U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-282/91011(DRP); 50-306/91011(DRP)

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

Licensee: Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Facility Name: Prairie Island Nuclear Generating Plant

Inspection At: Prairie Island Site, Red Wing, MN

Inspection Conducted: May 11 through July 8, 1991

Inspectors: P. L. Hartmann H. B. Clayton R. L. Bywater

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Approved By: B. L. Jorgensen, Chief
Reactor Projects Section 2A

7/18/91
Date

Inspection Summary

Inspection on May 11 through July 8, 1991 (Reports No. 50-282/91011(DRP); 50-306/91011(DRP))
Areas Inspected: Routine unannounced inspection by resident inspectors of plant operational safety including onsite followup of events and outage activities, maintenance, surveillance, simulator/procedure evaluation, and design changes and modifications.
Results:

No violations of NRC requirements were identified in any of the five areas inspected.

Operations

Strengths were noted in the control of work activities and equipment availability to minimize shutdown risk. Management was closely involved in those efforts which included careful planning to limit the duration of time the plant was at mid-loop operations. Zero events were classified by the licensee as requiring notification of the NRC via the Emergency Notification System during the refueling outage period which is an improvement from past performance. The plant shutdown/cooldown and startup were well controlled and conservatively performed by the Operations staff. A questioning attitude utilized by personnel resulted in the curtailment of the system drain based on inaccurate instrumentation.

Maintenance and Surveillance

Extensive maintenance and surveillance activities took place during the inspection period due to the refueling outage. All observed activities were performed adequately. Additional inspection by regional inspectors was accomplished as follows: inservice inspection (reference Inspection Report 50-282/91009(DRS); 50-306/91009(DRS)) and containment integrated leak rate inspection (reference Inspection Report 50-282/91012 (DRS); 50-306/91012(DRS)).

Engineering and Technical Support

Engineering effort included response to balance of plant annunciation, fuel sipping and core reload efforts, and in particular onshift involvement with core coolant inventory during mid-loop operations. The efforts involved careful attention to reactor safety.

Safety Assessment/Quality Verification

The licensee demonstrated conservative judgement in resolving safety issues which came up during this inspection period. Management was actively involved in evaluating those issues.

The corporate quality assurance group conducted an outage audit which will be reviewed by inspectors once the audit report is completed.

Emergency Preparedness

A Notice of Unusual Event was declared on May 14, 1991, due to loss of control room alarm indications. The plant response was appropriate and timely.

Radiation Protection

Radiological control during the period was good. The ALARA controls for the refueling outage continued to be well planned and coordinated. A detailed inspection by a regional specialist was conducted (reference Inspection Report 50-282/91013(DRSS); 50-306/91013(DRSS).

DETAILS

1. Persons Contacted

E. Watzl, General Manager, Prairie Island
#M. Sellman, Plant Manager
#D. Mendele, General Superintendent, Engineering and Radiation Protection
#M. Wadley, General Superintendent, Operations
G. Lenertz, General Superintendent, Maintenance
#R. Lindsey, Assistant to the Plant Manager
#D. Schuelke, Superintendent, Radiation Protection
G. Miller, Superintendent, Operations Engineering
#M. Reddemann, Superintendent, Technical Engineering
T. Breene, Superintendent, Nuclear Engineering
M. Klee, Superintendent, Quality Engineering
R. Conklin, Supervisor, Security and Services
#G. Eckholt, Nuclear Support Services
J. Leveille, Nuclear Support Services
A. Hunstad, Staff Engineer
M. Brossart, Nuclear Engineering

#Denotes those present at the management interview of July 12, 1991.

2. Operational Safety Verification (71707, 93702)

The licensee began a 29 day refueling outage on May 31, 1991, on Unit 1. The shutdown and cooldown of the unit were uneventful with the exception of securing the drain down off the pressurizer when operators concluded the cold calibrated pressurizer level did not appear to be indicating accurately. This questioning attitude led to prevention of further complications by an unplanned inventory loss. Operations personnel continued the cooldown utilizing other pressurizer level indications without incident. Startup commenced on June 29, 1991 with criticality reached at 11:07 a.m., and the unit placed on line on June 30, 1991. Unit 2 operated at full power the entire inspection period.

a. Operational Safety Verification (71707)

The inspector observed control room operations, reviewed applicable logs, conducted discussions with control room operators and observed shift turnovers. The inspector verified operability of selected emergency systems, reviewed equipment control records, and verified the proper return to service of affected components, conducted tours of the auxiliary building, turbine building and external areas of the plant to observe plant equipment conditions, including potential fire hazards, and to verify that maintenance work requests had been initiated for the equipment in need of repairs.

b. Onsite Followup of Events (93702)

On May 14, 1991, with both units at full power, the licensee declared a Notification of Unusual Event (NUE) at 8:58 a.m. The NUE

was based on "nonfunctional alarms in the control room" following loss of Balance-of-Plant (BOP) Annunciators and loss of the Emergency Response Computer System (ERCS).

The initiating event was evaluated and the appropriate emergency classification was made within 17 minutes. Notifications to offsite state and county agencies were completed within the required 15 minutes.

The event was caused by loss of power to the operating Computer Processing Unit (CPU) which occurred when inverter power was unintentionally interrupted to the A CPU for Unit 1 ERCS, with subsequent failure of the B CPU to pick up computer loads properly.

Following prompt restoration of ERCS, the NUE was terminated at 9:32 a.m. The inspector investigated the apparent root cause of the event which was personnel error. The inspector interviewed the operator performing the electrical isolation of 33 inverter and reviewed the isolation procedure. The 33 inverter normal power was to be isolated, and inverter loads powered by the alternate supply via a "maintenance" bypass breaker. The work request procedure listed the order of breaker positioning correctly; however, the nomenclature and labeling of the associated breakers is confusing since two breakers have bypass designations. The licensee is conducting an error reduction task force assessment of the event. Their assessment and correction actions in response to this event will be reviewed in a future report.

On May 25, 1991, at 11:25 a.m. the licensee discovered that the ventilation system noble gas radiation monitor alarm for the radioactive waste building had been bypassed for about 13 hours. The radiation monitor is required by Technical Specification 3.9.F. The inspectors discussed this event with licensee personnel and determined that no radioactive noble gases had been discharged through the ventilation system during the time that the alarm was bypassed. The inspectors discussed this event with a Region III health physics inspector who plans to review the event during a separate inspection.

On May 28, 1991, at 1:50 p.m. a personnel error caused the 122 Auxiliary Building Special Ventilation System to start automatically. A radiation protection technician who intended to test radiation monitor IR37 tested IR30 instead. The test input caused the automatic start. The inspectors will further review Licensee Event Report 50-282/91007 associated with the event.

The inspectors will review the licensee root cause analyses and corrective actions in response to the above three events. Each involves personnel error and human factors considerations. Inspector review of licensee root cause analysis and corrective action will be tracked as an open item: 50-282/91011-01.

At 5:27 p.m. on June 5, 1991, while Unit 1 was in cold shutdown, an unplanned increase in residual heat removal (RHR) system flow occurred. Air operated throttle valve 31236 failed open, which resulted in maximum RHR flow through the B train. In response, operators entered abnormal operating procedure 1C15AOP3 (Rev. 0) "Residual Heat Removal System Local Operation" upon loss of control room instrumentation/flow and/or control. This procedure provides for reestablishing throttled RHR flow through the affected train of RHR, by using motor valve (MOV) 32066, which is further upstream. Since this MOV normally provides only for full open or closed throttled position, this valve is placed in a mid position by remote operation at the motor control center (MCC) breaker. The inspectors determined by interviewing involved personnel that MV 32066 was locally operated (closed, then timed open to a mid-position) as a controlled evolution, and RHR flow was reestablished. By review of system parameters, the inspectors confirmed the licensee conclusion that flow was secured for about three minutes and indicated RHR suction temperature increased about 15 degrees F, to a peak of 127 degrees F. The A train of RHR was available but not used in accordance with procedure.

The licensee did not initially inform the inspectors of the flow transient, they learned of it by later review of logs. Following discussions with the inspectors, the licensee revised administrative procedures to require inspector notification upon entry into abnormal operating procedures.

The inspectors discussed the RHR valve failure with the system engineer who investigated the failure. The copper air supply tubing, which broke diagonally, appeared to have been the original tubing installed. This tubing was replaced. The licensee has not yet completed its evaluation of the failure or its long term correction action. The licensee is considering generic implications of the event.

c. Outage Activities (71707, 60710)

The inspectors observed outage activities such as fuel movements, reactor reassembly, pre-operational testing, and portions of the plant's return to power. The inspectors also attended outage planning meetings to ascertain whether work was coordinated such that required systems remained operable and shutdown risk was minimized.

The inspectors verified that the licensee appropriately reviewed maintenance activities that could impact mid-loop operations. The use of mid-loop operations was carefully planned, resulting in a minimum time period using mid-loop operations. The inspectors observed strong technical support and management control of mid-loop activities.

During refueling operation, fuel handling was stopped due to refueling containment integrity questions regarding penetrations 37 A and B and 38 A and B. The licensee placed the isolation valves for these positions in the closed position and resumed fuel handling, which was conservative. Following this action, the licensee, the inspectors and NRR representatives discussed the interpretation of Technical Specification requirements of Containment Integrity and Refueling Integrity. Due to the lack of specificity for refueling integrity, the licensee agreed to submit a license amendment to clarify this requirement.

No violations, deviations, or unresolved items were identified. One open item was identified.

3. Maintenance Observation (71707, 37700, 62703)

Routine, preventive, and corrective maintenance activities were observed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, industry codes or standards, and in conformance with Technical Specifications. The following items were considered during this review: adherence to limiting conditions for operation while components or systems were removed from service, approvals were obtained prior to initiating the work, activities were accomplished using approved procedures and were inspected as applicable, functional testing and/or calibrations were performed prior to returning components or systems to service, quality control records were maintained, activities were accomplished by qualified personnel, radiological controls were implemented, and fire prevention controls were implemented.

Portions of the following maintenance activities were observed during the inspection period:

- Installation of Auxiliary Feedwater (AFW) System Flushing Tees
- Repair of Leak from AFW Turbine Steam Admission Valve (CV 31998)
- Modification of Unit 1 Balance-of-Plant Annunciator System
- Inspection and Cleaning of Component Cooling Water Heat Exchanger Internals
- Repair of Unit 1 Fan Coil Unit Outlet Isolation Valve
- Boric Acid Residue Removal From Unit 1 Safety Injection System Flow Elements
- Repair of RHR Motor Valve 32236
- Repair to Unit 1 Balance-of-Plant Annunciation System (following startup)

No violations, deviations, unresolved or open items were identified.

4. Surveillance (61726, 71707)

The inspector reviewed Technical Specifications required surveillance testing as described below and verified that testing was performed in accordance with adequate procedures. Additionally, test instrumentation was calibrated, Limiting Conditions for Operation were met, removal and restoration of the affected components were properly accomplished, and

test results conformed with Technical Specifications and procedure requirements. The results were reviewed by personnel other than the individual directing the test and deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

Portions of the following test activities were observed or reviewed:

SP2093	D2 Diesel Generator Slow Start
SP1071	Containment Integrated Leakage Rate Test. Region III inspectors also inspected this activity and the results are documented in a separate report.
SP2001aa	Reactor Coolant System Leakage Test.
SP1106A	12 Diesel Cooling Water Pump Test

No violations, deviations, unresolved or open items were identified.

5. Simulator/Procedure Evaluation (42700)

During the period June 4-6, 1991, a group of NRC inspectors, including the NRC Region III Projects Branch Chief and Section Chief - each of whom has emergency response duties associated with the Prairie Island Nuclear Generating Plant - joined the Prairie Island Resident and Senior Resident inspectors for an evaluation of the plant control room simulator and selected procedures.

The selected procedures and simulator operations were used to familiarize the NRC team with plant behavior during certain evolutions. Additionally, the effectiveness of the procedures applying to the circumstances was assessed. Licensee support consisted of two professional staff trainers and free access to the simulator during the three day evaluation. The trainers operated the simulator scenarios and provided guidance on both hardware and procedures.

The following emergency operating procedures were covered by the evaluation:

1E-0	Reactor Trip or Safety Injection
1E-1	Loss of Reactor or Secondary Coolant
1ES-0.1	Reactor Trip Recovery
1ES-1.1	Post-LOCA Cooldown and Depressurization
1ECA-0.0	Loss of all AC Power
1ECA-3.1	Steam Generator Tube Rupture with Loss of Reactor Coolant: Subcooled Recovery
1FR-S.1	Response to Nuclear Power Generation/ATWS
1FR-Z.1	Response to High Containment Pressure
1F-0	The Critical Safety Function Status Trees
1F-0.1	Subcriticality

In addition, the following annunciator alarm response procedures were covered by the evaluation:

47010.0101	11 Feedwater Pump Locked Out
47011.0301	11 Steam Generator Level Deviation

47012.0408 Pressurizer Hi/Lo Pressure Channel Alert
47012.0503 Reactor Coolant System RTD Bypass Loops - Low Flow
47022.0106 Fire Detection Panel Alarm
47022.0108 Hi Radiation Train B

No substantive discrepancies were identified in the simulator or procedures; however, there were examples where human factors improvements could be made to enhance the response to transients:

- While there was some logic to the way controls were arranged on the control boards, there was no mimicking of systems. Mimicking can be effective in reducing errors by providing a visual reminder of a flow path.

- The plant Emergency Operation Procedures (EOPs) used equipment terminology from the Westinghouse Owners Group Emergency Response Guidelines, instead of the terminology used on the equipment labels. Examples of inconsistent terminology were the "Pressurizer PORVs and Block Valves" of the plant EOPs, while the control board labels were "PRZR Relief Vlv" and PRZR Relief Isol."

- There was no indication on the labels or control switches to remind the operators that some motor operated valve switches must be held in position in order to fully stroke the valves. The control switches for the auxiliary feedwater supply valves are examples (MV-32238, MV-32239, MV-32381, MV-32382).

Comments were provided to the licensee representatives for their consideration and, if deemed appropriate, further follow-up action.

No violations, deviations, unresolved, or open items were identified.

6. Design, Design Changes, and Modifications (37700)

The inspector reviewed plant design change packages and safety evaluations for the Unit 1 cycle 15 core reload modification (Modification Numbers 91L257 and 91L257, Rev. 1). Notable changes in the new core design from previous Unit 1 core designs include the insertion of fresh Westinghouse High Burnup Optimized Fuel Assemblies (HB-OFA) equipped with debris filter bottom nozzles (DFBN). NRC has approved a Westinghouse topical report specifying the requirements and evaluation of the HB-OFA (WCAP-10125-P-A). The DFBNs are designed to preclude debris from entering the core. Westinghouse has documented in a letter to the licensee that the DFBN is structurally and hydraulically equivalent to the previous nozzle design and can be considered a replacement part.

During fuel handling activities for core reload, the licensee used the "fuel sipping" technique on each fuel assembly from the previous cycle that was to be reinserted into the core. In this manner, the licensee identified that assembly Q-24, a twice-burned assembly originally inserted in cycle 13, was leaking. The licensee decided that a core redesign would be made to replace the leaking assembly and three additional assemblies. The four replacement assemblies were twice-burned

assemblies originally inserted in cycle 11. The replacement assemblies were visually inspected via periscope to be free from damage or debris. The licensee's safety evaluation of the redesigned core was acceptable and the plant design change was approved by the Operations Committee.

No violations, deviations, unresolved, or open items were identified.

7. Open Items

Open items are matters which have been discussed with the licensee, and will be reviewed further by the inspector. These involve some action on the part of the NRC or licensee or both. An open item identified during the inspection is discussed in Paragraph 2.b.

7. Management Interview (71707)

The inspectors met with the licensee representatives denoted in paragraph 1 at the conclusion of the report period on July 12, 1991. The inspectors discussed the purpose and scope of the inspection and the findings. The inspectors also discussed the likely information content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any documents or processes as proprietary.