U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-263/87012(DRP)

Docket No. 50-263 License No. DPR-22

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

Facility Name: Monticello Nuclear Generating Station

Inspection At: Monticello Site, Monticello, Minnesota

Inspection Conducted: August 25 through October 7, 1987

Inspectors: P. L. Hartmann

C. L. Vanderniet

Approved By: I. N. Jackiw, Chief
Reactor Projects Section 2C

10-30-87
Date

Inspection Summary

Inspection on August 25 through October 7, 1987 (Report No. 50-263/87012(DRP))
Areas Inspected: A routine, unannounced inspection by the resident inspectors of licensee action on previous inspection findings; operational safety verification; surveillance; maintenance; and Licensee Event Reports.
Results: Of the five areas inspected, no violations or safety concerns were identified.

DETAILS

1. Persons Contacted

*W. A. Shamla, Plant Manager
M. H. Clarity, Assistant to the Plant Manager
D. E. Nevinski, General Superintendent, Engineering and Radiation Prot.
H. M. Kendall, Plant Office Manager
D. D. Antony, General Superintendent, Operations
R. L. Scheinost, Superintendent, Quality Engineering
J. R. Pasch, Superintendent, Security and Services
L. H. Waldinger, Superintendent, Radiation Protection
W. J. Hill, Superintendent, Technical Engineering
W. W. Albold, General Superintendent, Maintenance
B. D. Day, Superintendent, Operating Engineering
L. L. Nolan, Superintendent, Nuclear Technical Services

The inspector also contacted other licensee employees including members of the technical and engineering staffs, and reactor and auxiliary operators.

*Denotes the licensee representatives attending the management exit interviews.

2. Licensee Action on Previous Inspection Findings

a. (Closed) Open Item 263/82-05-12: Minimum Shift Staffing and Control Room Modification

This open item was issued to track the addition of a second Senior Reactor Operator (SRO) to each shift to meet the minimum shift manning required by NUREG-0737. The item also tracked the proposed modifications to the Shift Supervisor's office which would enable its inclusion in the boundary of the control room.

Shift manning has been increased to include a second SRO as of December 31, 1983, as documented in the Emergency Procedure (EP-5), Table 5-1, Minimum Shift Staffing and Capability for Additions for Nuclear Power Plant Emergencies. A recent inspection by resident inspectors confirms the addition of the second SRO in accordance with the licensee's commitment to the NRC.

The modification to the shift supervisor's office has been completed including the installation of the control room video system, SPDS monitor, keycard access, and installation of key annunciators. These modifications were completed in December 1985 per Modification Package No. 84M007.

With the completion of both the above commitments, the above stated open item is considered closed.

b. (Closed) Open Item 263/82-05-19: Location of Meteorological Measurements Readout

This open item was issued to identify the current installation of the meteorological measurement readouts in the shift supervisor's office as being unsatisfactory because this office was not part of the control room.

The meteorological measurement readouts have been relocated inside the licensee's control room per Modification Package No. 84M004 completed in March 1985.

With the completion of this modification, the above open item is considered closed.

3. Operational Safety Verification

a. The unit operated in coastdown to refueling during the inspection period. One exception was on September 12, 1987, When a power reduction occurred in order to facilitate work on the No. 12 condensate pump seal return line and installation of a vent line on the pump's discharge. The restart of the condensate pump and return to full power was without incident.

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during the inspection period. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components, and walked down the emergency service water system. Tours of the reactor building and turbine building were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance, plant housekeeping/cleanliness conditions, and radiation protection controls were implemented.

b. Core Spray System Walkdown

The inspector performed an inspection walkdown of the accessible portions of the core spray system in an effort to identify any discrepancies with that system. The following is a list of the deficiencies identified during the walkdown.

(1) Drawings M-122 and M-114-1, P&ID Core Spray System and Service Condensate System, discrepancies.

(a) Pressure sensing piping arrangements are incorrect for Valves PCV-2458 and PCV-2459

(b) Pressure indicators for PI-7324, Service Cond. Sys to Core Spray A, and PI-7325, Service Cond. Sys to Core Spray B were not shown on their corresponding prints.

(c) Valves associated with instrumentation do not appear on any of the P&IDs.

The licensee was informed of these discrepancies and stated that a review of P&IDs M-114-1 and M-122 would be completed and that any identified problems would be corrected. The licensee also stated that valves associated with instrumentation were not required to be shown on the P&IDs and that these valves were controlled by the I&C Department.

(2) Housekeeping and Labeling Discrepancies

(a) Flow Sightglasses FG-2630 and FG-2631 were dirty and the flow of water through them could not be observed.

(b) Valve CST-100-2 packing gland follower, stud, and nut are corroded.

(c) Valve CS-3-1 has rust on the packing gland nut and follower due to standing water.

(d) The following valves and components have no identification labels:

RV-1745	RO-1743	RO-2854	CST-104-1	CST-103-1
RV-1746	RO-1744	RO-2855	CST-104-2	CST-103-2
RV-2470	RO-1747	RO-1748	CST-101-1	RV-2471
CS-24-1				

All Core Spray instrumentation root, isolation, vent, and drain valves.

The above concerns were discussed with the licensee and appropriate corrective action has been taken. No further review of these items is necessary at the present time.

(3) Material Problems

(a) High point vent valve on instrument line to PT-14-38B had both packing gland nuts and one packing gland stud missing.

(b) Instrument cover for PT-14-38B has a missing bolt.

(c) Instrumentation lines from FE-14-39A are not attached to the mounting brackets provided.

The high point vent on PT-14-38B was immediately repaired and the other above items have had corrective action initiated by the licensee. No further review of these items is necessary at the present time.

(4) Configuration Control Problem

Valves on the instrumentation in the Core Spray system are not listed in the Pre-Startup Valve Check List. Operations personnel stated that they do not verify the correct position of any instrumentation valves, that it is the responsibility of I&C personnel performing instrument calibration.

The inspector interviewed I&C personnel and walked-through a calibration procedure with a technician, observing the process of valve alignment used. The technician performing the walk-through utilized only the isolation valve at the base of the instrument in performing the calibration. When asked if the other instrument line valves were checked the technician stated that only the valves used are checked and seal wired.

The inspector noted that several instrument valves near the instruments had been seal wired; however, several of the root, drain, and vent valves were not. The lack of seal wire indicates that the instrument valves in question are not verified in the correct position by I&C personnel; therefore, they appeared to be under no positive configuration control.

Plant Pre-Startup Checklist (Form 2161) for process instrumentation is an I&C procedure that requires all instrument valves to be positioned and have lock wires attached. The checklist, however, only lists the instrumentation and does not list all the valves associated with each instrument. This procedure has been under review by I&C personnel and is planned to be revised in the near future to provide clearer direction to I&C technicians performing the checklist.

During a reinspection, the inspector determined that the following instrumentation valves were not locked wired.

PT-14-38A Root Valve
PS-14-44B Root Valve

These valves were brought to the immediate attention of I&C management and have since been verified in the correct position and locked wired. The inspector will review the revised plant prestart checklist for adequacy. (Open Item 50-263/87012-01(DRP))

(5) Discrepancies Involving Locked Valves

(a) Valve CST-101-2, Bypass of Service Cond. to Core Spray B is locked closed and shown as such on Dwg. M-114-1. Valve CST-101-1 serves the same function for the "A" header of core spray, however, it is not locked and is not shown as being locked on Dwg. M122.

This inconsistency was discussed with the licensee and the licensee stated that CST-101-2 is not required to be locked per the Pre-Startup Valve Checklist. The valve has had the lock removed and is now in agreement with the valve checklist. The control of locked valves was an INPO-identified problem that the licensee is in the progress of addressing. A review of the licensee's progress in this area is necessary to fully evaluate the current situation. The locking of this valve, however, indicates a possible failure to follow procedure and will also be tracked under Unresolved Item 50-263/87012-02(DRP).

(b) Valves CST-150 and CST-151 are not shown in their current locked open position on Dwgs. M-122 and M-114-1. Valve positions also are not checked as part of the Core Spray Pre-Startup Checklist.

This was discussed with the licensee and it was determined that these valves were recently locked due to a finding of the Safety System Functional Inspection (SSFI) and are due to be removed from the lines. This item will be tracked as part of the SSFI effort.

4. Surveillance

a. The inspector observed surveillance testing and verified that testing was performed in accordance with adequate procedures, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate personnel. The following tests were observed:

- Reactor Surveillance Lo-Lo Level ECCS Initiation and Hi Level RCIC/HPCI Turbine Trip Unit Test
- No. 11 Standby Diesel Generator Emergency Service Water Monthly Tests
- Standby Liquid Control Monthly Operability Test (Discussed in detail below)

b. The inspector observed Standby Liquid Control System Surveillance Test 0255-02-III, Pump Monthly Operability, and Test 0255-02-IA, Part I, Valve Quarterly Exercise, Revision 9. The inspector found that the tests were conducted in a professional manner by knowledgeable personnel; however, the following discrepancies in the procedure were noted.

(1) Steps 7 and 15 of the procedure instruct the operator to throttle discharge of the SBLC pumps to achieve a 1275 psig discharge pressure. The gage used (PI-11-53) is a 0-2000 psig gage that is marked in 20 psi increments, therefore, it is not capable of clearly indicating 1275 psig. The setting of discharge pressure precisely at 1275 psig is further hampered by the gage oscillation due to the natural discharge pulsations of the SBLC system's positive displacement, triplex plunger pumps.

(2) Steps 10 and 18 of the procedure require the recording of the SBLC pump's discharge pressure and provide guidance that the reading should be greater than or equal to 1275 psig not 1275 psig exactly.

(3) Technical Specification 4.4.A.1 requires only that SBLC pumps are proven capable of producing 24 gpm at 1275 psig, but both SBLC pumps tested at capacities in excess of this requirement.

The above observations were discussed with the licensee and it was determined that the directions in Steps 7 and 15 should be changed to reflect the guidance given in Steps 10 and 18. This change will provide the operator performing the test with a conservative band for SBLC pump discharge pressure. The change will ensure that the testing of the SBLC pumps will meet or exceed the required discharge pressure.

(4) Step 31 required the closing of Valve XP-17, suction to the SBLC pumps from SBLC test tank. Step 32 then instructed the operator to drain the test tank to the 55 gal. drums on the 935' level of the Reactor Building. This draining process took approximately 30 minutes and upon completion the operator is to realign the SBLC system into its normal configuration. These steps appeared to be in less conservative order in that the SBLC system could be aligned prior to the draining of the test tank. By reordering the steps, the SBLC system could be made operable prior to and during the test tank draining operation.

This observation was also discussed with the licensee and the licensee agreed that, if possible, the reordering of the steps would be a more conservative method of performing the test.

The review and possible revision of the SBLC pump test procedure incorporating the changes described in Parts 1 through 4 above will be tracked as an Open Item (50-263/87012-03(DRP)).

(5) After completion of pump capacity and vibration checks, the SBLC test tank is required to be drained to 55 gal drums on the 935' level of the reactor building. This process required the opening of XP-15, SBLC test tank drain. When the valve was opened water began flowing out of the drip tray around both SBLC pumps and onto the floor. The operator stated that this occurred because XP-25, SBLC drain valve to the 55 gal drums

was closed and that the water was backing up the drain lines to the pump drip trays. XP-25 is a locked open valve and its operation is not addressed by the test procedure. If XP-25 had been in its correct position, there would have been no overflowing of the SBLC pump drip trays. The operation of XP-25 was not in accordance with the procedure, however, it is not a safety significant valve.

This item was discussed with the facility management and requires further review by the inspectors. This item will be tracked as an Unresolved Item pending the results of the further review. (50-263/87012-04(DRP))

5. Maintenance

Station maintenance activities on safety-related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, and industry codes or standards, and in conformance with technical specifications.

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

- Recirculation Loop Speed Control Calibration
- Calibration of Emergency Filtration Train Instruments
- Continuity Checks of Alternate Shutdown System Panel Motor-Operated Valve Fusing

6. Licensee Event Reports

The inspectors reviewed the following Licensee Event Reports by means of direct observation, discussions with licensee personnel, and review of records. The review addressed compliance with reporting requirements and, as applicable, that immediate corrective action and appropriate action to prevent recurrence had been accomplished.

a. (Closed) LER 263/84025 - Start of No. 11 Emergency Diesel Generator (EDG) while isolating No. 16 Safeguards Bus. While isolating the No. 16 bus for relay maintenance, the No. 11 EDG started when an operator opened a bus potential transformer (BPT) door on the rear of Cubicle 152-610 rather then Cubicle 152-601 as required by the procedure. The EDG started properly. It did not load onto the bus as the bus was being supplied by an offsite power source and was shutdown. Operators were reminded to take care in reading numbers and to be aware of potential consequences of inverting the two digits. A caution tag was placed on the BPT door warning of potential EDG start upon opening of subject bus BPT door. The inspector reviewed the LER with the licensee and was satisfied with the corrective action.

b. (Closed) LER 263/84028 - Trip of Reactor Protection System (RPS) Motor Generator (MG) Set. The overload/overtemperature relay for the "B" RPS-MG Set open-circuited. This resulted in shutdown of the motor, loss of power to Bus "B" loads, and initiation of secondary containment and Group 2 isolations. Power was restored to Bus "B" from the alternate source until the relay was replaced. After replacement of the relay, the MG Set was returned to service. Subsequently, the comparable relay in "A" RPS-MG Set was replaced.

The relays were replaced with identical replacements except that the coil voltage was increased from 115 VAC to 120 VAC. This was done to improve the relays' reliability. The inspector discussed the LER with the system engineer and concluded that the corrective actions were implemented.

c. (Closed) LER 263/84029 - Emergency Diesel Generator Starts Associated with 1 AR Transformer. While restoring 1 Auxiliary Reserve (AR) transformer to service, the transformer supply breaker disconnects were closed out of sequence; also, the potential transformer fuses for the transformer were not reinstalled before removing the emergency diesel generator auto-start isolation. Both instances caused a start of No. 12 emergency diesel generator.

Offsite power to the plant was being supplied by the 1 AR transformer during this time and these occurrences had no effect on plant operations. The diesel was subsequently shut down in both instances.

Corrective actions included a modification to eliminate automatic anticipatory starts of the diesels, training was provided to operations personnel regarding bus potential transformers, and an administrative instruction was issued which required plant engineering review of all outage and restoration procedures. The inspector reviewed documentation of the training provided and verified that the administrative procedure was revised.

d. (Closed) LER 263/84032 - Scram Due to Intermediate Range Monitor (IRM) Spike During Surveillance. During cold shutdown on November 27, 1984, a reactor scram occurred. The cause of the scram was a noise spike on IRM Channel 13 which tripped Channel A of the Reactor Protection System while Channel B was tripped to perform a main steamline radiation monitor surveillance test. Before all control rods were fully inserted at the time of the scram, no rod motion occurred.

The noise spike is attributed to construction activity in the drywell in the area of the IRM detector cables. The licensee took no specific corrective action in response to this event. A later similar event (LER 263/86019) resulted in closer control of activities under the reactor vessel to preclude jarring of IRM cabling.

7. Exit Interview

The inspectors met with the licensee representatives denoted in Section 1 at the conclusion of the inspection on October 7, 1987. The inspectors discussed the purpose of the inspection and the findings.

The inspector also discussed the likely informational 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/processes as proprietary.