April 25, 2000

Mr. M. Wadley
President, Nuclear Generation
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

SUBJECT: PRAIRIE ISLAND INSPECTION REPORT 50-282/2000004(DRP); 50-306/2000004(DRP)

Dear Mr. Wadley:

On April 1, 2000, the NRC completed a baseline inspection at your Prairie Island Nuclear Generating Plant. The results of this inspection were discussed on April 6, 2000, with Mr. J. Sorensen and other members of your staff. The enclosed report presents the results of that inspection.

The inspection was an examination by the resident inspectors and regional inspectors of activities conducted under your license as they relate to reactor safety, radiation safety, verification of performance indicators, and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

During this inspection, one issue of very low safety significance involving coordination of work plans between onsite and offsite organizations and two issues of very low safety significance involving radiological controls of a high radiation area were identified and are discussed in the summary of findings and in the body of the attached inspection report.

The two radiological control issues were considered violations of NRC requirements, but because of their very low safety significance, the violations are not cited. If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with a copy to the Regional Administrator, Region III; the Director, Office of Enforcement, Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Prairie Island.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one, will be placed in the NRC Public Electronic Reading Room (PERR) link at the NRC homepage, http://www.nrc.gov/NRC/ADAMS/index.html.

Sincerely,

/RA/

Roger D. Lanksbury, Chief
Reactor Projects Branch 5

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

Enclosure: Inspection Report 50-282/2000004(DRP);
50-306/2000004(DRP)

cc w/encl: Site General Manager, Prairie Island
Plant Manager, Prairie Island
S. Minn, Commissioner, Minnesota
Department of Public Service
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community

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Reactor Projects Branch 5

Docket Nos. 50-282, 50-306
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Enclosure: Inspection Report 50-282/2000004(DRP); 50-306/2000004(DRP)

cc w/encl: Site General Manager, Prairie Island
Plant Manager, Prairie Island
S. Minn, Commissioner, Minnesota Department of Public Service
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community

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REGION III

Docket Nos:	50-282, 50-306
License Nos:	DPR-42, DPR-60
Report No:	50-282/2000004(DRP); 50-306/2000004(DRP)
Licensee:	Northern States Power Company
Facility:	Prairie Island Nuclear Generating Plant
Location:	1717 Wakonade Drive East Welch, MN 55089
Dates:	February 16, 2000, through April 1, 2000
Inspectors:	S. Ray, Senior Resident Inspector S. Thomas, Resident Inspector S. Burton, Senior Resident Inspector, Monticello P. Lougheed, Reactor Engineer R. Powell, Resident Inspector, Point Beach A. Stone, Operator License Examiner
Approved by:	Roger D. Lanksbury, Chief Reactor Projects Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

Prairie Island Nuclear Generating Plant, Units 1 & 2
NRC Inspection Report 50-282/2000004(DRP); 50-306/2000004(DRP)

The report covers a 6½-week period of resident and regional inspector inspection.

The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process in draft Inspection Manual Chapter 0609.

Cornerstone: Initiating Events

- GREEN. On February 21, 2000, the licensee removed one of the four 345,000-volt alternating current offsite power lines from service while the D1 emergency diesel generator was already out of service. The licensee received little advance notice from the licensee's offsite organization performing the work on the power line and did not have time to completely coordinate and evaluate the risk of the combination of degraded offsite power capability and degraded emergency onsite power capability before initiating the isolation of the offsite source. Although the licensee's offsite organization characterized the work to the plant operators as emergency work, the inspectors and plant staff later determined that the work was not an emergency and had apparently been planned by the licensee's offsite organization for some time.

  Using draft NRC Inspection Manual 0609, "Significance Determination Process," the NRC determined that the issue was of very low risk significance because of the robust nature of the offsite power sources and capabilities for cross feeding electrical power from one unit's diesel generators to the opposite unit's emergency buses. Therefore, the issue was determined to be within the licensee response band. The finding was assigned to Unit 1. (Section 1R03.1)

Cornerstone: Occupational Radiation Safety

- GREEN. On March 27, 2000, the licensee sluiced resin from the 21 evaporator feed ion exchanger, the 21 evaporator condensate ion exchanger, and the 11 evaporator ion exchanger to a low-level resin liner located in the spent resin decontamination pit. The post-sluicing radiation surveys of the spent resin decontamination pit were not begun until 3 hours after the completion of the evolution, after a licensee intern engineer questioned whether a survey had been performed. A survey was completed approximately 4 hours after the sluicing evolution was finished and revealed radiation exposure levels between 1000 and 1100 millirem per hour at 30 centimeters from the spent resin liner.

  The inspectors performed a risk significance determination of this issue using the Occupational Radiation Safety Significance Determination Flowchart in accordance with draft NRC Inspection Manual 0609, "Significance Determination Process." Since no unintended personnel exposure occurred as a result and there was no substantial potential for overexposure to occur, this finding was considered to be of very low risk significance and within the licensee response band. The finding was assigned to both Unit 1 and Unit 2. This issue was determined to be a Non-Cited Violation (NCV) for improper procedure implementation. The tracking number for this NCV is 50-282/2000004-01(DRP); 50-306/2000004-01(DRP). (Section 2OS1)

- GREEN. On March 27, 2000, subsequent to a resin sluicing evolution, the licensee failed to properly control the access to the spent resin decontamination pit, which had become a high radiation area requiring locked or guarded doors, as a result of the sluicing operation. Access to this area remained unlocked and unguarded for approximately 20 hours after the survey results indicated that it had become a locked high radiation area until identified by the licensee.

  The inspectors performed a risk significance determination of this issue using the Occupational Radiation Safety Significance Determination Flowchart in accordance with draft NRC Inspection Manual 0609, “Significance Determination Process.” Since no unintended personnel exposure occurred as a result and there was no substantial potential for overexposure to occur, this finding was considered to be of very low risk significance and within the licensee response band. The finding was assigned to both Unit 1 and Unit 2. This issue was determined to be a Non-Cited Violation (NCV) for the failure to control the access to a high radiation area as required by Technical Specifications. The tracking number for this NCV is 50-282/2000004-02(DRP); 50-306/2000004-02(DRP). (Section 2OS1)

Report Details

Both units operated at or near full power for the entire inspection period except that Unit 1 was reduced to about 40 percent power on March 11-12, 2000, for turbine valve testing and condenser cleaning and Unit 2 was reduced to about 45 percent power on March 18-19, 2000, for turbine valve testing.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R03 Emergent Work

.1 Lack of Coordination Between Work on Offsite Power Line and Plant Emergency Diesel Generator

a. Inspection Scope

The inspectors reviewed the circumstances leading to the licensee removing an offsite power line from service on February 21, 2000, at the same time that an emergency diesel generator was out of service for planned maintenance. As part of this inspection, the following documents were reviewed:

- Weekly Planning Meeting Results 2/19/00-2/25/00;
- Updated Safety Analysis Report, Section 8.2, "Transmission System," Revision 20;
- Updated Safety Analysis Report, Section 14.4.11, "Loss of All AC [alternating current] Power to the Station Auxiliaries," Revision 20;
- Request for Outage (RFO) Number 132693, dated February 21, 2000; and
- Condition Report 20000462, "Substation Isolation of Red Rock 2 Line While D1 Emergency Diesel Generator Out of Service Without Risk Evaluation."

b. Observations and Findings

The inspectors determined that the licensee did not have an opportunity to adequately plan, schedule, and evaluate the risks associated with the combination of emergent work on the Red Rock #2, 345,000-volt AC, offsite power line and the Unit 1 D1 emergency diesel generator. Both jobs ended up being conducted simultaneously which increased the risk of a loss of offsite power at the same time that one of the primary mitigating systems for that initiating event was out of service.

On February 20, 2000, the D1 diesel generator was removed from service for planned preventive maintenance. The work was expected to last about four days. In the late afternoon of February 21, 2000, the licensee received RFO Number 132693 from the offsite Operation and Maintenance Support - Electrical group to isolate the Red Rock #2 345,000-volt AC offsite power line, one of four 345,000-volt AC lines connected to the plant switchyard. The operators realized that, with the diesel generator out of service, the isolation would increase the core damage risk to the plant, but the RFO was characterized as an emergency by the licensee's offsite organization because it was for replacing a burnt power pole.

In order to reduce the overall risk, operators waited to isolate the offsite power line until a screenhouse fan, which affected operability of two of the cooling water pumps, was returned to service. However, neither the operators nor the licensee's substation coordinator consulted anyone from the probabilistic risk assessment group nor did they determine whether the power line work was truly an emergency before taking the line out of service on the morning of February 21, 2000.

The inspectors reviewed the RFO and noted that the Mid-continent Area Power Pool had been notified of the pending power line work on February 9, 2000, and that the procedure for the work had been written and approved on February 16 and 19 respectively. The licensee's offsite group responsible for the work on the power line had not notified plant personnel of the pending job. The substation coordinator stated that company policy was that RFOs would be presented to the plant no later than noon of the day before the outage. The policy was not met in this case. The licensee entered the issue into its corrective action program as Condition Report 20000462.

The configuration affected both the initiating event frequency and mitigating systems for a loss of offsite power event. Therefore, the inspectors evaluated the risk significance using Phase 2 of the Significance Determination Process described in draft NRC Manual Chapter 0609. The inspectors used the conservative assumption that the loss of offsite power frequency was increased by a factor of ten during the 16-17 hours that the offsite power line was isolated. This placed the event frequency in Row I of Table 1, "Estimated Likelihood for Initiating Event Occurrence During Degraded Period." This was a very conservative assumption because the probability of simultaneously losing all of the three remaining 345,000-volt AC offsite power lines was quite low considering that there was no adverse weather conditions.

The most limiting core damage sequence was determined to be a loss of offsite power with a failure of the emergency power system and failure to restore offsite power within 5 hours. The remaining mitigation capability for that sequence consisted of the D2 diesel generator, the ability to cross-connect either of the two safeguards electrical divisions on Unit 1 to the Unit 2 diesel generators using operator action, and the use of operator action under high stress to restore offsite power. This placed the issue in Row C, Column 5, of Table 2, "Risk Significance Estimation Matrix." Therefore, the issue was determined to be in the licensee response band (GREEN) and was considered to be a finding of very low risk significance. The finding was assigned to Unit 1.

Technical Specification (TS) Limiting Conditions for Operation for both diesel generators and offsite power sources were met throughout the period. Thus, no violation of regulatory requirements was identified.

.2 Other Emergent Work

a. Inspection Scope

The inspectors reviewed and observed the following emergent work activities that involved risk significant systems and/or required coordination with other scheduled risk significant work:

- WO [Work Order] 0000752, “Replace Diaphragm on VC-15-35"; and
- WO 0001214, “21 Auxiliary Feedwater Pump Outboard Pump Packing Failed.”

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial walkdown of the 22 diesel-driven cooling water pump and the 121 motor-driven cooling water pump while the 12 diesel-driven cooling water pump was out-of-service for planned maintenance. The system was selected due to the increased core damage frequency with the 12 pump not available.

The inspectors reviewed the following documents as part of this walkdown:

- Operating Procedure 2C20.5, “Unit 2 - 4.16KV [kilovolts] System,” Revision 14;
- Operating Procedure C35, “Cooling Water System,” Revision 34;
- Integrated Checklist C1.1.35-1, “Cooling Water System Unit 1,” Revision 7;
- Integrated Checklist C1.1.35-2, “Cooling Water System Unit 2,” Revision 7; and
- Drawing NF-39216-1, “Flow Diagram Unit 1 & 2 Cooling Water - Screenhouse,” Revision X.

b. Observations and Findings

The were no findings identified and documented during this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed a walkdown of fire areas 59 and 84 located in the Unit 1 auxiliary building, 715-foot elevation. This area was chosen because it contained a large number of electrical supplies and cables for risk significant equipment. In order to perform this walkdown, the inspectors referred to Plant Safety Procedure F5, Appendix A, “Fire Strategies,” Revision 5.

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed licensee analysis and procedures for protection of equipment during both internal and external flooding conditions. The inspectors also reviewed whether recommendations identified in the licensee’s Individual Plant Examination for internal flooding had been implemented. The following documents were reviewed:

- Prairie Island Nuclear Generating Plant Individual Plant Examination NSPLMI-94001, Revision 0;
- Prairie Island Nuclear Generating Plant Individual Plant Examination of External Events NSPLMI-96001, Revision 1;
- Prairie Island Updated Safety Analysis Report, Appendix F, “Probable Maximum Flood Study Mississippi River at Prairie Island, Minnesota,” Revision 4;
- Prairie Island Updated Safety Analysis Report, Appendix I, Section I.4.4, “Flooding,” Revision 14;
- TSs 5.1, “Site Design Features,” Revision 147;
- Abnormal Procedure AB-4, “Flood,” Revision 15;
- Operating Procedure C34, “Station Air System,” Revision 15;
- Abnormal Operating Procedure C34 AOP1, “Loss of Instrument Air,” Revision 9;
- Abnormal Operating Procedure C35 AOP2, “Loss of Pumping Capacity or Supply Header Without SI [safety injection],” Revision 5;
- Surveillance Procedure (SP) 1293, “Flood Preparation-Flood Control Panel Inspection/Installation,” Revision 6;
- Preventive Maintenance Procedure (PM) 3505-1-121, “121 Instrument Air Compressor 1000 Hour Inspection,” Revision 11;
- Drawing NF-120723, “Wiring Diagram B26 Load Sequencer Cabinet,” Revision A; and

- Drawing Series X-HIAW-2713-10, “Safeguard Load Sequencer,” various revisions.

In addition, the inspectors conducted a walkdown of the external flood protection features of the turbine, auxiliary, and service buildings and a walkdown of internal flooding issues in the auxiliary feedwater (AFW) pump room.

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R09 Inservice Testing of Pumps and Valves

a. Inspection Scope

The inspectors reviewed and observed the inservice testing for risk significant mitigating systems. These systems were selected based on their respective importance ranking as described in the licensee’s Probabilistic Risk Assessment. The tests observed were:

- SP 1358 [2358], “VC-2-2 [2VC-2-2], RWST [Refueling Water Storage Tank] to Charging Pumps Check Valve Quarterly Test,” Revision 7 [3];

- SP 2100, “21 Motor Driven AFW Pump Monthly Test,” Revision 55; and

- SP 1106B, “22 Diesel Cooling Water Pump Test,” Revision 53.

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee implementation of the maintenance rule requirements for the following systems:

- nuclear instrumentation;

- D3/D4 nonsafeguards diesel generators; and

- fuel oil.

The systems were selected based on their being designated as risk significant under the Maintenance Rule, or their being in the increased monitoring (Maintenance Rule category a(1)) group. The inspectors reviewed the Forth Quarter Equipment Performance Report, dated February 4, 2000, as well as applicable system work orders and condition reports as part of these inspections.

Observations and Findings

There were no findings identified and documented during this inspection.

1R13 Maintenance Work Prioritization and Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk, scheduling, configuration control, and performance of simultaneous preventive maintenance on the 11 safeguards screenhouse roof exhaust fan breaker in accordance with WO 9800086, "Perform Breaker Electrical PM In Accordance With MCC [motor control center] G7," and the D1 diesel generator in accordance with PM 3001-2-D1, "D1 Diesel Generator 18 Month Inspection," Revision 16. The inspectors chose to evaluate these maintenance activities based on the moderately high combined risk as discussed in the Weekly Planning Meeting Results, Unit 1 Risk Profile, February 19 - February 25, 2000.

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following condition reports (CRs) containing operability evaluations involving risk significant mitigating systems:

- CR 20000527, "Nonsafety-related Washers Used in a Safety-Related Application";
- CR 20000571, "Used Lockwashers in Damper Supports for CD-34142"; and
- CR 20000592, "21 Motor Driven AFW Pump Developed a Steam Leak at the Outboard Pump Seal During Performance of SP 2100."

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors evaluated the following operator workarounds (OWAs):

- OWA 19950907, "Unit 1 and Unit 2 Reactor Coolant System Vent System Solenoid Valves Leak By Causing Alarms and Pressure Indications in the Control Room";

- OWA 19983388, “The Reliability of the Intake Screenhouse Bypass Gates is Suspect”;

- OWA 19983390, “Unit 1 and Unit 2 Allowable Steam Generator Level at Low Load/No Load Conditions are Too Restrictive;” and

- OWA 19992527, “Boric Acid Heat Trace Local Annunciator 47015-0102 Alarms Frequently and is a Distraction to the Operators.”

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R17 Permanent Plant Modifications

.1 Design Changes to the Component Cooling System and the Spent Fuel Heat Exchangers

a. Inspection Scope

The inspectors reviewed the schedule, status, and current work for the on-line installation of modifications to the component cooling water supply to the spent fuel heat exchangers and replacement of one of the heat exchangers in accordance with Design Change 99CC01, “Component Cooling Leakage Modification,” and 99SF02, “Replace 122 Spent Fuel Pool Heat Exchanger.”

b. Observations and Findings

There were no findings identified and documented during this inspection. The inspectors intended to continue the inspection activities for these modifications until they are completed.

.2 Design Change to the Diesel Generator Output Breakers

a. Inspection Scope

The inspectors reviewed the project description, safety assessment, and the 50.59 screening associated with Design Change 97FP26, “Diesel Generator Source Breaker Modification.” The proposed design change would address potential hot short conditions which could cause the D1 and D5 diesel generator output breakers to spuriously operate. In addition to reviewing the documents mentioned above, the inspectors discussed the proposed design change with the cognizant system engineer.

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and observed the following post-maintenance testing activities involving risk significant mitigating system equipment:

- Diesel generator D1 testing performed in accordance with PM 3001-2-D1, "D1 Diesel Generator 18 Month Inspection," Revision 16;

- AFW pump 21 testing performed in accordance with WO 0001214, "21 Auxiliary Feedwater Pump Outboard Pump Packing Failed," and SP 2100, "21 Motor Driven AFW Pump Monthly Test," Revision 55; and

- Diesel generator D5 testing performed in accordance with WO 0001301, "Troubleshoot D5 Diesel Generator Loss of Voltage Regulation," and SP 2093, "D5 Diesel Generator Monthly Slow Start Test," Revision 65.

b. Observations and Findings

There were no findings identified and documented during this inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed the performance of the following surveillance testing on risk significant mitigating system equipment:

- SP 2218 [2219], "Monthly 4KV Bus 25 [26] Undervoltage Relay Test," Revision 31 [27];

- SP 1295, "D1 Diesel Generator 6 Month Fast Start Test," Revision 26; and

- SP 1168.8A, "Cooling Water Auxiliary Operating Pressure Test," Revision 1.

b. Observations and Findings

There were no findings identified and documented during this inspection.

Cornerstone: Emergency Preparedness

1EP1 Drill, Exercise, and Actual Events

a. Inspection Scope

The inspectors observed an operating crew during a simulator scenario involving an earthquake which included two opportunities for event classification and notification of offsite authorities. The scenario did not involve any protective action recommendations.

As part of this inspection, the inspectors reviewed Lesson Plan P9160S0–1, Simulator Exercise Guide 00-08, Revision 0.

b. Observations and Findings

There were no findings identified and documented during this inspection.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control

a. Inspection Scope

On March 28, 2000, the inspectors were notified by the licensee that, after a resin sluicing evolution, the spent resin decontamination pit area became a high radiation area requiring locked or continuously guarded doors and the proper access controls were not put in place for approximately 24 hours. The inspectors evaluated this event and the licensee's response.

b. Observations and Findings

The inspectors determined that the licensee had violated NRC regulations by failing to perform a timely radiation survey and by failing to control access to the high radiation area requiring locked or continuously guarded doors.

On March 27, 2000, the licensee sluiced resin from the 21 evaporator feed ion exchanger, the 21 evaporator condensate ion exchanger, and the 11 evaporator ion exchanger to a low-level resin liner located in the spent resin decontamination pit. The post-sluicing radiation surveys of the spent resin decontamination pit were not begun until 3 hours after the completion of the evolution, after a licensee intern engineer questioned whether the survey had been performed. The survey was completed approximately 4 hours after the sluicing evolution was finished and revealed radiation exposure levels between 1000 and 1100 millirem per hour at 30 centimeters from the spent resin liner, representing a locked high radiation area as defined by licensee procedures. Although these survey results were documented, the entrance to the pit remained unguarded and unlocked for approximately an additional 20 hours before the access to the area was secured.

Technical Specification 6.4.A required, in part, that written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, be established and implemented. Regulatory Guide 1.33 required that radiation protection procedures include provisions for conducting radiation surveys. Contrary to the requirements of TS 6.4.A, on March 27, 2000, surveys required by Radiation Protection Implementing Procedure 1721 "Resin Sluice," Revision 7, step 4.0, were not performed in a timely manner after the completion of the resin sluicing evolution. This violation is being treated as a Non-Cited Violation (NCV) consistent with Interim Enforcement Policy for

Pilot Plants, Appendix F of the NRC Enforcement Policy (50-282/2000004-01(DRP); 50-306/2000004-01(DRP)). This violation is in the licensee's corrective action program as Issue 20000843.

The inspectors performed a risk significance determination of the failure to perform a timely radiation survey using the Occupational Radiation Safety Significance Determination Flowchart in accordance with draft NRC Inspection Manual Chapter 0609, "Significance Determination Process." Since no unintended personnel exposure occurred as a result and there was no substantial potential for overexposure to occur, this finding was considered to be within the licensee response band (GREEN) and was assigned to the Occupational Radiation Safety Cornerstone for Unit 1 and Unit 2.

Technical Specification 6.7.B required, in part, that areas with radiation levels greater that 1000 millirem per hour at 30 centimeters be provided with locked or continuously guarded doors to prevent entry and that these doors remain locked except during periods of access by personnel under an approved radiation work permit. Contrary to the requirements of TS 6.7.B, on March 27 and 28, 2000, the access to the spent resin decontamination pit was left unlocked and unguarded for approximately 20 hours following determination that radiation exposure levels near the spent resin liner were in excess of 1000 millirem per hour. This violation is being treated as a Non-Cited Violation (NCV) consistent with Interim Enforcement Policy for Pilot Plants, Appendix F of the NRC Enforcement Policy (50-282/2000004-02(DRP); 50-306/2000004-02(DRP)). This violation is in the licensee's corrective action program as Issue 20000843.

The inspectors performed a risk significance determination of this issue using the Occupational Radiation Safety Significance Determination Flowchart in accordance with draft NRC Inspection Manual 0609, "Significance Determination Process." Since no unintended personnel exposure occurred as a result and there was no substantial potential for overexposure to occur, this finding was considered to be within the licensee response band (GREEN) and was assigned to the Occupational Radiation Safety Cornerstone for Unit 1 and Unit 2.

4. OTHER ACTIVITIES

4OA2 Performance Indicator Verification

Cornerstone: Mitigating Systems

.1 Safety System Unavailability, High Pressure Injection System

a. Inspection Scope

The inspectors verified the Safety System Unavailability, High Pressure Injection System Performance Indicator data reported by the licensee for October 1998 through December 1999 for Unit 1 and Unit 2. This was accomplished in part through evaluation of the Limiting Conditions for Operation Log times for the safety injection system and required support systems, review of applicable work orders, and discussions with licensee personnel.

b. Observations and Findings

There were no findings identified and documented during this inspection.

4OA4 Other

.1 (Closed) Inspection Followup Item (IFI) (50-282/98006-02(DRP); 50-306/98006-02(DRP)): Concerns Related to the Component Coolant System Flow Instrumentation. The inspectors noted that the inlet and outlet flow indicators for the spent fuel pool heat exchanger showed vastly different indications and that this deficiency was accepted by the operators. In addition, the inspectors were concerned that these flow indications were used in determining the flow balance within the component cooling system. The licensee initiated a modification to replace the flow indicators. With respect to flow balancing, the licensee indicated that flows to safety-related equipment were verified in the respective unit's component cooling system checklist. The inspectors had no further concerns.

.2 (Closed) Violation (VIO) 50-282/97290-01023(DRS); 50-306/97290-01023(DRS): Violation of 50.71(e) Involving Failure to Update the Updated Final Safety Analysis Report Auxiliary Feedwater Accident Flow Rates. The licensee incorporated the correct auxiliary feedwater flow rates into the updated final safety analysis report in Revision 16. Additionally, the licensee has undertaken an extensive updated final safety analysis report upgrade project. The inspectors had no concerns with the licensee's corrective actions to this violation.

.3 (Closed) VIO 50-282/97290-01043(DRS); 50-306/97290-01043(DRS): Violation of Corrective Action Involving Failure to Correct the Updated Final Safety Analysis Report Auxiliary Feedwater Flow Rates. In their response to the violation, the licensee committed to also resolve fourteen additional items which arose from the same design basis document verification effort that identified the auxiliary feedwater discrepancy. Following its response to the violation, the licensee added three more items to the issue. All but two of the seventeen items are completed. The remaining two items are associated with the high energy line break analysis and are covered under a separate corrective action tracking number from the original violation. The inspectors had no concerns with the licensee's corrective actions to this violation.

4OA5 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Sorensen and other members of licensee management on April 6, 2000. 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. Amundson, General Superintendent Engineering
L. Ganser, Acting Manager Nuclear Performance Assessment
J. Goldsmith, General Superintendent Engineering, Nuclear Generation Services
J. Gonyeau, Life Cycle and Management Support Engineer
A. Johnson, General Superintendent Radiation Protection and Chemistry
G. Lenertz, General Superintendent Plant Maintenance
D. Schuelke, Plant Manager
T. Silverberg, General Superintendent Plant Operations
M. Sleigh, Superintendent Security
J. Sorensen, Site General Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened — None.

Opened and Closed

50-282/2000004-01(DRP); 50-306/2000004-01(DRP)	NCV	Failure to follow procedures for performing timely radiation surveys of a spent resin liner following a contaminated resin sluice
50-282/2000004-02(DRP); 50-306/2000004-02(DRP)	NCV	Failure to lock a high radiation area as required by TS 6.7.B

Closed

50-282/98006-02(DRP); 50-306/98006-02(DRP)	IFI	Concerns Related to the Component Coolant System Flow Instrumentation
50-282/97290-01023(DRS); 50-306/97290-01023(DRS)	VIO	Violation of 50.71(e) Involving Failure to Update the Updated Final Safety Analysis Report Auxiliary Feedwater Accident Flow Rates
50-282/97290-01043(DRS); 50-306/97290-01043(DRS)	VIO	Violation of Corrective Action Involving Failure to Correct the Updated Final Safety Analysis Report Auxiliary Feedwater Flow Rates

LIST OF ACRONYMS USED

AC	Alternating Current
AFW	Auxiliary Feedwater
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
IFI	Inspection Followup Item
KV	Kilovolts
MCC	Motor Control Center
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OWA	Operator Workaround
PERR	Public Electronic Reading Room
PM	Preventive Maintenance Procedure
RFO	Request for Outage
RWST	Refueling Water Storage Tank
SP	Surveillance Test Procedure
VIO	Violation
WO	Work Order

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.