October 2, 2006

Mr. Thomas J. Palmisano
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NOS. MC9272
AND MC9273)

Dear Mr. Palmisano:

The Commission has issued the enclosed Amendment No. 174 to Facility Operating License No. DPR-42 and Amendment No. 164 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 13, 2005, supplemented by letters dated June 7, and July 21, 2006.

The amendments revise technical specification (TS) 5.5.14 "Containment Leakage Rate Testing Program" for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2, to allow a one-time interval extension of no more than 5 years for the Appendix J Type A, Integrated Leakage Rate Test.

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/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 174 to DPR-42
2. Amendment No. 164 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

Mr. Thomas J. Palmisano
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

October 2, 2006

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NOS. MC9272 AND MC9273)

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The amendments revise technical specification (TS) 5.5.14 "Containment Leakage Rate Testing Program" for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2, to allow a one-time interval extension of no more than 5 years for the Appendix J Type A, Integrated Leakage Rate Test.

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Sincerely,

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Prairie Island Nuclear Generating Plant,
Units 1 and 2

cc:

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. DPR-42

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated December 13, 2005, supplemented by letters dated June 7, and July 21, 2006, 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-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 174, 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 its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Martin Murphy, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: October 2, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 174 AND 164

FACILITY OPERATING LICENSE NO. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Facility Operating License No. DPR-42 and DPR-60 with the attached revised pages. The changed area is identified by a marginal line.

REMOVE	INSERT
DPR-42, License Page 3	DPR-42, License Page 3
DPR-60, License Page 3	DPR-60, License Page 3

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
5.0-28	5.0-28

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NMC to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to transfer byproduct materials from other job sites owned by Northern States Power Company for the purpose of volume reduction and decontamination.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NMC is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 174, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NMC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 0, submitted by letter dated October 18, 2004.

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to transfer byproduct materials from other job sites owned by Northern States Power Company for the purposes of volume reduction and decontamination.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NMC is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 164, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NMC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: 'Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 0, submitted by letter dated October 18, 2004.

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.164
License No. DPR-60

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated December 13, 2005, supplemented by letters dated June 7, and July 21, 2006, 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-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 164, 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 its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Martin Murphy, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: October 2, 2006

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 164 TO FACILITY OPERATION LICENSE NO. DPR-60

NUCLEAR MANAGEMENT COMPANY, LLC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated December 13, 2005, supplemented by letters dated June 7, and July 21, 2006, the Nuclear Management Company, LLC (the licensee), requested changes to the Technical Specifications (TSs) for the Prairie Island Nuclear Generating Plant, Units 1 and 2. The proposed changes would revise TS 5.5.14 “Containment Leakage Rate Testing Program” for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2, to allow a one-time interval extension of no more than 5 years for the Appendix J, Type A, Integrated Leakage Rate Test (ILRT). The TS revision is based on the risk-informed approach developed using Regulatory Guide (RG) 1.174 (Ref. 5.2).

This evaluation addresses the ability of the licensee’s Inservice Inspection (ISI) program to ensure the leak-tight integrity of the containment, if the ILRT test interval is extended as proposed by the licensee.

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination, and did not expand the scope of the original *Federal Register* notice.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (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. Prairie Island TS 5.5.14, “Containment Leakage Rate Testing Program,” 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 RG 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995, with an exception which is noted in the TS. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995.

RG 1.163, Section C, “Regulatory Position” states, “licensees intending to comply with Option B of Appendix J in the amendment should establish test intervals based upon the criteria in ANSI/ANS-56.8-1994 (Ref. 5.6).” Section 11 of NEI 94-01 states that Type A testing shall be performed at a frequency of at least once per 10 years.

A Type A test is an overall (integrated) leakage rate test of the containment structure. 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 for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at Prairie Island have been successful, so the current interval requirement is 10 years.

The licensee's proposed TS change is an extension of the currently specified 10-year interval for ILRTs to a 15-year interval on a one-time basis. There are no changes to any Code, regulatory requirement, or acceptance criteria.

The licensee is requesting a change to TS 5.5.14, which would add an additional exception from the guidelines of RG 1.163 and NEI 94-01, Revision 0, regarding the Type A test interval. Specifically, the exception states that the first Unit 1 Type A test performed after December 1, 1997, shall be performed by December 1, 2012, and the first Unit 2 Type A test performed after March 7, 1997, shall be performed by March 7, 2012.

The local leakage rate tests (Type B and Type C tests), including their schedules, are not affected by this request.

3.0 TECHNICAL EVALUATION

3.1 Deterministic Evaluation

For each unit at Prairie Island, the primary containment consists of a freestanding carbon steel cylindrical pressure retaining shell with a hemispherical dome and ellipsoidal bottom. The containment has a number of access penetrations (equipment hatch, air-locks), other process piping and electrical penetrations. The leak tight integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests (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 (i.e. Type A tests). These tests are performed to verify the essentially leak-tight characteristics of the containments at the design-basis accident pressure. The last ILRTs for the PINGP primary containments were performed in December 1997, and March 1997, respectively. With the extension of the ILRT time interval, the licensee is committing to perform the next Type A tests prior to December 2012, for Unit 1, and prior to March 2012, for Unit 2. Because the ILRT, the LLRTs, and ISI of the containment, collectively ensure the leak-tightness and structural integrity of the containment, the staff normally requests information regarding the licensee's program for containment ISI and potential areas of weakness in the containment penetrations that may not be apparent in the risk assessment. In Section 4.2.3 of Exhibit A of the amendment request, the licensee provided a summary of the containment. A review of the Exhibit warranted certain clarifications and additional information. The staff's request for additional information (RAI), and the licensee's response are discussed below.

The licensee is using the 1992 Edition and the 1992 Addenda of Subsections IWE of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code) (Ref. 5.7), with certain approved relief from some Code requirements, for conducting the inservice inspection of PINGP's primary containments.

In RAI 2, the staff commented that for the examination of seals and gaskets, and the examination and testing of bolts associated with the primary containment pressure boundary (Examination Categories E-D, and E-G), the licensee had requested relief from the requirements of the Code (Section 4.2.4.3 of Exhibit A of Ref. 5.7). As an alternative, the licensee committed in 2001 (see Section 4.2.3.1, Exhibit A. Ref. 5.1) to examine them during the leak rate testing of the primary containment. With the flexibility provided in Option B of 10 CFR 50, Appendix J for Type B and Type C testing (per NEI 94-01 and RG 1.163), and the extension requested in this amendment for the Type A testing interval, the licensee was requested to provide an examination schedule for examination and testing of seals, gaskets, and bolts (pressure seating and pressure unseating penetrations) that provides periodic assurance regarding the integrity of the containment pressure boundary.

In Enclosure 1 of the response letter dated June 7, 2006 (Ref. 5.8), the licensee provided two tables and explained that Tables 1 and 2 provide the current frequency and schedule for the next LLRT for each containment penetration containing seals, gaskets or bolts for PINGP Units 1 and 2, respectively. The tables also provide the date when a visual inspection of the bolted joints was or will be completed for each containment penetration that includes seals, gaskets or bolts associated with the primary containment pressure boundary. The licensee, furthermore, notes that the current interval for the IWE program will end on September 9, 2008. At that time, the IWE program will be updated to the latest approved Edition and Addenda of ASME Section XI, 12 months prior to the end of the interval as specified in 10 CFR 50.55a(g)(4)(ii).

A review of the tables indicates that in Unit 1, out of 23 penetrations having seals, gaskets or bolts, 14 penetrations will go through LLRTs every refueling outage. The airlocks, equipment hatch, and fuel transfer tube (representing large penetrations) will be tested every refueling outage. Nine other penetrations (relatively small) will be leak rate tested at 120 months, as permitted by NEI 94-01 and RG 1.163. The same LLRT frequency trend is shown for Unit 2 in Table 2.

The staff considers the licensee’s approach in ensuring the leak tight integrity of the containment penetrations with seals, gaskets, and bolts, reasonable and acceptable. Thus, the concern expressed in this RAI is resolved.

In RAI 3, the staff noted that based on the review of Section 4.2.3.2 of Exhibit A of Reference 5.1, the staff understands that the licensee is using the 1992 Edition and the 1992 Addenda of Subsection IWE of the ASME Section XI Code for the examination of the containment steel shell. Section 4.2.3.6 indicates that VT-1 examinations are performed for areas accessible from both sides, and ultrasonic thickness measurements are performed for surface areas accessible from only one side. The staff requested the licensee provide the locations of the containment surfaces where the licensee has identified measurable degradation (other than coating irregularities), and a summary of findings of the examinations performed.

In Enclosure 1 of the response letter dated June 7, 2006, the licensee pointed out that degradation of the liner has been identified at PINGP. The areas of concern where measurable degradation was found are discussed in Exhibit A of the License Amendment Request (LAR) dated December 13, 2005. Section 4.2.4.2 identified an area where measurable degradation has been observed (other than coating irregularities), provided a summary of examination findings and stated the status of any subsequent examinations. Sections 4.2.4.3 and 4.2.4.4 identified areas where surface degradation has been observed but was not measurable.

The staff agrees with the licensee’s response that Section 4.2.4 of Exhibit A of the LAR identifies the areas of degradation found during inspections. Section 4.2.4 identifies six areas of degradation. Based on the inspection findings indicated and corrective actions taken, the staff finds the licensee’s actions and judgement regarding the degradation noted in Sections 4.2.4.1, 4.2.4.2, 4.2.4.4, 4.2.4.5, and 4.2.4.6 reasonable and acceptable. However, the staff needed clarification of the degradation noted in Section 4.2.4.3.

RAI 3A In discussion of degradation in Section 4.2.4.3 of the LAR, the licensee makes a judgement: “Because the corrosion is obviously due to the lack of paint on the vessel wall in these areas and moisture from the containment atmosphere, no degradation below the moisture (barrier), which is inaccessible, is suspected.” The staff experience with corrosion in the inaccessible areas indicates that if the moisture barrier were degraded, the inaccessible area has to be identified as a “suspect area,” per IWE-1241(a). The staff requests the licensee to provide information regarding the condition of the moisture barrier in this area for both PINGP units. If the moisture barrier was found degraded, provide justification for not characterizing the inaccessible area(s) as “suspect areas.”

In a letter dated July 21, 2006 (Ref. 5.9), the licensee indicated that its administrative process requires a 100 percent inspection of the containment moisture barrier each refueling outage. The licensee, moreover, added that the moisture barrier is in good condition on both the inside and outside of the containment structure and it tightly adheres to both the steel and concrete in all locations.

The staff finds the licensee’s response to RAI 3A acceptable. The process used for examination, and the condition indicated in the response, ensures that the liner below the moisture barrier (i.e., the inaccessible) is not subjected to corrosion. Based on the response, the staff’s concern identified in RAI 3A is resolved.

In RAI 4, the staff noted that Section 4.2.3.3 of Exhibit A (Ref. 5.1) indicates a number of areas exempt from the ISI examinations. Inspections of some steel containments have indicated degradation from the uninspectable (embedded) side of the primary containments. These degradations cannot be found by VT-3 or VT-1 examinations unless they are through the thickness of the shell. For the ellipsoidal bottom of the containment, it is not feasible to examine any part of it. The staff asked the licensee to provide information as to how potential leakages due to aging-related degradation of the primary containment areas, exempted from the ISI examinations, are factored into the risk assessment related to the ILRT interval extension.

In a response letter dated June 7, 2006, the applicant explained that Exhibit D of the LAR dated December 13, 2005, Section 5.1, first paragraph of "Class 3 Sequences" discussion on page 32 of 102, discusses how potential leakages due to aging-related degradation of the primary containment areas, exempted from the ISI examinations, are factored into the risk assessment related to the ILRT interval extension. The baseline frequency for Class 3b sequences (pre-existing leakage in the containment structure) was increased by a factor of 1.000269 (0.0269 percent increase for the 3 to 15 year extension as calculated in Appendix A to Exhibit D). However, the number was incorrectly stated as 1.00269 instead of 1.000269. The number is correctly listed elsewhere in the document (for example, page 40 of 102). Tables 5-2 and 5-4 show the Electric Power Research Institute (EPRI) Class 3b frequency without the corrosion factor included.

Moreover, the applicant notes that the actual application of the corrosion factor to Class 3b is presented in Section 5.3, where the process is reiterated with a little more detail under the "Risk Impact of Corrosion-Related Leakage on Increase to 15-year Test Interval" discussion (page 40 of 102). Here the discussion states incorrectly that the corrosion factor was applied to the Class 7 sequences, when in reality, it was applied to Class 3b sequences. Table 5-5 shows the Class 3b accident class frequencies after they are increased by the correct corrosion factor (factor of 1.000269) and the 10-year interval extension failure to detect probability (factor of 1.1). The applicant pointed out the following corrections to Appendix A to Exhibit D.

- The incorrect statement that the factor was applied to the Class 7 sequences reappears on page 66 of 102, where the application of the NEI Methodology is discussed. The factors and tables are correct; the text should have stated "Class 3b" instead of "Class 7";

- Page 32 of 102: The corrosion factor value listed should have been 1.000269 instead of 1.00269;

- Page 40 of 102: "Risk Impact of Corrosion-Related Leakage on Increase to 15-year Test Interval" first paragraph, second to last sentence should be changed to read, "The increased likelihood of corrosion related leakage is assumed to increase large early release frequency (LERF) frequency contributions from containment structure pre-existing leakage (EPRI Class 3b) …"; and

- Page 66 of 102: Second paragraph reference to "Class 7" should have been to "Class 3b."

The adequacy of this response is evaluated in Item 2, under probalistic risk assessment (Section 3.2) of this safety evaluation.

Considering the performance of the primary containment components during earlier leakage rate tests, implementation of programs and procedures for monitoring degradation in the primary containment components, and treatment of the uninspectable areas of the primary containment structure in the risk-informed analysis, the staff believes that licensee has appropriately addressed the potential areas of concern that could affect the leak-tight integrity of the primary containments.

<u>Summary</u>

Based on the licensee's procedures related to the potential degradation of the pressure retaining primary containment components, the staff finds that granting the requested ILRT extension will not adversely affect the leak-tight integrity of the primary containment. It should be noted that Subarticle IWE-5000 of the ASME Code, Section XI, requires leak rate testing following a major repair, modification, or replacement of containment components. An ILRT might be required to confirm that these activities are adequate and that further degradation does not exist in other areas of the containment. The licensee is required to report serious degradation of the containment pressure boundary pursuant to 10 CFR 50.72 or 10 CFR 50.73.

3.2 Probabilistic Risk Assessment

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the December 13, 2005, application for license amendment. Additional analysis and information was provided by the licensee in its letters dated June 7, and July 21, 2006. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in EPRI 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 the development of the performance-based Option B to 10 CFR 50, Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to 10 CFR 50, Appendix J. The basis consisted 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 this basis, industry 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 10 CFR 50, Appendix J, Option A, requirements that were in effect for Prairie Island early in the plant's life, required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years, would increase the average time that a leak that was 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 pre-existing leakage for the pressurized-water reactor and boiling-water reactor representative plants in the EPRI study, confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years, leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in increased public dose.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking was completed in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance 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} per year and increases in LERF less than 10^{-7} per 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 frequency of three tests in a 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 analyses, as discussed below. The following comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under 10 CFR 50, Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be less than 0.01 person-rem per year. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one 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 frequency from the original three in 10 years to one in 15 years is estimated to be about 1.4×10^{-7} per year based on the internal events PRA, and 7.7×10^{-7} per year when external events (fire and seismic) are included. 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/IWL visual examination of the containment surfaces (as identified in ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE/IWL). Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 1×10^{-8} per year.

   When the calculated increase in LERF is in the range of 10^{-7} per year to 10^{-6} per year, applications are considered if the total LERF is less than 10^{-5} per year. The licensee estimates that the total LERF for internal and external events, including the requested change, is about 7×10^{-6} per year, which meets the total LERF criteria. The 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 between 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 approximately one percentage point for the cumulative change of going from a test frequency of three in 10 years to one in 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Summary

Based on the foregoing evaluation, the staff finds that the interval until the next Type A test at each unit of the Prairie Island Nuclear Generating Plant may be extended to 15 years, and that the proposed change to TS 5.5.14 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 REFERENCES

5.1 Letter from T. Pulmisano (Nuclear Management Company) to NRC, "License Amendment Request to Technical Specification 5.5.14 for One-time Extension of Containment Integrated Leak Rate Test Interval," December 13, 2005.

5.2 USNRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, July 1998."

5.3 10 CFR, Part 50, Appendix J, Option B, "Performance-Based Leakage-Test Requirements."

5.4 USNRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

5.5 Nuclear Energy Institute Document, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 1995.

5.6 ANSI/ANS-56.8, "Containment System Leakage Testing Requirements," American Nuclear Society, La Grange Park, IL, 1994.

5.7 ASME Boiler and Pressure Vessel Code, Section XI, 1992 Edition including 1992 Addenda.

5.8 Letter from T. Pulmisano (Nuclear Management Company) to NRC, "Supplement to License Amendment Request to Technical Specification 5.5.14 for One-time Extension of Containment Integrated Leak Rate Test Interval," June 7, 2006.

5.9 Letter from T. Pulmisano (Nuclear Management Company) to NRC, "Supplement to License Amendment Request to Technical Specification 5.5.14 for One-time Extension of Containment Integrated Leak Rate Test Interval," July 21, 2006.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 (71 FR 5081). 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.

7.0 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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