January 20, 2006

Mr. T. 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 EVENT FOLLOW-UP INSPECTION REPORT 05000282/2005012(DRS); 05000306/2005012(DRS)

Dear Mr. Palmisano:

On December 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Prairie Island Nuclear Generating Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on January 5, 2006, with you and members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one self-revealed finding of very low safety significance which involved violations of NRC requirements was identified. However, because this violation was of very low safety significance and because the issue was entered into the licensee's corrective program, the NRC is treating this finding and issue as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, a licensee identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Prairie Island Nuclear Generating Plant facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room, or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Julio F. Lara, Chief
Engineering Branch 3
Division of Reactor Safety

Docket Nos. 50-282; 50-306
License Nos. DPR-42; DPR-60

Enclosure: Inspection Report No. 05000282/2005012(DRS); 05000306/2005012(DRS)
w/Attachment: Supplemental Information

cc w/encl: C. Anderson, Senior Vice President, Group Operations
J. Cowan, Executive Vice President and Chief Nuclear Officer
Regulatory Affairs Manager
J. Rogoff, Vice President, Counsel & Secretary
Nuclear Asset Manager
Tribal Council, Prairie Island Indian Community
Administrator, Goodhue County Courthouse
Commissioner, Minnesota Department of Commerce
Manager, Environmental Protection Division Office of the Attorney General of Minnesota

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Office of the Attorney General of Minnesota

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-282; 50-306
License Nos:	DPR-42; DPR-60
Report No:	05000282/2005012(DRS); 05000306/2005012(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Prairie Island Nuclear Generating Plant, Units 1 and 2
Location:	1717 Wakonade Drive East Welch, MN 55089
Dates:	October 27 through December 30, 2005
Inspectors:	R. Langstaff, Senior Reactor Inspector A. Klett, Reactor Inspector
Approved by:	J. Lara, Chief Engineering Branch 3 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000282/2005012(DRS); 05000306/2005012(DRS); 10/27/2005 - 12/30/2005; Prairie Island Nuclear Generating Plant, Units 1 & 2; Event Follow-Up.

This report covers an announced baseline event follow-up inspection. The inspection was conducted by two Region III inspectors. One green finding associated with a non-cited violation was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified and Self-Revealed Findings:

Cornerstone: Mitigating Systems

- Green. A finding was self-revealed when operators discovered that a carbon dioxide suppression system isolation valve for the relay room had been mis-positioned in the closed position rendering the suppression system non-functional. The valve was opened on the same day that it had been discovered closed, thereby, addressing the immediate safety concern. The primary cause of this finding was related to the cross-cutting area of Human Performance. Operators failed to open the valve following a maintenance activity. In addition, operators failed to identify that the valve was mis-positioned in the closed position during two subsequent valve position surveillance activities.

  This issue was more than minor because the suppression capability of a fire suppression system was adversely affected in that the system was rendered non-functional for 64 days. Both trains of equipment could be affected by a fire in the relay room and a shutdown from outside the control room could be required due to such a fire. The issue was determined to be of very low safety significance based on a Phase 3 significance determination evaluation. The issue was a Non-Cited Violation of Technical Specification 5.4.1 which required that written procedures be established, implemented, maintained for the implementation of the fire protection program, and of a license condition requirement to maintain the carbon dioxide system protecting the relay room operable. (Section 4OA3.a)

B. Licensee-Identified Violations:

A violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee has been entered into the licensee's corrective action program. This violation and the licensee's corrective action tracking number are listed in Section 4OA7.

4OA3 Event Follow-Up (71153)

a. Unresolved Item 05000282/2005004-03; 05000306/2005004-03, Configuration Control Event Causes a Loss of Automatic Fire Suppression to the Relay Room (Closed):

Introduction: The inspectors identified a non-cited violation (NCV) of very low safety significance associated with the self-revealed isolation of the fire suppression system for the relay room.

Description: On February 11, 2005, operators discovered that the carbon dioxide outlet valve, valve FP-14-1, was mis-positioned in the closed position. Valve FP-14-1 was the isolation valve from the carbon dioxide tank, which provided carbon dioxide suppression to the relay room. The valve was identified as mis-positioned when operators had planned to close the valve to provide isolation for a maintenance activity; and operators noted that the valve was already closed. The valve was then cycled open and then closed to provide isolation for the maintenance activity. The valve was opened on February 11, 2005, following completion of the maintenance activity.

The relay room is located directly below the dual unit control room and served as a cable spreading area for both units. The room contained a number of 120 Volt alternating current and some direct current electrical cabinets. Almost all control room circuits passed through the relay room. As such, a fire in the relay room had the potential to affect both trains of either unit and result in a shutdown from outside the control room being necessary. In addition, a computer room is located within the relay room. Smoke detectors located in both the relay room and the computer room provide annunciation in the control room of a potential fire. The carbon dioxide suppression system was designed to automatically activate if any of the thermal detectors in the relay room or computer room reach a temperature of 140 degrees Fahrenheit. The carbon dioxide suppression system could also be activated manually from a local control station located just outside one of the doors to the relay room. If a fire had activated the thermal detectors and, as a result, activated the carbon dioxide suppression system, control room operators would have received indication that the carbon dioxide suppression system had been activated. However, for the period that valve FP-14-1 was isolated, the system would have been non-functional and, as such, control room operators would not have been aware that the carbon dioxide system had failed to discharge. A local visual verification would have been required to confirm that the carbon dioxide suppression system had discharged. However, neither the pre-fire plans for the relay room nor the procedure for operation of the carbon dioxide suppression system, procedure C31, “Fire Protection & Detection Systems,” directed operators to verify the position of valve FP-14-1 in the event that the carbon dioxide suppression system had failed to discharge.

The inspectors reviewed the licensee’s investigation of the issue, documented in Root Cause Evaluation (RCE) 000200, “Cardox Unavailability due to Valve Mispositioning.” The licensee determined that the valve had most likely been closed since December 9, 2004. The licensee made this determination based on their review of operating logs, work orders, system isolations, and procedures, which did not identify any time between December 9, 2004, and February 11, 2005, when valve FP-14-1 would have been closed. The valve had been closed on December 9, 2004, to isolate the carbon dioxide system for a maintenance activity conducted under work order 0406411, “Relay/Comp Rm Cardox

Indication Light Failed.” Step 7.3.2 of work order 0406411 directed operators to open valve FP-14-1. The licensee determined that for this restoration step, the operator most likely turned the valve 1.5 turns until the slack in the gearbox was taken up. Discussions with licensee staff indicated that the carbon dioxide in the tank was stored below zero degrees Fahrenheit and, when the valve was closed on December 9, 2004, the piping downstream of the valve warmed up resulting in a differential temperature and pressure across the valve disc. The differential temperature and pressure across the valve disc would have caused thermal binding of the valve accounting for the additional resistance that the operator encountered when attempting to open the valve on December 9, 2004. The licensee concluded that the operator interpreted the encountered resistance as indication that the valve was fully open.

The inspectors reviewed the vendor manual for the carbon dioxide tank isolation valve and determined that when the valve was in its fully open position, the locking tab on the gear case would be in alignment with the locking tab on the sector. With the two locking tabs aligned, the valve could be padlocked in the open position. The inspectors interviewed operations personnel and reviewed procedure SP 1200, “Fire Protection System Supply to Safety Related Areas Valve Check.” Based on this information, the inspectors determined that the licensee’s process for locking the valve in its open position was to place a lock wire through holes in both drilled locking tabs and then around the handwheel. RCE000200 documented that on February 11, 2005, when the valve was identified as being mispositioned in the closed position, one of the valves tabs was rotated down with the lock wire through one tab and around the handwheel. The inspectors noted that the valve did not have a rising stem or other form of indication for operators to rely on for determining the position of the valve.

The licensee identified that operators had performed procedure SP 1200, "Fire Protection System Supply to Safety-Related Areas Valve Check," on January 1, 2005 and on January 29, 2005. The operators performing these valve position verification surveillances failed to identify that valve FP-14-1 was closed and documented the valve position as being open. The failure to correctly implement procedure SP 1000 on two occasions was a licensee identified violation of Technical Specification 5.4.1.d which required that written procedures be established, implemented, and maintained for Fire Protection program implementation.

Requirement 4.4.1 of Operations Manual, Section F5, Appendix-K, required that the carbon dioxide system protecting the relay and cable spreading room area be operable. Requirement 4.4.3 of Operations Manual, Section F5, Appendix-K, specified that if requirement 4.4.1 could not be met, that a continuous fire watch with backup fire suppression equipment be stationed in the relay and cable spreading room. Based on discussions with licensee engineering personnel, the inspectors determined that continuous fire watches in the relay room had been temporarily established to support maintenance activities on December 9, 2004, and February 11, 2005. However, there was no continuous fire watch from December 9, 2004, through February 11, 2005. Hourly fire watches for the relay room had been established as a compensatory measure from December 2003 through October 2005 to address safe shutdown analysis circuit issues. However, the hourly fire watches did not satisfy requirement 4.4.3 for a continuous fire watch.

<u>Analysis</u>: The inspectors determined that the failure to return the carbon dioxide suppression system for the relay room to an operable status was a performance deficiency warranting a significance determination. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix-B, "Issue Disposition Screening," issued on September 30, 2005. The finding involved the attribute of protection against external factors (fire) and affected the mitigating system cornerstone because the finding adversely affected the suppression capability of a fire suppression system. Specifically, the continued isolation of valve FP-14-1 rendered the relay room fire suppression system non-functional for 64 days. The finding was related to the cross-cutting area of Human Performance because a personnel error resulted in the valve continuing to be closed following a maintenance activity and the mispositioning remained undetected despite two surveillance activities to verify that the valve was open. The inspectors considered the manner in which the valve was discovered to be closed to be self-revealing in nature. Specifically, the licensee program for periodically verifying the position of the valve had twice failed to identify that the valve was mispositioned. Operators discovered that the valve was already closed when they had planned to close the valve for a maintenance activity. The inspectors reviewed IMC 0609F, "Fire Protection Significance Determination Process," issued on February 28, 2005, and determined that a Phase 3 significance determination evaluation was required due to the complexity of potential fire scenarios in the relay room. The inspectors noted that both trains of equipment could be affected by a fire in the relay room and a shutdown from outside the control room could be required due to such a fire.

The inspectors reviewed a significance analysis performed by the licensee, "Significance Determination Process (SDP) Phase 2, Prairie Island Unit 1 and 2, CAP 040948, 'Found FP-14-1 closed while performing isolation for WO#0406228," dated December 16, 2005. The licensee's significance analysis primarily used IMC 0609F methodology. However, the following exceptions were noted:

- A duration factor of 0.175 was assigned instead of 1.0. The 0.175 value was based on the carbon dioxide system being isolated for 64 days rather than a full year; and

- The analysis used a horizontal flame spread rate of 0.71 inches per minute and a vertical flame spread rate of 26 inches per minute for fires involving thermoset cables versus the IMC 609F, Attachment 3, "Guidance for Identifying Fire Growth and Damage Scenarios," value of 10 feet per hour (2 inches per minute) for horizontal cable trays. The licensee's values were based on the NUREG/CR-6850, "Fire PRA [Probabilistic Risk Assessment] Methodology for Nuclear Power Facilities," Appendix-R, "Appendix for Chapter 11, Cable Fires," values of 0.3 millimeters per second for horizontal flame spread and 11 millimeters per second for vertical flame spread for cross-linked polyethylene cables.

The licensee had identified that the primary contributor to risk was fire initiating in electrical cabinets and affecting both trains of a single unit through fire spread via cable trays. Through computations performed using the National Institute of Standards and Technology Fire Dynamics Simulation software, the licensee determined that development of a hot gas layer was not a significant contributor towards risk. The following key assumptions were made by the licensee:

- A remote shutdown, with a failure probability of 0.1, would be required for the affected unit when both trains were affected;

- The auxiliary feedwater cross-tie from the opposite unit, with a failure probability of 0.1, would be available for shutting down the affected unit;

- A fire would be detected in one minute;

- Unit 2 was similar to Unit 1. (The licensee had performed their analysis based on Unit 1); and

- The Mean Rate Constant of 0.177 from IMC 609F, Table 2.7.1, "Non-Suppression Probability Values for Manual Fire Fighting Based on Fire Duration (Time to Damage after Detection) and Fire Type Category," for cable fires was applicable for determining the probability of manual non-suppression. The inspectors considered the Mean Rate Constant of 0.117 for electrical fires to be applicable because the postulated fires originated in electrical cabinets, not cables. The inspectors performed independent calculations which showed that use of 0.117 for the Mean Rate Constant did not substantially affect the result.

The licensee performed a detailed analysis for 15 electrical cabinets contributing towards time to damage in less than 40 minutes. The inspectors noted that seven cabinets had times to damage in excess of 30 minutes and three cabinets had times to damage less than 10 minutes. Based on field observations, the inspectors had identified four additional cabinets with times to damage of approximately 30 minutes. The inspectors performed independent calculations which showed that the risk contribution from the additional four cabinets was not significant.

The inspectors noted that the licensee had modeled the fire scenario for panel 13 DC with a cable air drop to the center of the cabinet. Field observations performed by the inspectors indicated that the there were air drops at both ends of the cabinet which would reduce the time to damage estimate. The inspectors performed independent calculations which showed that the discrepancy did not significantly change the result of the analysis.

The licensee's analysis in conjunction with the inspectors' review constituted a Phase 3 significance determination evaluation. The inspectors performed a sensitivity analysis taking into consideration the more conservative Mean Rate Constant, contribution from the additional cabinets, and revised time to damage modeling for panel 13 DC, and determined that safety significance of the finding did not exceed that of very low safety significance (i.e., Green).

Enforcement: Technical Specification 5.4.1.d required that written procedures be established, implemented, and maintained for implementation of the Fire Protection program. Work order 0406433 was a written procedure for implementation of the Fire Protection program. Step 7.3.2 of work order 0406433 directed that valve FP-14-1 be opened. License condition 2.C.(4) of the Prairie Island Nuclear Generating Plant Unit 1 and Unit 2 Facility Operating License stated, in part, that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described and referenced in its Updated Final Safety Analysis Report (UFSAR). Section 10.3.1 of the

UFSAR described the licensee's fire protection program, and Section 10.3.1.1 of the UFSAR stated that Operations Manual, Section F5, Appendix-K, "Fire Detection and Protection Systems," was part of the UFSAR by reference. Requirement 4.4 of Operations Manual, Section F5, Appendix-K, described the operability requirements for the Carbon Dioxide System. Specifically, requirement 4.4.1 of Operations Manual, Section F5, Appendix-K, required that the carbon dioxide system protecting the relay and cable spreading room area (i.e., the relay room) be operable. Requirement 4.4.3 of Operations Manual, Section F5, Appendix-K, specified that if requirement 4.4.1 could not be met, that a continuous fire watch with backup fire suppression equipment be stationed in the relay and cable spreading room area within one hour. Contrary to the above, on December 9, 2004, step 7.3.2 of work order 0406433 was not implemented in that valve FP-14-1 was not opened as directed by the step. As a result, the carbon dioxide system protecting the relay and cable spreading room area was inoperable in that the carbon dioxide tank isolation valve, valve FP-14-1, was closed from December 9, 2004, through February 11, 2005, without a continuous fire watch. With valve FP-14-1 closed, the carbon dioxide system would not have discharged if the system had been activated either automatically or manually. The licensee opened the valve on February 11, 2005, and entered the issue into their corrective action program as CAP040948. In addition, the inspectors observed that, as a corrective action, the licensee had added a highly visible yellow "Open/Close" tab on the gear case and a highly visible yellow "arrow" tab on the sector that points to the Open/Close tab so that operators could more easily identify the position of the valve. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000282/2005012-01; 05000306/2005012-01). The unresolved item associated with this issue is closed.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. Palmisano and other members of licensee management at the conclusion of the inspection on January 5, 2005. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violations of very low significance were identified by the licensee and are violations of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as NCVs.

Cornerstone: Mitigating Systems

As discussed in Section 4OA3.a, the licensee identified a violation. Specifically, Technical Specification 5.4.1.d required that written procedures be established, implemented, and maintained for Fire Protection program implementation. Fire Protection Program procedure SP 1200 directed operators to verify that valve FP-14-1 was open. Contrary to the above, on January 1, 2005 and on January 29, 2005, licensee operators performing procedure SP 1200 failed to implement procedure SP 1200 in that they failed to verify that valve

FP-14-1 was open. Valve FP-14-1 was subsequently determined to have been closed since December 9, 2004. This issue was documented in the corrective action program as CAP40948.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
T. Palmisano, Site Vice-President
P. Huffman, Plant Manager
M. Brossart, Manager, Programs Engineering
T. Asmus, Probabilistic Risk Assessment Engineer, Engineering Programs
K. Began, Appendix-R Program Engineer, Engineering Programs
M. Davis, Regulatory Analyst, Regulatory Assurance
F. Forrest, Manager, Operations
J. Kivi, Regulatory Compliance Engineer, Regulatory Assurance
J. Masterlark, Fleet Probalistic Risk Assessment Lead, Nuclear Management Company

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000282/2005012-01; 05000306/2005012-01	NCV	Configuration Control Event Causes a Loss Fire Suppression to the Relay Room (Section 4OA3.a)

Closed

05000282/2005004-03; 05000306/2005004-03	URI	Configuration Control Event Causes a Loss of Fire Suppression to the Relay Room (Section 4OA3.a)

DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Analyses

Significance Determination Process (SDP) Phase 2, Prairie Island Unit 1 and 2, CAP 040948, "Found FP-14-1 closed while performing isolation for WO#0406228"; dated December 16, 2005

Corrective Action Documents

CAP040948; Found FP-14-1 closed while performing isolation for WO# 0406228; dated February 11, 2005

RCE000200; Cardox Unavailability Due to Valve Mispositioning; Revision 2

Procedures

C31; Fire Protection & Detection Systems; Revision 36

F5, Appendix-A, Fire Detection Zone 12; Revision 16

F5, Appendix-B; Control Room Evacuation (Fire); Revision 33

F5, Appendix-K; Fire Detection and Protection Systems; Revision 9

SP 1200; Fire Protection System Supply to Safety Related Areas Valve Check; Revision 27

Work Orders

0406228; Replace Glass for Manual PD broke during SP 1194; completed February 11, 2005

0406433; Relay/Comp Rm Cardox Indication light Failed; completed December 9, 2004

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
AV	Apparent Violation
CFR	Code of Federal Regulations
DPR	Demonstration Power Reactor
DRS	Division of Reactor Safety
IMC	Inspection Manual Chapter
LLC	Limited Liability Corporation
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
RCE	Root Cause Evaluation
SDP	Significance Determination Process
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item