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L-PI-04-042
10 CFR 50.90
10 CFR 50.54
10 CFR 50 Appendix J

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKET 50-282
LICENSE No. DPR-42

**SUPPLEMENT TO LICENSE AMENDMENT REQUEST DATED AUGUST 27, 2003,
"EXCEPTION TO TECHNICAL SPECIFICATION 5.5.14 TESTING REQUIREMENTS
ASSOCIATED WITH STEAM GENERATOR REPLACEMENT"**

In Nuclear Management Company, LLC (NMC) letter L-PI-03-46 dated August 27, 2003, and supplemented by letter L-PI-03-103 dated December 16, 2003, NMC requested an amendment to Appendix A of the Operating License for Prairie Island Nuclear Generating Plant (PINGP) Unit 1 that would except PINGP Unit 1 from the requirements of Regulatory Guide 1.163 as specified in the Technical Specifications for post-modification containment leakage rate testing associated with steam generator replacement. The technical analysis for not performing an integrated leak rate test (ILRT) presented in Exhibit A of the December 16, 2003 letter was based in part on the pressure test requirements of the American Society of Mechanical Engineers (ASME) Code Section III. In letter L-PI-04-036 dated March 19, 2004, NMC submitted a request for relief from the ASME Code Section III requirements for a secondary (shell) side overpressure hydrostatic test of the Unit 1 replacement steam generators. As a consequence of this relief request, the technical analysis presented in the December 16, 2003 letter is being revised to remove reference to ASME Section III. The technical analysis now relies on the Section XI pressure test requirements that will still require a system test pressure at least 20 times that of an ILRT test pressure.

Exhibit A contains the revised licensee evaluation reflecting the changes as indicated by the side bars. There are no changes to the Exhibits B, C and D of the December 16, 2003 letter.

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In summary, this exception is being requested to avoid performing an unnecessary ILRT. As discussed in the attached Exhibit A, the ILRT is unnecessary because the ASME Code Section XI pressure test requirements for the replacement steam generators will satisfy the intent of Regulatory Guide 1.163.

Based on the discussion in the attached Exhibit A, NMC concludes that this supplement to the August 27, 2003 proposed amendment, (as supplemented by the December 16, 2003 letter) does not affect the conclusions of the previous no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

NMC requests approval of the proposed amendment by August 26, 2004.

This license amendment request (LAR) contains no new commitments and no revisions to existing commitments.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this supplement to the August 27, 2003 LAR by transmitting a copy of this letter and attachment to the designated State Official.

Please address any comments or questions regarding this LAR to Mr. H Oley Nelson at 1-651-388-1121.

I declare under penalty of perjury that the foregoing is true and accurate.

Executed on **MAR 22 2004**


Joseph M. Solymessy
Site Vice President, Prairie Island Nuclear Generating Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector - Prairie Island Nuclear Generating Plant
Glenn Wilson, State of Minnesota

Attachment:

Exhibit A, Licensee Evaluation

Exhibit A
Letter L-PI-04-042

LICENSEE EVALUATION

Subject: Supplement to License Amendment Request dated August 27, 2003, "Exception to Technical Specification 5.5.14 Testing Requirements Associated with Steam Generator Replacement"

1.0 DESCRIPTION

This letter is a request to amend the Operating License DPR-42 for Prairie Island Nuclear Generating Plant (PINGP) Unit 1.

The proposed change would revise Appendix A of the Operating License to except PINGP Unit 1 from the requirements of Regulatory Guide 1.163 as specified in the Technical Specifications for post-modification containment leakage rate testing associated with steam generator replacement. This exception is being requested so that the American Society of Mechanical Engineers Code (ASME) Section XI pressure test requirements may be used to satisfy the intent of the Regulatory Guide 1.163 requirements rather than performing a Type A test, i.e. containment integrated leak rate test (ILRT). To accomplish this, the Nuclear Management Company, LLC (NMC) is requesting that this license amendment request (LAR) be approved prior to the steam generator replacement currently scheduled for the Fall of 2004.

2.0 PROPOSED CHANGE

A brief description of the proposed change is provided below along with a discussion of the justification for the change. The specific wording changes to the Technical Specification (TS) are provided in Exhibits B and C of letter L-PI-03-103 dated December 16, 2003.

PINGP Technical Specification (TS) 5.5.14.a states:

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory

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

Regulatory Guide 1.163 (Reference 1) endorses Nuclear Energy Institute (NEI) 94-01, Revision 0 (Reference 2) for methods acceptable to comply with the requirements of Option B. Prior to returning the primary containment system to operation, NEI 94-01 requires leakage rate testing following repairs and modification that affect the containment leakage integrity.

The proposed amendment would except PINGP Unit 1 from the requirements of Regulatory Guide 1.163 as specified in the Technical Specifications for post-modification leakage rate testing associated with the replacement of the steam generators. This would be accomplished by adding a requirement to PINGP TS 5.5.14 that clearly states that there is an exception to the post-modification containment leakage testing requirements associated with replacement of the Unit 1 steam generators. The proposed revision to the PINGP TS 5.5.14 is shown in Exhibits B and C of letter L-PI-03-103 dated December 16, 2003.

In summary this LAR will provide an exception to the post-modification containment leakage testing requirements of Regulatory Guide 1.163 as specified in the Technical Specifications associated with the replacement of Unit 1 steam generators.

3.0 BACKGROUND

PINGP is a dual unit site. Each unit is a two-loop 1650 MWt Westinghouse design. The NMC will replace the Unit 1 original Westinghouse Model 51 steam generators that have been in service since commercial operation was achieved in 1973. The NMC is currently preparing to replace the Unit 1 Westinghouse steam generators with steam generators fabricated by Framatome ANP during an outage in the Fall of 2004.

Each replacement steam generator (RSG) consists of a new lower subassembly and new upper subassembly, the final assembly of which will be performed within the Unit 1 containment during the Fall 2004 outage. The RSGs will occupy the same physical envelope as the original steam generators (OSGs). There are no changes to interfaces with the reactor coolant, main steam, feedwater, or

auxiliary feedwater systems. The piping attaching these systems to the OSGs will be cut and welded back to the RSGs after they are installed. The Unit 1 reactor containment vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom which houses the reactor pressure vessel, the steam generators, reactor coolant pumps, the reactor coolant loops, the accumulators of the safety injection system, the primary coolant pressurizer, the pressurizer relief tank, and other branch connections of the reactor coolant system. The reactor containment vessel is, in turn, housed completely within the shield building. Since the rigging and handling necessary to perform the Unit 1 steam generator replacements are designed to use the equipment hatch that services the reactor containment vessel, no alteration or modification of the reactor containment vessel structure will be required. For the same reason, no modifications to the structure of the Unit 1 shield building will be required to achieve the access to the equipment hatch for performing the rigging and handling of the steam generators. Thus, there are no structural effects to the reactor containment vessel resulting from the steam generator replacement activities.

Although the steam generators are not part of the reactor containment vessel, during a design basis loss of coolant accident (LOCA) portions of them are relied upon to act as a barrier against the uncontrolled release of radioactivity to the environment. Thus the outer shell of the steam generators, the inside containment portions of the main steam lines, the main and auxiliary feedwater lines, the steam generator blowdown lines, the steam generator water level instrument lines, the steam generator tubes, and the steam generator tube sheets are all considered part of the primary containment system boundary. All of these components will be impacted by the steam generator replacement activities. Thus, replacing the steam generators will constitute a modification to the primary containment system boundary.

PINGP's TS 5.5.14.a requires that a program be established to implement the leakage testing of the containment as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Regulatory Guide 1.163 (Reference 1) endorses NEI 94-01, Revision 0 (Reference 2) for methods acceptable to comply with the requirements of Option B. Section 9.2.4 of NEI 94-01 requires that a Type A or local leakage rate testing be conducted

prior to returning the primary containment system to operation following a modification that affects the containment leakage integrity. As stated above, replacing the steam generators will constitute a modification to the primary containment system boundary and thus affects the containment leakage integrity. As discussed in Section 4 below, performing local leakage rate testing for this modification is not practical. Therefore, to satisfy TS 5.5.14.a, a Type A (i.e. an ILRT) test would have to be performed. Since the next ILRT for Unit 1 is not scheduled to occur until 2007, an additional ILRT would have to be performed unless an exception to the requirement is obtained.

This exception is requested to avoid performing an unnecessary ILRT. As discussed below, the ILRT is unnecessary because the ASME Section XI pressure test requirements for the replacement steam generators will satisfy the intent of the Regulatory Guide 1.163 and NEI 94-01 requirements.

This exception is similar to that granted to Calvert Cliffs Nuclear Power Plant, Unit No. 2 in reference 3.

4.0 TECHNICAL ANALYSIS

The PINGP Unit 1 plant design incorporates a closed system for transferring steam from the steam generators inside of the primary containment to the main turbine generator in the turbine building. The inside containment portion of this closed system consists of the outer shell of the steam generators, the main steam lines, the main and auxiliary feedwater lines, the steam generator blowdown lines, the steam generator water level instrument lines, the steam generator tubes, and the steam generator tube sheets. During a design basis LOCA these elements inside containment form a barrier against the uncontrolled release of radioactivity to the environment and thus are considered part of the primary containment system boundary.

The planned replacement of the PINGP Unit 1 steam generators includes the following activities:

- Cutting and removing the main steam lines, main and auxiliary feedwater lines, steam generator blowdown lines, steam generator water level instrument lines.
- Cutting and removing the upper assemblies of the steam generators.

• Cutting the reactor coolant piping and removing the steam generator lower
 subassemblies and re-welding the main steam lines, main and auxiliary
 feedwater lines, steam generator blowdown lines, and steam generator
 water level instrument lines.

• Installing the new steam generator upper subassemblies on the new lower
 subassemblies.

• Re-installing and re-welding the main steam lines, main and auxiliary
 feedwater lines, steam generator blowdown lines, and steam generator
 water level instrument lines.

The planned replacement of the Unit 1 steam generators affects only these
 closed piping systems inside the reactor containment vessel. The steam
 generator replacement activities do not affect the reactor containment vessel
 structure or the structure of the shield building.

NEI 94-01 requires integrated leakage testing (Type A) or local leakage rate
 testing (Type B or Type C) prior to returning the primary containment system to
 operation following repairs and modification that affect the containment leakage
 integrity. The Type C testing requirements apply to leakage testing of
 containment isolation valves. The planned replacement does not affect any
 containment isolation valves, and therefore the Type C testing requirements are
 not applicable. The Type B testing requirements apply to leakage testing of
 gasketed or sealed containment penetrations (e.g., electrical penetrations), air
 lock door seals, and other doors with resilient seals or gaskets. Although the
 secondary side of the steam generators has access manways and handhole
 ports with gaskets, it is impractical to perform a Type B test for these items.
 Hence, since Type B or Type C testing cannot test all the affected areas,
 NEI 94-01 would require that a Type A test be performed prior to startup following
 the planned steam generator replacement. Type A test measures the primary
 containment system overall integrated leakage rate under conditions
 representing design basis accident containment pressure and system alignment.

However, for preservice and inservice inspection requirements the affected area
 of the primary containment system boundary is classified as ASME Class 2 per
 Section XI. As such the replacement of the steam generators is subject to the
 pressure test requirements of ASME Section XI. The acceptance criteria for
 ASME Section XI system pressure testing for the base metal and welds is no
 leakage. The testing will also show that the access manways and handholes will
 meet their current leakage requirements. Since the base metal and welds are not

allowed to leak and the access manways and handholes will meet their current leakage requirements, the ASME Section XI pressure test requirements are more stringent than the Type A testing requirements. In addition, the test pressure for the system pressure test will be at least 20 times that of a Type A test.

The intent of performing a Type A test is to assure the leak-tight integrity of the area affected by the modification (i.e., the closed system inside the reactor containment vessel formed by the outer shell of the steam generators and the main steam, feedwater, steam generator blowdown, feedwater piping, steam generator tubes, and steam generator tube sheets) does not alter the overall leakage rate of the primary containment. Although the leak test is in a direction reverse that of a LOCA environment, the leak tightness of the components, piping, and welds is not dependent on the direction the pressure is applied. Thus, the ASME Section XI inspection and testing requirements more than fulfill the intent of the requirements of Regulatory Guide 1.163 and NEI 94-01. Likewise the post installation testing of the steam generator instrument lines will be in the direction reverse that of a LOCA environment and will show that the lines meet their current leakage requirements. This also fulfills the intent of the requirements of Regulatory Guide 1.163 and NEI 94-01 since the leak tightness of the fittings in the instrument lines is dependant on the mechanical makeup of the fitting and not the direction of the pressure being applied.

Therefore, the NMC proposes a revision to TS 5.5.14 to except Unit 1 from the requirements of Regulatory Guide 1.163 as specified in the Technical Specifications for post-modification integrated leakage rate testing associated with replacement of the steam generators. The effect of this amendment request would be to eliminate the post-modification containment leakage rate (Type A) testing required for the modifications to the primary containment system boundary specifically associated with replacement of the steam generators.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The Nuclear Management Company, LLC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

The change that is being evaluated below is the addition of a requirement to the Technical Specification that provides an exception for Unit 1 from post-modification integrated leak rate test requirements associated with replacement of the steam generators.

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No. The Technical Specification requirement for integrated leak rate testing is not a direct initiator of an accident.

The proposed change would provide the Prairie Island Nuclear Generating Plant an exception from performing a required containment integrated leak rate test following the replacement of the steam generators in Unit 1.

Integrated leak rate tests are performed to assure the leak-tightness of the primary containment boundary system, and as such they are not accident initiators. Therefore, not performing an integrated leak rate test will not affect the probability of an accident previously evaluated.

The intent of post-modification integrated leak rate testing requirements is to assure the leak-tight integrity of the area affected by the modification. For the Unit 1 steam generator replacement modification, this intent will be satisfied by performing the American Society of Mechanical Engineers code required inspections and tests. Since the leak-tightness integrity of the primary containment boundary affected by replacement of the steam generators will be assured, there is no change in the primary containment boundary's ability to confine radioactive materials during an accident.

Therefore adding a Technical Specification requirement that provides an exception for Unit 1 from the steam generator replacement post-modification integrated leak rate testing requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No. The integrated leak rate testing requirements are not a direct initiator of an accident.

The proposed change would provide the Prairie Island Nuclear Generating Plant an exception from performing a required containment integrated leak rate test following the replacement of the steam generators in Unit 1.

Providing an exception from performing a test does not involve a physical change to the plant nor does it change the operation of the plant. Thus it cannot introduce a new failure mode.

Therefore adding a Technical Specification requirement that provides an exception for Unit 1 from the steam generator replacement post-modification integrated leak rate testing requirements does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed change would provide the Prairie Island Nuclear Generating Plant an exception from performing a required containment integrated leak rate test following the replacement of the steam generators in Unit 1.

The intent of post-modification integrated leak rate testing requirements is to assure the leak-tight integrity of the area affected by the modification. This intent will be satisfied by performing American Society of Mechanical Engineers code required inspections and tests. The acceptance criterion for American Society of Mechanical Engineers code system pressure testing for the base metal and welds is no leakage. In addition, the test pressure for the system pressure test will be several times that required during an integrated leak rate test. Since the leak-tight integrity of the primary containment boundary affected by replacement of the steam generators will be assured, there is no change in the primary containment boundary's ability to confine radioactive materials during an accident.

Therefore, adding a Technical Specification requirement that provides an exception for Unit 1 from the steam generator replacement post-modification integrated leak rate testing requirements does not involve a

significant reduction in a margin of safety.

Based on the above, the Nuclear Management Company, LLC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

PINGP Technical Specification 5.5.14 states:

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," endorses NEI 94-01, Revision 0 for methods acceptable to comply with the requirements of Option B. Prior to returning the primary containment system to operation, NEI 94-01 requires leakage rate testing following repairs and modification that affect the containment leakage integrity.

For preservice and inservice inspection requirements the affected area of the primary containment system boundary is classified as American Society of Mechanical Engineers code Class 2 per Section XI. As such the replacement of the steam generators is subject to the pressure test requirements of American Society of Mechanical Engineers code Section XI. The acceptance criteria for American Society of Mechanical Engineers code Section XI system pressure testing for the base metal and welds is no leakage. Thus American Society of Mechanical Engineers code Section XI pressure test requirements are more stringent than the Regulatory Guide 1.163 and NEI 94-01 testing requirements. In addition, the test pressure for the system pressure test will be several times higher than that required for a Regulatory Guide 1.163 and NEI 94-01 test.

The American Society of Mechanical Engineers code Section XI inspection and testing requirements more than fulfill the intent of the requirements of Regulatory Guide 1.163 as specified in the Technical Specifications. Since the leak-tight

integrity of the primary containment boundary affected by replacement of the steam generators will be assured, there is no change in the primary containment boundary's ability to fulfill its design function.

In conclusion, based on the considerations discussed above, (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.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22 (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.
2. Nuclear Energy Institute 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, dated July 26, 1995, including Errata.
3. Letter from Donna Skay, NRC to P. E. Katz, Calvert Cliffs Nuclear Power Plant, dated June 27, 2002; Subject: "Calvert Cliffs Nuclear Power plant, Unit No. 2 – amendment RE: Exception to Post-Modification Integrated Leakage Rate Testing (TAC NO. MB3444)".