March 31, 2003

Mr. David L. Wilson
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAK-RATE TEST INTERVAL (TAC NO. MB4919)

Dear Mr. Forbes:

The Commission has issued the enclosed Amendment No. 134 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application of April 22, 2002, as supplemented by your letters of October 25, 2002, January 23, and February 12, 2003.

The amendment changes TS Surveillance Requirement 4.7.A.2.b, "Primary Containment Integrity," to allow a one-time, 5-year extension to the 10-year interval for performing the next Type A containment integrated leakage rate test (ILRT). The change allows ILRT testing within 15 years from the last ILRT, which was performed in March 1993.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

L. Mark Padovan, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures: 1. Amendment No. 134 to DPR-22
2. Safety Evaluation

cc w/encls: See next page

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DISTRIBUTION:

PUBLIC	OGC	SWeerakkody	DTerao
PDIII-1 Reading	ACRS	JPulsipher	GBedi
LRaghavan	WBeckner	MRubin	
MPadovan	GHill(2)	RPalla	
RBouling	WLeFave	BBurgess, RGN-III	

ADAMS Accession No. ML030440673 *Provided SE input by memo

OFFICE	PDIII-1/PM	PDIII-1/LA	SLPB/SC*	SPSB/SC*	EMEB/SC*	OGC/NLO	PDIII-1/SC
NAME	MPadovan	THarris for RBouling	SWeerakkody	MRubin	DTerao	RHoefling	LRaghavan
DATE	03/20/03	03/28/03	02/04/03	02/04/03	12/06/02	03/27/03	03/28/03

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated April 22, 2002, as supplemented by letters dated October 25, 2002, January 23, and February 12, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.2 of Facility Operating License No. DPR-22 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 134, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 31, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 134

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
159	—
185	185
—	258

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-22

NUCLEAR MANAGEMENT COMPANY, LLC

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

Nuclear Management Company, LLC's (NMC, or the licensee) application of April 22, 2002, as supplemented by their letters of October 25, 2002, January 23, and February 12, 2003, requested changes to the Technical Specifications (TSs) for the Monticello Nuclear Generating Plant. The supplemental letters of October 25, 2002, January 23, and February 12, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 3, 2002 (67 FR 56324).

The proposed amendment would add an exception to the surveillance requirement of TS 4.7.A.2.b, "Primary Containment Integrity," to permit a one-time, 5-year extension to the 10-year interval for performing the next Type A[1] containment integrated leakage rate test (ILRT). TS 4.7.A.2.b currently requires the licensee to "Perform required visual examinations and leakage rate testing for Type A containment integrated leakage rate tests in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and Regulatory Guide 1.163 dated September 1995." The proposed amendment would continue this statement by adding: "as modified by the following exception: NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the March 1993 Type A test shall be performed no later than March 2008."

2.0 REGULATORY EVALUATION

The licensee requested the proposed changes in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.101, "Filing of application," 10 CFR 50.59, "Changes, tests, and experiments," and 10 CFR 50.90, "Application for amendment of license or construction permit."

---

[1] A Type A test is an overall (integrated) leakage rate test of the containment structure.

The regulations at 10 CFR Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. Monticello TS 4.7 requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995. RG 1.163, Section C, “Regulatory Position,” states “licensees intending to comply with the Option B in the amendment to Appendix J should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 [Nuclear Energy Institute report 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995], rather than using the test intervals specified in ANSI/ANS-56.8-1994.” NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision in NEI 94-01 for extending the test interval an additional 15 months in certain circumstances. The two most recent Type A tests at Monticello have been successful, so the current interval requirement is 10 years.

Thus, the licensee is requesting an addition to TS 4.7.A.2.b that would permit an exception from the guidelines of RG 1.163 regarding the Type A test interval by extending the currently specified 10-year interval to a 15-year interval on a one-time basis. Specifically, the proposed TS states that the first Type A test performed after the March 1993, Type A test shall be performed no later than March 2008. The proposed change would not involve any change to a code, regulatory requirement, or acceptance criteria.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of the proposed license amendment as described in the licensee’s application dated April 22, 2002, and the additional information provided in NMC’s supplemental letters of October 25, 2002, January 23, and February 12, 2003. The detailed evaluation below will support the conclusion that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

By License Amendment No. 95, dated April 3, 1996, the NRC previously approved a revision to the Monticello TSs to incorporate the requirements of Option B for the Type A tests. In License Amendment No. 132, dated February 4, 2003, the NRC approved a revision to the TSs to incorporate the requirements of Option B for the Types B and C local leak rate tests (LLRTs).

3.1 Inservice Inspection for Primary Containment Integrity

The leak rate testing requirements of Option B of Appendix J, and the containment inservice inspection (ISI) requirements mandated by 10 CFR 50.55a complement each other in ensuring the leak-tightness and structural integrity of the containment. Therefore, in this section the NRC staff evaluates the inservice inspection of the containment and potential areas of weaknesses in the containment.

Monticello is a General Electric boiling-water reactor (BWR) enclosed within a Mark I-type containment. The containment design includes a steel drywell and suppression chamber with interconnecting vent pipes with bellows, primary containment access penetrations, and other process piping and electrical penetrations. The integrity of the penetrations and isolation valves are verified through Type B and Type C LLRTs as required by 10 CFR Part 50, Appendix J, and the overall leak-tight integrity of the primary containment is verified through an ILRT. The LLRTs and ILRTs are performed to verify the essentially leak-tight characteristics of the containment at the design-basis accident (DBA) pressure. The last ILRT for Monticello was performed in March 1993. The next ILRT is scheduled to begin in April 2003. With the proposed extension of the ILRT time interval, the next overall verification would be performed no later than March 2008. The licensee provided information related to the ISI of the containment and discussed potential areas of weaknesses in the containment that may not be apparent in the risk assessment. In addition, in its supplemental letter of October 25, 2002, NMC responded to an NRC staff request for additional information (RAI) dated September 23, 2002, to explicitly address six issues. The NRC staff's evaluation of the licensee's responses to these issues is discussed below.

3.1.1 ISI Program at Monticello

The licensee stated that the ISI program was established in 1995, in accordance with Articles IWE and IWL of Section XI of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), 1992 edition with the 1992 addenda, to assure detection of degradation affecting containment integrity. The first 10-year containment inspection (IWE) is divided into three periods as follows:

First Period:	March 20, 1998 - September 8, 2001
Second Period:	September 9, 2001 - September 8, 2005
Third Period:	September 9, 2005 - September 2008

The licensee's inspections performed in accordance with ASME Code, Section XI, Article IWE included inspections of the containment interior shell. In general, the areas and items subject to inspection include the accessible Class MC pressure-retaining containment surface areas, including structure attachments and penetrations, seals, gaskets, moisture barriers, pressure retaining bolts, and Class MC supports. By letter dated October 4, 2000, the NRC staff authorized relief requests MC-2 and MC-3 to the ASME Code, Section XI, requirements as explained in Section 3.1.3 below. NMC states that inaccessible areas are evaluated for degradation when conditions in accessible areas indicate the presence of degradation mechanisms that cause the accessible containment component to not meet established acceptable standards.

3.1.2 Implementing Subsection IWE-1240 at Monticello

In its supplemental letter of October 25, 2002, NMC states that the development of the original containment ISI plan determined that no areas of the Monticello containment required augmented examination per Subsection IWE-1240. The licensee states that there are several reason for this:

- Containment atmosphere is inerted with nitrogen during operation and is monitored routinely.

- Monticello has a well established and proven Nuclear Coating Program:

  — Coatings on the drywell and suppression chamber interiors are examined each refueling cycle by qualified personnel, and the suppression chamber exterior is required to be examined on a 5-year basis.

— Monticello has routinely drained the suppression chamber in the past and has used qualified personnel to inspect the coatings. These inspections have shown that no significant base metal corrosion has occurred on the interior of the suppression chamber.

— [Torus Inspection:] Following the 1997 Suction Strainer Replacement Outage in which the suppression chamber was drained and interior was examined by the qualified engineer, NRC Inspection Report 50-263/97011, dated August 26, 1997, states that "The MNGP [Monticello Nuclear Generating Plant] torus appeared to be maintained in an excellent manner."

- Previous degradation of the moisture barrier, which caused minor corrosion at the drywell shell to basemat interface, has been repaired to its original condition. . . .

- Other normally inaccessible areas were evaluated in the past; there was no indication of degradation.

3.1.3 IWE Table-2500-1, Examination Categories E-D and E-G for Seals and Gaskets, and Examination and Testing of Bolts

By letter dated October 4, 2000, the NRC staff authorized Relief Requests MC-2 and MC-3 for Examination Categories E-D and E-G. In its supplemental letter of October 25, 2002, NMC states that seals and gaskets for containment penetrations are tested in accordance with Option B of the 10 CFR Part 50, Appendix J, and:

> The initial test frequency for performing a leak test on seal, gaskets and bolts, which are Type B components, is at least once every 30 months. If two consecutive as-found Type B tests are less than their administrative limit, the test interval is extended to 60 months. If three consecutive as-found Type B tests are less than their administrative limit, the test interval is extended to 120 months. If a test result is greater than the administrative limit for the components, the component is restored to a leak rate below the administrative limit and the test interval is re-established at 30 months.
>
> Regardless of the schedule, any repair or disassembly of a component with a seal, gasket, or bolted connection requires a post-maintenance Appendix J, Type B test.
>
> Therefore, Monticello does not rely solely on Type A testing for seals, gaskets, or bolted connections.

Accordingly, the extension of the Type A testing does not affect this frequency. All penetrations utilizing gaskets and seals as part of the primary containment boundary are tested for leak-tight integrity within each 10-year inspection interval.

3.1.4 Integrity of Stainless Steel Bellows

In the past, the NRC staff has found that two-ply stainless steel bellows are susceptible to transgranular stress-corrosion cracking, and the leakage through them is not detectable by Type B testing (see NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," dated March 3, 1992). NMC states that the Monticello containment design includes a steel drywell and suppression chamber with interconnecting vent lines with bellows. The vent lines are welded to the drywell and to the vent-header down-comer assembly and pass through the suppression chamber penetration. The suppression chamber penetration, extended by the expansion bellows, is welded to the vent line. In this configuration, degradation of the suppression chamber penetration expansion bellows would not allow the drywell steam and air to bypass the suppression pool during a loss-of-coolant accident or core damage accident. The suppression pool chamber penetration expansion bellows are a design feature of the containment and consist of two single-ply bellows, in series. The suppression chamber penetration bellows are inspected in accordance with ASME Code, Section XI, Article IWE requirements, and tested during the Type A test.

3.1.5 Inspection of Embedded Side of the Containment Steel Shell

In its RAI of September 23, 2002, the NRC staff expressed concern to the licensee that inspections of reinforced concrete and steel containment structures at some nuclear facilities have identified degradation on the uninspectable (embedded) side of the containment steel shell of the primary containment. In its supplemental letter of October 25, 2002, NMC replied that Monticello personnel have taken aggressive efforts in the past to ensure that uninspectable areas of containment have retained their integrity and are not in a degraded state that would allow leakage through the containment boundary. Examples of these efforts include the following:

- Monticello maintains an inerted nitrogen atmosphere in containment during operation. The atmosphere is monitored routinely.

- The intact moisture barrier and drywell floor are designed to channel any water accumulation to the sump system to prevent corrosion at the basemat to floor interface.

- Qualified personnel at each refueling outage inspect the containment coatings and structures per approved procedures.

- A nuclear coatings program is in place to restore any degraded areas of coating that are evaluated to be in need of repair.

- The IWE Containment Inspection Program is in place and has completed the first period examinations.

- The successful results of previous ILRT's.are also on record.

3.1.6 Drywell Shell-to-Basemat Interface

In it's response to Generic Letter 87-05, "Request for Additional Information Assessment of Licensee Measures to Mitigate And/or Identify Potential Degradation of Mark I Drywells," NMC stated that there was minor corrosion identified at the drywell shell to drywell concrete floor interface. The minor corrosion was cleaned and repaired. In 1996, selected portions of the caulk (moisture barrier) were removed for inspection, and corrosion was discovered under the caulk. Seven areas of thickness reading were taken. Two readings were taken and recorded in each area. Testing results showed the most severe wall loss was only 0.057 inches. The drywell surface was repaired prior to the caulking replacement. In 1997, maintenance was performed to ensure the integrity of the drywell basemat-to-shell interface. A preservice visual inspection was performed on the moisture barrier. Examination reports are contained in the licensee's ASME Code, Section XI, Subsection IWE 1st Period Summary Report.

3.1.7 Sand Pockets

Monticello is designed with an air gap between the drywell vessel and the biological shield wall. There are three drainage paths for removing leakage that may result from refueling or from spillage of water into the drywell air gap. The first path prevents drywell refueling bellows leakage from entering the air gap. The second is at the drywell sand pocket interface where there is a galvanized steel plate, which is sealed to the drywell. The third pathway is from the sand pocket itself. During refueling operations at Monticello, sand pocket and air gap drains are inspected for signs of leakage after floodup.

All sealing materials between the refueling cavity and drywell air gap are steel pieces that are joined by watertight welds. A flow switch is provided on the drywell refueling bellows leakage drain line to detect leakage from the seal area.

The licensee states that the outlets of the sand pocket and air gap drains were inspected in 1987 and, with only one exception, were found to be unobstructed. The licensee believes that the obstruction was the result of drying of the sand pocket during construction and was not caused by leakage during plant operation.

Drywell shell thickness measurements were taken in 1987. Some minor interior corrosion was detected and was visible at the interface of the concrete floor and drywell shell. This area was repaired. The licensee found no thinning of the exterior shell.

3.1.8 Maintaining Positive Pressure in Containment

In response to the NRC staff's question on maintaining a positive pressure in the Monticello primary containment, the licensee stated that during power operation, the primary containment atmosphere is inerted with nitrogen to ensure that no external sources of oxygen are introduced into containment. The containment atmosphere control system provides a supply of makeup nitrogen to maintain primary containment oxygen concentration within TS limits. The primary containment is typically maintained at approximately 0.4 psig during power operation. Primary containment hi/lo pressure is annunciated in the control room. The primary containment pressure is graphically displayed on a control room recorder and monitored via the safety parameter display system (SPDP) critical plant variable displays. Trends can be produced on demand by using data from the process computer. The NRC staff finds that this continuous monitoring of slight positive pressure provides reasonable assurance that areas of containment degradation would be detected before they could result in large leakage rates.

3.1.9 Conclusion

On the basis of the above, the NRC staff finds that implementation of the licensee's containment ISI program, including the areas covered by augmented inspections, provides adequate assurance that the containment structural integrity will be maintained during the requested 5-year extension of the ILRT period.

Additionally, the NRC staff finds that NMC performs the following:

- Verifies the structural integrity of the containment vessel through the periodic ISI conducted as required by Subsections IWE and IWL of the ASME Code, Section XI.

- Periodically verifies the integrity of the penetrations, containment isolation valves, and mechanical bellows through Type B and Type C tests as required by 10 CFR Part 50, Appendix J and the Monticello TSs.

In addition, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities in accordance with Subarticle IWE-5000 of the ASME Code, Section XI. Significant degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 or 10 CFR 50.73.

The NRC staff concludes, on the basis of the considerations discussed above, that the licensee has adequate procedures to examine and monitor potential age-related and environmental degradations of the pressure-retaining components of Monticello's primary containment. Thus, the NRC staff finds that NMC's proposed one-time, 5-year extension for performing the ILRT, is acceptable.

3.2 Risk Impact Assessment

NMC's application of April 22, 2002, provided a risk impact assessment of extending the Type A test interval to 15 years. The licensee provided additional analyses and information in its supplemental letters of October 25, 2002, January 23, and February 12, 2003. In performing the risk assessment, NMC considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute's (EPRI's) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consists of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10-percent increase in the overall probability of leakage. The risk contribution of preexisting leakage for the pressurized-water reactor and boiling-water reactor representative plants confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 in 10 years to 1 in 20 years leads to an "imperceptible" increase in risk on the order of 0.2 percent and a fraction of one person-rem per year.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak was present. Since the Option B rulemaking in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6}/year and increases in large early release frequency (LERF) less than 10^{-7}/year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3-in-10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided an analysis which estimated all of these risk metrics and whose methodology is consistent with previously NRC-approved submittals. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 3-in-10-year test interval to a 1-in-15-year test interval, the increase in the total integrated plant risk is estimated to be 0.2 person-rem/year. This increase is comparable to that estimated in NUREG-1493, in which it was concluded that a reduction in the frequency of tests from 3 in 10 years to 1 in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 1.7×10^{-7}/year. However, there is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE visual examination of the containment surfaces (as identified in ASME Code, Section XI, Article IWE). The most recent visual inspection of the Monticello containment was performed in 2001. The next scheduled IWE containment inspection will be in May 2003. Visual inspections are expected to be effective in detecting large flaws in the visible regions of the containment, and would reduce the impact of the extended test interval on LERF. The licensee performed additional risk analysis to consider the potential impact of corrosion in inaccessible areas of the containment shell on the proposed change. The risk analysis considered the likelihood of an age-adjusted flaw that would lead to a breach of the containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type A ILRT. The increase in LERF associated with corrosion events is estimated to be less than 1×10^{-8}/year. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of 1.1 percentage points for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC staff finds that the defense-in-depth philosophy is

maintained based on the change in the conditional containment failure probability for the proposed amendment.

On the basis of the above, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (67 FR 56324). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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