NMC
Committed to Nuclear Excellence

Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

December 16 , 2003

L-PI-03- 103
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, the NMC requested an amendment to Appendix A of the Operating License for the Prairie Island Nuclear Generating Plant (PINGP) Unit 1 that would except PINGP Unit 1 from the requirements of 10 CFR 50, Appendix J, Option B for post-modification leakage rate testing associated with the steam generator replacement currently scheduled for the Fall of 2004. This letter revises that request 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 letter also revises the Technical Specification (TS) pages so that they are more consistent with the improved Standard Technical Specifications in NUREG 1431. Although the technical arguments presented in Exhibit A of the August 27, 2003 letter have not changed, they are repeated in the attached Exhibit A along with the changes (as indicated by the side bars) necessary to clarify what is being requested.

As stated in the August 27, 2003 letter, this exception is being requested to avoid performing an unnecessary integrated leak rate test (ILRT). As discussed in the August 27, 2003 letter and in Exhibit A, the ILRT is unnecessary because the American Society of Mechanical Engineers Code Sections III/XI pressure test requirements for the replacement steam generators will satisfy the intent of Regulatory Guide 1.163.

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Based on the discussion in the attached Exhibit A, the NMC 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.

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

Exhibit A contains the licensee's evaluation of this proposed change. Exhibit B presents the proposed TS pages mark-up. Exhibit C presents the revised TS pages incorporating the proposed changes. Exhibit D provides the commitments made in this LAR.

In accordance with 10 CFR 50.91, the NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and attachments 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 December 16, 2003

Joseph M. Solymossy
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

Attachments:

Exhibit A, Licensee Evaluation
Exhibit B, Proposed Technical Specification Changes (mark up)
Exhibit C, Revised Technical Specification Changes
Exhibit D, List of Commitments

Exhibit A
Letter L-PI-03-103

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 III/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.

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 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 on the marked-up page in Exhibit B and C.

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 III/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 assemblies.
- Installing the new steam generator lower subassemblies and re-welding the reactor coolant piping.

- Installing the new steam generator upper subassemblies on the new lower assemblies.
- 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. The pressure boundary of the RSGs is constructed in accordance with ASME Section III Class 1. As such the replacement of the steam generators is subject to the requirements of ASME Sections III and XI. The acceptance criteria for ASME Section III/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 III/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 III/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 <u>No Significant Hazards Consideration</u>

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 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 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. The pressure boundary of the replacement steam generators are constructed in accordance with American Society of Mechanical Engineers code Section III Class 1. As such the replacement of the steam generators is subject to the requirements of American Society of Mechanical Engineers code Sections III and XI. The acceptance criteria for American Society of Mechanical Engineers code Section III/XI system pressure testing for the base metal and welds is no leakage. Thus American Society of Mechanical Engineers code Section III/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 III/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)".

Exhibit B
Letter L-PI-03-103

Proposed Technical Specification Changes (mark-up)
(Additions shaded, deletions strikethrough)

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

Technical Specification pages

5.0-27 through 5.0-29

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5 Programs and Manuals (continued)

5.5.14 Containment Leakage Rate Testing Program

a. 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~~;~~ ~~as modified by the following exception:~~.

~~1. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.~~

~~5.5.14 Containment Leakage Rate Testing Program (continued)~~

b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.1% of primary containment air weight per day at pressure P_a. For pipes connected to ystems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01% of primary containment air weight per day at pressure P_a.

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

1. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to unit startup, following testing in accordance with the program, the combined leakage rate acceptance criteria are $\leq 0.60\ L_a$ for all components subject to Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at ≥ 46 psig.

b) For each door intergasket test, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

~~5.5.14 Containment Leakage Rate Testing Program (continued)~~

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.15 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance of the 125V plant safeguards batteries and service building batteries, which may be used instead of the safeguards batteries during shutdown conditions in accordance with manufacturer's recommendations, as follows:

a. Actions to restore battery cells with float voltage < 2.13 V will be in accordance with manufacturer's recommendations, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

Exhibit C
Letter L-PI-03-103

Revised Technical Specification Changes

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

Technical Specification pages

5.0-27 through 5.0-29

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5 Programs and Manuals (continued)

5.5.14 Containment Leakage Rate Testing Program

a. 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, as modified by the following exception:

   1. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.1% of primary containment air weight per day at pressure P_a. For pipes connected to ystems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01% of primary containment air weight per day at pressure P_a.

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

1. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to unit startup, following testing in accordance with the program, the combined leakage rate acceptance criteria are $\leq 0.60\ L_a$ for all components subject to Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at ≥ 46 psig.

b) For each door intergasket test, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.15 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance of the 125V plant safeguards batteries and service building batteries, which may be used instead of the safeguards batteries during shutdown conditions in accordance with manufacturer's recommendations, as follows:

a. Actions to restore battery cells with float voltage < 2.13 V will be in accordance with manufacturer's recommendations, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

Exhibit D
Letter L-PI-03-103

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

LIST OF COMMITMENTS

The following table identifies those actions committed to by Nuclear Management Company, LLC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments.

REGULATORY COMMITMENT	DUE DATE
None	