March 22, 2001

EA-01-050

Mr. J. Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NRC INSPECTION REPORT 50-282/01-02(DRP); 50-306/01-02(DRP)

Dear Mr. Sorensen:

On February 22, 2001, the NRC completed an inspection at your Prairie Island Nuclear Generating Plant. The enclosed report documents the inspection findings which were discussed on February 22, 2001, with you and other members of your staff.

This 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, the inspectors identified two issues of very low safety significance (Green). One of these issues was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating the issue as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region 3; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Prairie Island facility.

In accordance with 10 CFR 2.790 of the NRC's “Rules of Practice,” a copy of this letter and its enclosure 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 Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

**/RA/**

Roger D. Lanksbury, Chief
Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 50-282/01-02(DRP); 50-306/01-02(DRP)

cc w/encl:
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REGION III

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| Docket Nos: | 50-282, 50-306 |
| License Nos: | DPR-42, DPR-60 |
| Report No: | 50-282/01-02(DRP); 50-306/01-02(DRP) |
| Licensee: | Nuclear Management Company, LLC |
| Facility: | Prairie Island Nuclear Generating Plant |
| Location: | 1717 Wakonade Drive East<br>Welch, MN 55089 |
| Dates: | January 1 through February 22, 2001 |
| Inspectors: | S. Ray, Senior Resident Inspector<br>S. Thomas, Resident Inspector |
| Approved by: | Roger D. Lanksbury, Chief<br>Projects Branch 5<br>Division of Reactor Projects |

# NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

**Reactor Safety**

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

**Radiation Safety**

- Occupational
- Public

**Safeguards**

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

## SUMMARY OF FINDINGS

IR 05000282-01-02(DRP); IR 05000306-01-02(DRP), on 1/1 through 2/22/2001; Nuclear Management Company, Prairie Island Nuclear Generating Plant, Units 1 & 2. Adverse weather protection, maintenance risk assessment and emergent work control.

The inspection was conducted by resident inspectors. The inspection identified two Green findings, one of which was a non-cited violation. The significance of the findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process” (SDP).

<u>Inspector-Identified Findings</u>

**Cornerstone: Mitigating Systems**

- Green. The inspectors identified that ice blockage was forming in the cooling water emergency dump-to-grade line due to a leaking isolation valve. The problem was discovered and resolved before the piping became substantially blocked, so no regulatory concerns were identified.

  The finding was of very low safety significance because the issue would have only been a problem in the extremely unlikely event that the line had become completely blocked by ice at the same time that both of the normal discharge lines were blocked due to a seismic or similar event (Section 1R01).

- Green. The inspectors identified a non-cited violation of Section (a)(4) of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” in that the licensee failed to assess the risk associated with the performance of Work Order 0007629, “Transfer 480 Volt Safeguards Buses 111 and 112 to Alternate Source,” which was later shown to cause an increase in the core damage frequency for Unit 2 because it caused the D5 diesel generator to be unavailable.

  This finding was of very low safety significance because, although the increase in risk rate was relatively high, the change in core damage probability was very low (approximately 1.18E-8) due to the short time that D5 was unavailable (1.55 hours) in comparison to the allowed outage time for this plant configuration (5.5 days) (Section 1R13.1).

## Report Details

Summary of Plant Status:

Unit 1 was operated with power gradually coasting down from 100 percent power to approximately 80 percent power before the unit was shutdown for a refueling outage on January 19, 2001. Unit 1 was maintained shut down for the remainder of the inspection period. Unit 2 was operated at or near full power for the entire inspection period.

### 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity**

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors inspected the cooling water emergency dump-to-grade line for protection from freezing conditions.

b. Findings

A Green finding was identified when the inspectors noted that the cooling water emergency dump-to-grade line was not adequately protected from freezing and was partially blocked by an ice buildup.

On January 31, 2001, the inspectors noted an ice buildup in the cooling water emergency dump-to-grade line outside of the auxiliary building. The line was designed to provide an alternate, safety-grade, discharge flow path for the cooling water system in case the two normal, nonsafety-grade, discharge paths through the circulating water system became blocked due to an event, such as a collapse as a result of an earthquake. A recent change to the valve alignment in the system had resulted in system pressure being present against the final isolation valve for the line. Normally, additional valves had provided extra isolation. The final isolation valve was apparently leaking and water was dripping out of the end of the pipe. The pipe end was exposed to the outside weather and the leaking water was freezing and starting to form an ice buildup on the inside of the dump line. Loss of the cooling water discharge paths would have resulted in no cooling water flow to the Unit 1 diesel generators and to heat exchangers for both units' component cooling, containment cooling, auxiliary building cooling, control room cooling, and emergency core cooling pump cooling functions.

The inspectors brought the situation to the attention of the shift supervisor. Licensee personnel immediately installed insulation around the exposed portion of the line and directed the discharge from a portable heater into the end of the pipe in accordance with Work Order (WO) 0100562, "Cooling Water Dump to Grade Pipe has Ice Buildup." Those corrective actions rapidly melted the ice. A few days later, when the system lineup was returned to normal, the insulation and heater were removed. The licensee entered the condition into its corrective action system as Condition Report (CR) 20011119, “Cooling Water Dump to Grade in the Auxiliary Building has Ice Buildup,” and also wrote WO 0100669, “Valve MV-32038 Does Not Seat Properly,” to direct the repair of the leaking isolation valve at a future date.

This finding was considered to be of more than minor safety significance because, if left uncorrected, it would have become a more significant safety concern. The entire dump line could have become blocked with ice, and no cooling water flow path would have been available to mitigating system loads in the event of collapse or blockage of the normal discharge paths. The issue also affected a cornerstone because loss of cooling water flow would have affected the operability of several mitigating systems as discussed above. The finding was assumed to be potentially risk significant for external initiating event core damage sequences using the Seismic, Fire, Flooding, and Severe Weather Screening Criteria in the Phase 1 SDP worksheet of IMC 0609. This was because the finding involved the degradation of equipment (the emergency dump line) specifically designed to mitigate a seismic initiating event and, if the safety function of the cooling water system was assumed to be completely unavailable, it would degrade more than a single train of a multi-train safety system or function.

The NRC had previously analyzed the risk significance of the potential loss of the entire cooling water system as a result of a seismic event as part of its review of the findings in Inspection Report 50-282/00-13(DRS); 50-306/00-13(DRS). The NRC determined that such a finding was a Green issue. This finding was very similar to the potential loss of cooling water pump bearing cooling water in a seismic event discussed in Section 1R21.2.b of that report and was bounded by that analysis since the exposure time of the system to the condition was much shorter in this case. Therefore this finding was of very low safety significance (Green) and was within the licensee’s response band. The finding was assigned to the mitigating systems cornerstone of both Units 1 and 2.

Since the inspectors identified, and the license corrected, the potential ice blockage before a significant portion of the flow area was blocked, no regulatory concerns were identified.

## 1R04 Equipment Alignment

### a. Inspection Scope

The inspectors performed partial walkdowns of the following equipment to verify that critical portions of the redundant system or train, or other significant protected equipment, was in the correct lineup during the time when one safety significant system or train was out-of-service:

- the electrical substation during unavailability of the 12 component cooling heat exchanger; and
- the 22 turbine-driven auxiliary feedwater pump during testing of the 21 motor-driven auxiliary feedwater pump.

As part of these inspections, the inspectors reviewed the following documents:

- Daily At-Power Risk Report for 7:00 1/5/2001;
- Daily At-Power Risk Report for 7:00 1/29/2001;
- Operating Procedure C20.2, "Substation System," Revision 5; and
- System Prestart Checklist C28-7, "Auxiliary Feedwater System Unit 2," Revision 43.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted fire protection walkdowns focused on availability, accessibility, and condition of fire fighting equipment and on the condition and operating status of installed fire barriers. The inspectors selected the following fire areas for inspection based on their overall contribution to internal fire risk, as documented in the Individual Plant Examination of External Events (IPEEE):

- Fire Area 3, 121 control room chiller room;
- Fire Area 92, 122 control room chiller room;
- Fire Area 2, Unit 1 auxiliary building, 755 foot elevation; and
- Fire Area 76, Unit 2 auxiliary building, 755 foot elevation.

As part of these inspections, the inspectors reviewed the following documents:

- IPEEE, NSPLMI-96001, Appendix B, "Internal Fires Analysis," Revision 1; and
- Plant Safety Procedure F5 Appendix F, "Fire Hazard Analysis," Revision 12.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the performance testing of the 11 and 12 component cooling heat exchangers subsequent to the Unit 1 shutdown for a refueling outage. The following aspects of heat exchanger testing were reviewed:

- test acceptance criteria and results appropriately considered the differences between testing conditions and design conditions;
- inspection results were appropriately categorized against pre-established engineered acceptance criteria, and were acceptable;

- frequency of testing was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values; and
- test results considered test instrument inaccuracies and differences.

The inspectors also reviewed WO 0004856, “Surveillance Procedure [SP] 1304 Unit 1 Heat Exchanger Performance Test,” and the following supporting attachments:

- Unit 1 Component Cooling Heat Exchanger Performance Test Report 1/01;
- 11 Component Cooling Heat Exchanger Test;
- 12 Component Cooling Heat Exchanger Test;
- Data; and
- Software Validation.

b. Findings

No findings of significance were identified.

## 1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors verified the licensee’s implementation of the maintenance rule for structures, systems, or components (SSCs) with performance problems. This evaluation included the following aspects:

- whether the SSC was scoped in accordance with 10 CFR 50.65;
- whether the performance problem constituted a maintenance rule functional failure;
- safety significance classification;
- the proper 10 CFR 50.65 (a)(1) or (a)(2) classification for the SSC; and
- the appropriateness of the performance criteria for SSCs classified as (a)(2) or the appropriateness of goals and corrective actions for SSCs classified as (a)(1).

The inspectors reviewed the licensee’s implementation of the maintenance rule requirements for the following SSCs:

- Unit 2 chemical and volume control system;
- Unit 1 safety injection system; and
- Unit 2 safety injection system.

As part of these inspections, the inspectors reviewed the 1999 Annual and First Quarter Equipment Performance Report, dated May 2, 2000; Second Quarter Equipment Performance Report, dated July 28, 2000; Third Quarter Equipment Performance Report, dated October 26, 2000; Fourth Quarter Equipment Performance Report, dated January 23, 2001; and Prairie Island Maintenance Rule System Basis Document. The inspectors also reviewed the following procedures, WOs, and CRs:

Unit 2 Chemical and Volume Control System

- H Procedure H12, “Plant Check Valve Program,” Revision 3;
- H Procedure H10.1, “ASME [American Society of Mechanical Engineers] Section XI Inservice Testing Implementing Program,” Revision 9;
- H Procedure H5, “Motor Operated Valve Program,” Revision 6;
- H Procedure H24, “Maintenance Rule Program,” Revision 2;
- WO 9406284, “Wire Code Changes at MV-32062 Refueling Water Storage Tank [RWST] to Charging Pumps”;
- WO 9501058, “P32062L 21 RWST to Charging Pump Lubrication”;
- WO 9700505, “P32062 Remove/Replace/Test MV-32062 Actuator”;
- WO 9804538, “P32062 2 RWST Emergency Makeup to Charging Pump D70 Inspection”;
- WO 9807077, “P32062L 21 RWST to Charging Pump Motor Operated Valve Lubrication”;
- WO 9406665, “Wire Code Changes at MV-32194 Excess Letdown Line Isolation”;
- WO 9501089, “P32194L Unit 2 Excess Letdown Motor Operated Valve Lubrication”;
- WO 9607961, “P32194 Unit 2 Excess Letdown/Seal Isolation D70 Inspection”;
- WO 9700862, “Perform Differential Pressure Test of MV-32194 and MV-32210”;
- WO 9501090, “P32210L Unit 2 Reactor Coolant Pump Seal Return/Excess Letdown Motor Operated Valve Lubrication”;
- WO 9607962, “P32210 Unit 2 Excess Letdown/Seal Water Return Isolation D70 Inspection”;
- WO 9400711, “Perform Planned Maintenance on MV-32189 and Test”;
- WO 0013056, “22 Charging Pump Manual Pot on Hand Controller 2HC-428B Stop is Broke”;
- WO 9803777, “Both Level Transmitters (2LT-112 and 2LT-141) are Diverging in Reading - By Inspection It Appears that Transmitter 2LT-141 is Drifting High, Possibly Due to Loss of Fill Fluid in Sensor”;
- WO 9803810, “Over the Past Several Days, 2LT-112 has been Drifting Downward - It Reads About 20 percent Lower than Actual Level”;
- WO 9812633, “2VC-8-4 Did Not Pass Local Leak Rate Test Per SP 2072.13B”;
- CR 19993448, “Non-Outage Out of Service Between Unit 1 and Unit 2 Charging Pumps Varies by a Significant Margin”;
- CR 20000384, “Equipment Qualification Resolution for CV-31339 and CV-31430 Letdown Isolation Valves”; and
- CR 19980887, “2LT-112 and 2LT-141 Rosemount Transmitter Sealed Reference Leg Leaked.”

Unit 1 Safety Injection System

- WO 9402042, “Inspect Actuator and VOTES [Valve Operator Testing and Evaluation System] Test MV-32075 per D70”;
- WO 9506549, “P32075 Remove/Reinstall Actuator and VOTES Test MV-32075”;
- WO 9506550, “P32076 Remove/Reinstall Actuator and VOTES Test MV-32076”;
- WO 9712436, “MV-32077 Dual Indication; Investigate and Repair”;
- WO 9601313, “Packing Leak on MV-32078”;

- WO 9707667, “P32079 Safety Injection Pump Suction from RWST D70 Inspection”;
- WO 9707668, “P32080 Safety Injection Pump Suction from RWST D70 Inspection”;
- WO 9900603, “P32207 Residual Heat Removal to Safety Injection Pump D70 Inspection”;
- WO 9707667, “P32067 Safety Injection to Reactor Vessel Isolation D70 Inspection”;
- WO 9707663, “P32069 Safety Injection to Reactor Vessel Isolation D70 Inspection”;
- WO 9707669, “P32081 Safety Injection Pump Suction from BAST [boric acid storage tank] D70 Inspection”;
- WO 9707670, “P32082 Safety Injection Pump Suction from BAST D70 Inspection”;
- WO 9400070, “Remove Brakes, Static VOTES Test MV-32162”;
- WO 9402071, “Remove Brakes, Static VOTES Test MV-32163”;
- WO 9707682, “P32202 Safety Injection Test to 11 RWST D70 Inspection”; and
- WO 9707683, “P32203 Safety Injection Test to 11 RWST D70 Inspection.”

Unit 2 Safety Injection System

- WO 9911687, “P32178 Containment Sump B Residual Heat Removal Suction D70 Inspection”;
- WO 9911689, “P32180 Containment Sump B Residual Heat Removal Suction D70 Inspection”;
- WO 9911688, “P32179 Containment Sump B Residual Heat Removal Suction D70 Inspection”;
- WO 9501040, “P32181 Containment Sump B Residual Heat Removal Suction D70 Inspection”;
- WO 9800722, “P32182 Safety Injection Pump Suction from Refueling Water Storage Tank D70 Inspection”;
- WO 9911693, “P32208 Residual Heat Removal to 21 Safety Injection Pump D70 Inspection”;
- WO 9911694, “P32209 Residual Heat Removal to 22 Safety Injection Pump D70 Inspection”;
- WO 9607957, “P32170 Safety Injection to Reactor Vessel D70 Inspection”;
- WO 9607958, “P32172 Safety Injection to Reactor Vessel D70 Inspection”;
- WO 9800724, “P32184 Safety Injection Pump Suction from BAST D70 Inspection”;
- WO 9800725, “P32185 Safety Injection Pump Suction from BAST D70 Inspection”;
- WO 9804803, “P32191 22 Safety Injection Pump Suction D70 Inspection”;
- WO 9612443, “P32204 Safety Injection Test to 21 RWST D70 Inspection”; and
- WO 9612444, “P32205 Safety Injection Test to 21 RWST D70 Inspection.”

Other Documents

- CR 20011162, “Cooling Water System in Maintenance Rule (a)(1) Category Based on Out of Service Time Exceeding Performance Criteria”; and

- CR 20011178, “Assess the Maintenance Rule Significance of CRs 19992587 and 20002420 (Mispositioning of D3/D4 Output Breakers).”

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

.1 Risk Assessment and Control Associated With Unit 1 Bus 15 Outage

a. Inspection Scope

The inspectors reviewed the licensee’s management of plant risk for both units during the preparations and conduct of work associated with the Unit 1 safeguards 4160 volt Bus 15 outage. The inspectors verified that evaluation, planning, control, and performance of the work were done in a manner to reduce risk where practical, and that contingency plans were in place where appropriate. As part of this inspection, the inspectors attended various outage and scheduling meetings, and reviewed the following documents:

- Daily At-Power Risk Report for 8:53 1/27/2001;
- WO 0007629, “Transfer 480 Volt Safeguards Buses 111 and 112 to Alternate Source,” Revision 0;
- Abnormal Operating Procedure 2C20.7, “Bus 25 Load Sequencer Out of Service,” Revision 6; and
- CR 20010836, “D5 Unavailable Without Risk Assessment.”

b. Findings

A Green finding and non-cited violation was identified when the inspectors noted that the licensee failed to assess the risk associated with a plant configuration which made Breaker 15-8, the Bus 25 load sequencer, and the D5 diesel generator unavailable at the same time. During this time, D5 was unavailable to the operating unit (Unit 2) for 1.55 hours without a prior risk assessment having been performed for this plant configuration.

On the morning of January 27, 2001, licensee operators were in the process of isolating Bus 15, the Unit 1 Train ‘A’ safeguards 4160 volt bus, for refueling maintenance. To support this work, the power sources for Unit 1 480 volt safeguards buses (Buses 111 and 112) were transferred to Bus 25, the Unit 2 Train ‘A’ safeguards 4160 volt bus. This switching evolution required the Bus 25 load sequencer be placed in “manual” and the D5, Unit 2 Train ‘A’ safeguards emergency diesel generator, maintenance switch be placed in “maintenance.” This placed both Bus 25 and the D5 diesel generator in Technical Specification limiting conditions for operation. Prior to releasing the WO, the duty shift manager called a licensee risk analyst to determine whether a risk assessment was needed. The consensus was reached that the restoration would be simple and that since restoration instructions were provided in the WO procedure the work could continue without a quantitative risk assessment. This determination was made by the risk analyst without reviewing the procedure attached to the WO.

Later that morning, after reviewing the licensee's Daily At-Power Risk Report for January 27, the inspectors questioned the licensee risk analyst about the fact that the daily risk assessment did not evaluate the fact that Breaker 15-8, the Bus 25 load sequencer, and D5 would be unavailable at the same time. The risk analyst investigated and determined that the previous assumption made about the restoration of the Bus 25 load sequencer and D5 was incorrect. It was found that instructions for restoration of Bus 25 were provided for in the WO procedure, but not for the D5 diesel generator. Therefore, D5 was unavailable to the operating unit (Unit 2) for 1.55 hours without a prior risk assessment having been performed for this plant configuration.

A licensee risk analyst performed a risk evaluation for Unit 2 for a plant configuration which included Breaker 15-8 being out of service and D5 being unavailable. The result of the risk evaluation was a core damage frequency (CDF) of 8.42E-5 per year, with an allowed outage time of 5.5 days. This was approximately 4.7 times higher than the Unit 2 baseline CDF. This plant configuration placed Unit 2 in the "Yellow" band for risk rate in accordance with licensee procedures.

Section (a)(4) of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," required, in part, that before performing maintenance activities, the licensee assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to this, on January 27, 2001, the licensee failed to assess the risk associated with the performance of WO 0007629, "Transfer 480 Volt Safeguards Buses 111 and 112 to Alternate Source," which was later shown to cause an increase in the CDF for Unit 2. This violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy (50-282/01-02-01(DRP); 50-306/01-02-01(DRP)). This issue was placed in the licensee's corrective action program as CR 20010836. Operations management considered this a noteworthy event and prepared a PINGP (Prairie Island Nuclear Generating Plant) Form 1224, "Crew Meeting Review of Noteworthy Event/Near Miss/Change," and briefed the operations crews on this issue.

This finding was considered to be of more than minor safety significance because, due to its actual increase in risk, it had a credible impact on safety. The issue affected a cornerstone because it could have impacted electrical power to and sequencing of several mitigating system trains. Although the increase in risk was relatively high, the change in core damage probability was very low (approximately 1.18E-8) due to the short time that D5 was unavailable (1.55 hours) in comparison to the allowed outage time for this plant configuration (5.5 days). This increase in risk was less than the 1E-6 criteria in IMC 0609 for the Green/White threshold, therefore, this issue was considered to be of very low risk significance (Green) and within the licensee response band. This finding was assigned to the mitigating systems cornerstone for Unit 2.

.2 Other Activities

a. Inspection Scope

The inspectors reviewed the licensee's management of plant risk during maintenance activities and its control of emergent work activities. The inspectors verified that evaluation, planning, control, and performance of the work were done in a manner to reduce risk where practical, and that contingency plans were in place where appropriate. The following activities were inspected:

- replacement of the 11 regenerative heat exchanger auxiliary spray valve to 11 pressurizer control valve and the B loop pressurizer spray valve in accordance with WO 0003101, "CV-31225 Valve Replacement," and WO 9912283, "Replace CV-31329";
- troubleshooting and adjustment of the 12 component cooling heat exchanger temperature control valve controller in accordance with WO 0100025, "12 Component Cooling Heat Exchanger Outlet Temperature Fluctuating"; and
- troubleshooting and repair of the breakers for the 121 control room chiller and chilled water pump in accordance with WOs 0100325 and 0100326, "Breaker Would Not Close - Repair Handle Mechanism."

As part of these inspections, the inspectors reviewed the following additional documents:

- WO 9905374, "Repair SI-9-2 [cold leg safety injection line to loop A cold leg check valve] Due to Furmanite Repair";
- WO 0100776, "Control Reactor Coolant System (RCS) Boundary During RCS Level Changes";
- CR 20011257, "Discovered RCS Draining to Sump A Via Vent and Drain Valves Opened for CV-31329 Replacement"; and
- CR 20011259, "The Actuator and Valve Internals Were Removed from CV-31224 While the Valve Was Being Used for Boundary Control."

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed a sampling of operability evaluations for safety significant systems and conditions to determine that operability was justified, that availability was assured, and that no unrecognized increase in risk had occurred. The following evaluations were reviewed:

- CR 20003286, "Evaluate Short Circuit Rating of Transfer Switches for AB and MA Motor Control Centers";

- CR 20006112, “Discrepancy in Containment Heat Sink Information in Updated Safety Analysis Report”; and
- CR 20011213, “1SX-22 Failed to Isolate MV-32033 [Unit 2 turbine loop B cooling water header valve] during SP1126 [Turbine Building Cooling Water Header Isolation Safety Injection Relays 1SI-12X and 1SI-22X Refueling Test].”

As part of these inspections, the inspectors reviewed the following additional documents:

- Prairie Island Calculation ENG-ME-449, “Assessment of Containment Heat Sinks,” Revision 0;
- Prairie Island Calculation ENG-ME-312, “Calculation of Steam Generator Allowed Maximum Level Due to Main Steam Line Break Considerations,” Addendum 1, Revision 3;
- Prairie Island Calculation SPCEP048, “Unit 2 Steam Generator Narrow Range Level Control Room Indication Loop 2L-462 Uncertainty,” Revision 0; and
- SP 1126, “Turbine Building Cooling Water Header Isolation Safety Injection Relays 1SI-12X and 1SI-22X Refueling Test,” Revision 2.

b. Findings

No findings of significance were identified.

## 1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the design documents and observed the post-installation testing of a modification to the 1T1 motor control center (MCC) in accordance with Design Change 99EB01, “MCC 1T1/1T2 Transfer Switch,” and WO 0010297, “MCC 1T1 - Isolate, Disconnect, and Reland at Transfer Switch.” The inspectors verified that the design bases, licensing bases, and performance capability of risk significant systems had not been degraded through the modification and that performance of the modification and subsequent testing did not place the plant in an unsafe condition.

As part of this inspection, the inspectors also reviewed the following documents:

- CR 20010357, “Penetration 2603 Flammastic Cracked During Manipulation of Cables 2 12G-1/2”; and
- Prairie Island Calculation ENG-EE-136, “Cable Sizing and Transfer Switch Adequacy for 1T1 and 1T2 for Project 99EB01,” Revision 0.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance testing activities to ensure that the testing adequately verified system operability and functional capability. The post-maintenance testing activities were selected based on the respective system's importance to mitigating core damage or protecting barrier integrity.

The inspectors observed post-maintenance testing associated with the following work:

- testing of motor-operated valve MV-32144 in accordance with WO 0007136, "PM 32144 Loop A/B Cooling Water Header Crossover Motor-Operated Valve A D70.1 PM," and Maintenance Procedure D70.1, "Valve Testing Using VOTES," Revision 6, following rewiring in accordance with WO 9901008, "MV-32144-Transfer Source to MCC 1T1-A3" as part of Design Change 98EB02, "Cooling Water System Common Unit Motor Operated Valves";
- testing of 22 auxiliary feedwater overspeed trip tappet following repair in accordance with WO 0100563, "22 Auxiliary Feedwater Pump Trip Tappet Did Not Reset Normally";
- testing of the Unit 2 safety injection (SI) system after WO 0003399, "Install New SI Mini Recirculation Line Flowmeters in Unit 2"; and
- testing of newly installed solenoid-operated valves in accordance with WO 0004470, "Replace Reactor Coolant Gas Vent System Valves."

As part of these inspections, the inspectors reviewed the following additional documents:

- CR 20010380, "Two Motor Control Center Cubicles Found on 1T1 Without Mechanical Interlocks - Determine Scope and Resolution of Problem";
- CR 20010340, "Assess Communication to Operators Regarding Operation with SI Mini-Flow Alarm Out-of-Service";
- CR 20010344, "Operators Were Not Given Adequate Guidance for Operation With SI Mini-Flow Out-of-Service - Not Adequate Guidance for Post-Maintenance Testing";
- CR 20010354, "Equipment Problems Were Encountered During Installation of the SI Mini Recirculation Flowmeters";
- CR 20010974, "After Test of 22 Auxiliary Feedwater Pump Trip Throttle Valve the Trip Tappet Did Not Drop Down by Itself when Resetting";
- SP 2102, "22 Turbine Driven Auxiliary Feedwater Pump Monthly Test," Revision 65;
- SP 2088, "Safety Injection Pumps Monthly Test," Revision 40; and
- SP 1248, "Reactor Coolant Gas Vent System Test Each Cold Shutdown," Revisions 11 and 12.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors observed activities associated with the Unit 1 refueling outage that began on January 20, 2001. The inspectors reviewed the reactor cooldown rate, configuration management, clearance activities, reduced RCS inventory conditions, and refueling operations for management of risk, conformance to applicable procedures, and compliance with Technical Specifications. The following major activities were observed:

- RCS to 350 degrees Fahrenheit and placing of the residual heat removal (RHR) system into service;
- filling the pressurizer and cooling down with RHR;
- draining the RCS from the reactor vessel flange to the hot leg elevations;
- fuel offload to the spent fuel pool; and
- fuel loading into the reactor.

In addition to attending several outage planning meeting and pre-evolution briefings, the inspectors also reviewed the following documents:

- Unit 1 Shutdown Safety Assessment, January-February 2001;
- Operating Procedure 1C1.2, “Unit 1 Startup Procedure,” Revision 23;
- Operating Procedure 1C1.3, “Unit 1 Shutdown,” Revision 45;
- Special Operations Procedure 1C1.6, “Shutdown Operation - Unit 1,” Revision 12;
- Operating Procedure 1C4.1, “RCS Inventory Control - Pre-Refueling,” Revision 10;
- Maintenance Procedure 1D2, “RCS Reduced Inventory Operation,” Revision 9;
- Special Operations Procedure 1D2.1, “RCS Reduced Inventory Operation After Pool Flood,” Revision 10;
- Special Operations Procedure D5.2, “Reactor Refueling Operations,” Revision 26;
- Safety Evaluation Screening SES-821, “Installation of Bypass Jumper Across Transfer System Upender Frame Down Limit Contact”;
- CR 20011257, “Discovered RCS Draining to Sump A Via Vent and Drain Valves Opened for CV-31329 Replacement WO 9912283”;
- CR 20011587, “Three Incidents Involving SP 1366, Charging Pump Suction Check Valve Test, WO 0004879”;
- CR 20011601, “Loss of RCS Inventory at 1 Foot Below Flange Level - Found VC-16-17 Reactor Coolant Pump Leal Water Injection Line Drain Open”;
- CR 20011602, “An Unplanned ORANGE Condition was Entered for Containment Closure During Performance of WO 0100707”;
- CR 20011608, “Conduct a Common Cause Analysis of Outage Configuration Control”;
- CR 20011609, “Reduced Inventory Hold Pending Assessment of Readiness”;
- CR 20011638, “Valve Found Mispositioned - VC-1-1, MV-32060 Charging Pump Suction from RWST Bypass - Following SP 1366 - Open - Should Have Been Closed”; and

- CR 20010854, “Reactor Side Frame Down Limit Failed, Causing Carriage to Become Disengaged from Upender.”

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors verified, by witnessing selected surveillance testing and reviewing test data, that the equipment tested by the SPs met Technical Specifications, the Updated Safety Analysis Report, Design Basis Documents, and licensee procedural requirements, and demonstrated that the equipment was capable of performing its intended safety functions. The following tests were evaluated:

- SP 2305, “D6 Diesel Generator Monthly Slow Start Test,” Revision 14; and
- SP 1083, “Integrated SI Test with a Simulated Loss of Offsite Power,” Revision 26.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors verified that the performance indicator data submitted by the licensee were accurate and complete for the safety system unavailability performance indicators for the auxiliary feedwater system and the high pressure safety injection system. Control room logs and computerized limiting conditions for operations logs were reviewed for the periods of January 2000 through December 2000 to verify that the licensee had reported all unavailability for those four quarters.

As part of this inspection, the inspectors also reviewed CR 20010564, “For the 4th Quarter 2000 NRC Performance Indicators, Two Mitigating Systems Went WHITE (Unit 1 Residual Heat Removal and Unit 2 Safety Injection).”

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

a. Inspection Scope

The inspectors reviewed Licensee Event Report (LER) 50-282/2000-005-00; 50-306/2000-005-00, “Failure to Test Cooling (Service) Water Strainer Backwash Valves Due to Inadequate Surveillance Procedure.”

b. Findings

No additional findings of significance were identified in this inspection. This issue was first identified by the NRC during a Safety System Design and Performance Capability Inspection as discussed in Inspection Report 50-282/00-13(DRS); 50-306/00-13(DRS).

4OA6 Meeting(s)

Exit Meeting

The inspectors presented the inspection results to Mr. J. Sorensen and other members of licensee management on February 22, 2001. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

## PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Allen, General Superintendent Engineering, Nuclear Generation Services
T. Amundson, General Superintendent Engineering
T. Breene, Manager Nuclear Performance Assessment
L. Gard, General Superintendent Plant Maintenance
A. Johnson, General Superintendent Radiation Protection and Chemistry
T. Silverberg, General Superintendent Plant Operations
M. Sleigh, Superintendent Security
J. Sorensen, Site Vice President
M. Werner, Interim Plant Manager

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 50-282/01-02-01(DRP); 50-306/01-02-01(DRP) (EA-01-050) | NCV | Risk Assessment and Control Associated With Unit 1 Bus 15 Outage (Section 1R13.1) |

Discussed

| | | |
|---|---|---|
| 50-282/2000-005-00; 50-306/2000-005-00 | LER | Failure to Test Cooling (Service) Water Strainer Backwash Valves Due to Inadequate Surveillance Procedure (Section 4OA3) |