDF/Year**	**Person-Rem/Year**	**Extend to 1 in 15 Years CDF/Year**	**Person-Rem/Year**
7	1.88E+06	4.21E-05	7.92E+01	4.23E-05	7.94E+01	4.23E-05	7.96E+01
7a	3.87E+05	1.07E-06	4.15E-01	2.39E-06	9.25E-01	3.36E-06	1.30E+00
8	1.53E+06	8.83E-07	1.35E+00	8.83E-07	1.35E+00	8.83E-07	1.35E+00
Total		**5.86E-05**	**8.47E+01**	**6.01E-05**	**8.57E+01**	**6.11E-05**	**8.64E+01**

ILRT Dose Rate from 3a and 3b				
ΔTotal Dose Rate	From 3 Years	N/A	0.996	1.69
	From 10 Years	N/A	N/A	0.695
%ΔDose Rate	From 3 Years	N/A	1.18%	2.00%
	From 10 Years	N/A	N/A	0.811%
LERF				
ΔLERF	From 3 Years	N/A	2.76E-07	4.65E-07
	From 10 Years	N/A	N/A	1.89E-07
CCFP %				
ΔCCFP %	From 3 Years	N/A	0.983%	1.66%
	From 10 Years	N/A	N/A	0.68%

The ILRT extension analysis included a change in CDF estimate due to loss of containment accident pressure for low head ECCS NPSH. The total delta CDF is the sum of the delta CDF for internal events and the delta CDF for fire events. The following table shows the delta CDF results.

ILRT Extension Δ CDF			
	Base Case 3 in 10 Years	**Extend to 1 in 10 Years**	**Extend to 1 in 15 Years**
CDF	5.86E-05	6.01E-05	6.11E-05
Δ CDF (3/10 baseline)	N/A	1.43E-06	2.47E-06
Δ CDF (1/10 baseline)	N/A	N/A	1.04E-06

The total change in CDF and total change in LERF, when fire risk and internal events risk is combined, are in Region II of the Regulatory Guide 1.174 risk acceptance guidelines. As shown in the preceding table, the total CDF with the ILRT extended to 1 in 15 years is 6.11E-05, which is below the Region II upper boundary of 1E-04. The total fire and internal events LERF is estimated as 7.42E-06, which is below the Region II upper boundary of 1E-05. Based on these results it is reasonable to conclude that the changes in CDF and LERF are “small” per the Regulatory Guide 1.174 risk acceptance guidelines.

As an additional sensitivity evaluation, when the seismic CDF estimate shown in RAI 1a is included with the total CDF estimate from the internal events PRA and the Fire PRA, the total CDF is 7.19E-05, which is within Region II of Regulatory Guide 1.174. The seismic LERF estimates in the response to RAI 1a vary based on whether a small pre-existing leak is postulated to grow to a large leak given a seismic event with probability of 1.0 or if no such growth phenomena is postulated. When totaling the seismic analysis from RAI 1a (assuming guaranteed crack growth) and the internal events and fire CAP analysis from RAI 4b, the most conservative estimate (the 3 in 10 to 1 in 15 case) of delta LERF comes from adding the seismic delta LERF and internal events and fire delta LERF: 4.42E-07 + 4.65E-07 = 9.07E-07. Total LERF comes from adding the seismic baseline LERF, seismic delta LERF (assuming guaranteed crack growth), and internal events and fire CAP LERF: 1.25E-06 + 4.42E-07 + 7.43E-06 = 9.12E-06. These total estimates are near the upper boundary of Region II of Regulatory Guide 1.174, however, if a probability of less than 1.0 is assigned to the chance that a small leak grows to a large leak as a result of a seismic event, these totals would decrease, or if the change in LERF is based on the 1 in 10 to 1 in 15 interval, the results would also decrease. Based on the results of this sensitivity evaluation, it is reasonable to conclude that the change in CDF and LERF are “small” per the Regulatory Guide 1.174 risk acceptance guidelines.

The change in CCFP is 1.66% which is slightly above the 1.5% criteria assumed to be “very small” in most ILRT extension analyses. However, due to the dependence on CAP for low pressure ECCS NPSH, the CDF increase associated with the increase probability of pre-existing leak is by definition core damage with a breach in containment, so the increase in CCFP is expected and is considered small compared to plants that do not rely on CAP. The calculated CCFP for CAP is considered to be conservative since some of the sequences that would cause containment overpressure (without CAP) were retained as containment failures instead of possible transition to an intact state (with CAP).

The change in percent dose is 2.0%, which is larger than the 1.0% total dose criteria assumed to be “small” in most ILRT extension analyses. Given the dependence on CAP for low pressure ECCS NPSH, the increase in CDF results in a larger than normal dose increase compared to a plant that does not rely on CAP for low pressure ECCS NPSH, and therefore the dose increase is expected to be larger than typical, but is still considered small for this application.

The change in population dose is 1.69 person-rem, which is larger than the less than 1.0 person-rem criteria assumed to be “small” in most ILRT extension analyses. Given the dependence on CAP for low pressure ECCS NPSH, the increase in CDF results in a larger than normal dose increase compared to a plant that does not rely on CAP for low pressure ECCS NPSH, and therefore the population dose increase is expected to be larger than typical, but is still considered small for this application. The calculated population dose for CAP is considered to be conservative since some of the sequences that would cause containment overpressure (without CAP) were retained as containment failures instead of possible transition to an intact state (with CAP). The containment failure states have higher population doses than the intact population dose.

The analysis and sensitivity studies performed for these RAIs were done without crediting: 1) throttling of the low pressure ECCS pumps, or 2) online leakage monitoring that was discussed in the LAR and approved under EPU.[9] Considering that CAP is required for certain scenarios to ensure adequate low pressure ECCS NPSH, an increase in CDF would be expected when credit is removed. Hence, an increase in percent dose and population dose would be expected. It is informative to note the relatively small change in percent dose; i.e., 2.0% versus the less than 1.0% common standard, and in population dose; i.e., 1.69 person-rem versus the less than 1.0 person-rem common standard criteria that have been assumed to be “small” in most prior ILRT extension analyses.

In and of themselves these percent dose and population dose results indicate the relative acceptability of this risk informed application. While not quantified, if consideration were given to other potential analysis assumptions, e.g.; 1) throttling the low pressure ECCS pumps, 2) online leakage monitoring – decreases the probability of a pre-existing leak, and 3) features that could be relied upon that are not dependent on

9. Note that both of these features are credited in the non-risk informed portions of the MNGP licensing basis, most recently reviewed by the NRC under the EPU license amendment.

CAP, such as FLEX, the results would be lower. Therefore, the results, indicate the acceptability of an increase in the surveillance interval to 15-years for the ILRT.

b. Provide the overall plant results (the change in population dose, change in percent dose, change in LERF, and change in CCFP) for a sensitivity analysis in which PRA credit is removed from the CAP risk assessment for 1) plant conditions impacting NPSH, and 2) operator action to throttle low pressure ECCS flow.

Response to RAI 6b

The analysis discussed in the response to RAI Part 4b did not credit a probability estimate for plant conditions impacting NPSH or the operator action to throttle low pressure ECCS flow, so there is no change to the results documented in the response to RAI Part 6a.

c. Evaluate the need to decrease the change in LERF to account for key assumptions and uncertainties. Discuss any potential measures to decrease the change in LERF if necessary.

Response to RAI 6c

The key assumptions and uncertainties that are used in the CAP risk assessment discussed in the response to RAI Part 4b, have the net impact of over-estimating the change in LERF. Given that the LERF is over-estimated, there is no need to discuss measures that could reduce the LERF. However, some measures that can reduce LERF are as follows:

- Perform additional analysis that shows that the low head ECCS pumps will survive when NPSH is lost and provide sufficient flow for the PRA mission time. This would reduce the calculated change in LERF.

- Include sequence specific torus cooling timing in each sequence in the PRA model, as opposed to the bounding value (shortest time to implement torus cooling) determined using MAAP analyses. This would reduce the calculated change in LERF.

- Credit other means of containment heat removal with ensuring adequate ECCS NPSH is maintained if CAP is not available. The PRA model used in the CAP analysis only credits torus cooling. This would reduce the calculated change in LERF.

- Add credit for FLEX into the appropriate PRA model sequences as another method of providing inventory makeup and decay heat removal that does not require the torus as a suction source. This would reduce the change in LERF.

- Take credit for decreased time to detect containment leakage via the existing MNGP leakage monitoring that supports the MNGP design basis for crediting CAP, which is performed more frequently than the ILRT. These tests are expected to detect at least some portion of containment leakage events that would be detectable by an ILRT. Given that the risk analysis uses the probability of containment failure/leakage as an input, additional detection credit would reduce the change in LERF.

d. The LAR Section 5.3.1 of Enclosure 2 shows that the total LERF for the plant is 9.08E-6/yr, which is close to the Regulatory Guide (RG) 1.174 acceptance guideline of 1E-5/yr for LERF. Due to the small margin to 1E-5/y for total baseline LERF, the NRC staff requests that if the total LERF is above the RG 1.174 acceptance criteria of 1E-5/yr as a result of updated analyses, then discuss measures which may be taken to address this acceptance guideline. In addition, confirm that the total baseline CDF remains below the RG 1.174 guideline of 1E-4/yr, or, if this is not the case, address measures to reduce the total CDF also.

<u>Response to RAI 6d</u>

As shown in the updated risk analysis in response to RAI 6a, the total LERF is below the Regulatory Guide 1.174 risk acceptance guideline. However, the items discussed in response to Part c of this response shows that the total change in LERF is over-estimated. For the total baseline LERF, the largest contributor to the LERF is the Fire PRA. Measures that can be taken to reduce fire risk include:

- Perform additional analysis/refinement of the risk significant fire PRA scenarios.

- Credit FLEX as an additional means of reactor inventory makeup and decay heat removal for risk significant fire PRA scenarios

- Credit administrative controls that limit fire frequencies in risk significant fire areas.

Because the change in CDF due to loss of CAP has been shown to be in Region II in the response to RAI 6a, the total CDF was also assessed. The total CDF was shown to be in Region II.

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. 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

4. Boiling Water Reactor Owners Group (BWROG) Licensing Topical Report NEDC-33347P, Revision 1, “Containment Overpressure Credit for Net Positive Suction Head (NPSH)"

5. Nuclear Energy Institute letter to NRC (E. Leeds), “Seismic Risk Evaluations for Plants in the Central and Eastern United States, Project Number 689,” dated March 12, 2014 (ADAMS Accession No. ML14083A584)

6. Generic Issue 199 (GI-199), “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment” dated September 2010.

7. 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

8. NSPM Calculation PRA-CALC-16-006, “MAAP Analysis for ILRT Extension,” Revision 0

9. EPRI Topical Report 1018243, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,” dated October 2008

10. NSPM, Monticello Nuclear Generating Plant, “Application for Renewed Operating License, Appendix E – Environmental Report,” 2005:

Findings and Observations (F&Os)

Change Number: MT-17-0005

F&O Number: FO-1
Associated SR(s): FSS-D3, FSS-D4

Detailed Problem Description: The analysis results of the thermal heat soak method appear to credit ventilation limited burning in several PAUs without providing sufficient basis. An example is the Group 1 scenarios listed in Table J-6 of 016015-RPT-06. Each of the four fire case CFAST results have sensitivity cases due to the development of ventilation limited conditions. The baseline CFAST results do not result in damage to a generic target over a 60 min time interval. The CFAST sensitivity cases that were originally run with additional ventilation to verify constant exposure damage times would likely result in damage to a generic TP target when assessed in the heat soak model.

Proposed Solution: Evaluate the performance of the thermal heat soak model for the parameter sensitivity cases indicated in the SCA and where applicable for the MCA, or provide additional validation basis for the selection of the ventilation limited configurations used in the updated analysis.

Basis for Significance: The original evaluation included ventilation limited configurations that were outside the recommended validation range specified for the fire modeling tools applied. The results were then justified based on conservative outcomes relative to additional parameter sensitivity evaluations. This exercise was not documented for the thermal heat soak method. Therefore, it may be possible that the updated results are not validated, or justified as conservatively bounding.

Actual Solution: The heat soak method was extended to the sensitivity analysis (Section J.5 of FPRA-MNGP-SCA [Reference 51]) in order to evaluate the effect in simulations where no oxygen depletion is assumed. Under the assumption of no oxygen depletion, there is additional heat added to the enclosures, resulting in an increase in CDF of 3.48E-06 and an increase in LERF of 2.69E-07. The increases are approximately an order of magnitude lower than the plant Fire CDF and LERF results. It should be noted that the CFAST simulations assuming oxygen consumption include a lower oxygen limit of one percent. The resulting risk increase is a conservative estimate based on the following:

1) The CFAST volumes chosen for the analysis are selected in 5 different size groups, which bound the actual size of the fire compartments. That is, the fire modeling results are based on fire compartments that are smaller in size than the as built compartments in the plant. Therefore, the resulting temperatures are higher than the ones that would result if the actual compartment size is used.

The CFAST simulations assume worst case combination of ignition source and secondary combustibles (i.e., cable tray arrangement) for the fire compartment. This approach bounds a number of scenarios associated with lesser fire intensities.

Findings and Observations (F&Os)

Impact on ILRT Extension: The F&O was resolved by the performance of additional sensitivity evaluations. The sensitivity evaluations show the increase in CDF and LERF when conservatively including no oxygen depletion in the fire model and when including bounding configurations for the fire model relative to the plant configuration is small. There is no impact to the ILRT extension application because the fire modeling determined that the screening of the scenarios is valid.

Findings and Observations (F&Os)

Change Number: MT-17-0006

F&O Number: FO-2
Associated SR(s): FSS-D3,FSS-D4,FSS-H4, FSS-H5, FSS-H9

Detailed Problem Description: A number of documentation issues have been identified. These include:

a) There are a number of scenarios that appear to credit the thermal heat soak method listed in the FMDB but the HGL times do not match any scenario listed in Report 016015-RPT-06. An example is Equipment C-18 in tblIgnitionScenarios of the fire modeling database. Scenario 2 and the corresponding comment indicates HGL time is 25 minutes based on heat soak time. Table J-6 in Section J-6 of Report 016015-RPT-06 does not list any damage times from any ignition source – secondary combustible grouping of 25 minutes. The database should be checked for additional examples and addressed as necessary.
b) There are a number of scenarios listed in the FMDB indicating HGL timings but there is inconsistent indication for when a scenario credits the thermal heat soak method. The only method to verify that the thermal heat soak method was applied in the FMDB was to query the results in TblIgnitionScenarios and match HGL timing to those reported in Table J-6 in Section J-6 of Report 016015-RPT-06, or to search for comment fields in IgScnComment.
c) Description of the method in which the results from the thermal heat soak analysis is incorporated in the MCA. It is not clear where the MCA heat soak calculations or their direct inputs are among the reviewed material. Section 5.4 of the MCA Report 016015-RPT-08 points to Table J-6 in the SCA Report 016015-RPT-06 for heat soak results. However, the compartments listed in Table J-6 do not completely match the compartments that were screened from the MCA using the heat soak method. This suggests that there may be other heat soak results that are not documented. For example, the MCA screens combinations involving compartments 19B and 32A, but the SCA does not indicate that thermal heat soak analysis was performed for these compartments. In addition it appears that the heat soak method was used to increase the HGL timing for combinations involving compartment 32A, but there is no documentation of the results used to justify this timing.
d) It is difficult to link the CFAST Group and the Fire Case as listed in Table J-6 in the SCA Report 016015-RPT-06 with the damage integral result listed in the database for the SCA and MCA where applied. There is no consolidated table which includes the CFAST Group and the Fire Case as applied to a given scenario in the FMDB.
e) The thermal heat soak method does not fully document the approach for target damage accumulation at low temperatures. No technical deficiencies were noted in the method review; however, the treatment of the low temperature damage accumulation can have a significant influence on the overall result and should be clearly discussed.
f) Additional documentation of the limits of applicability for the thermal heat soak method is needed in Report 016022-RPT-01. For example, is there a maximum exposure temperature or maximum/minimum cable size over which the results can be used?
g) Documentation of sources of model uncertainty and its treatment in the analysis is needed to achieve a Cat II for FSS-H5 and FSS-H9. Since the heat soak method is an interpolation of the

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generic cable damage times listed in NUREG/CR-6850, there is no new uncertainty introduced with the heat soak method, except for the damage accrual estimates at temperatures below the damage threshold.

h) Reports 016015-RPT-06, Rev. 4 and 016015-RPT-08, Rev. 4 were draft at the time of the review. They will need to be finalized and signed.

Proposed Solution: Revise the SCA, MCA, and thermal heat soak documentation to clearly address all identified documentation concerns. Many of these can be easily addressed with minor revisions to existing documentation.

The database should be updated to clearly indicate which scenarios have credit for the heat soak method. It would improve the traceability of the data if a new table was created that stored the heat soak results, and new columns created in the SC and MC tables that indicate whether heat soak is being used, and if so point back to the heat soak table for data source.

Basis for Significance: It is not possible to verify that the application of the thermal heat soak method is appropriate or complete.

Actual Solution:

A) Scenarios crediting the heat soak method have been reviewed. The configuration for Group 2 volumes missing from table J-1 in the Single Compartment Analysis notebook has been added to document the scenarios mentioned in this F&O.
B) Table J-7 has been added to the SCA notebook that indicates the equip and equip type that have the heat soak method applied.
C) Reference to Appendix J has been removed. Table E-2 in Attachment E to the MCA report includes the multi-compartment combinations that were considered for heat soak application.
D) Table J-7 has been added to the SCA notebook that indicates the group and fire case for each equipment that was considered for the heat soak application.
E) Additional technical discussion added to the SCA notebook on the low temperature treatment of the heat soak method. The discussion of cable heating at low temperatures is now included in the analysis.
F) Additional discussion on the limits of applicability for the heat soak method has been included in the SCA notebook.
G) Additional discussion added on uncertainty has been added based on the verification and validation study performed on the heat soak analysis.

The Reports 016015-RPT-06, Rev. 4 and 016015-RPT-08, Rev. 4 are final and signed.

Impact on ILRT Extension: This is a documentation F&O and has been resolved. There is no impact on the ILRT Extension Analysis.