U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-263/89019(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: July 18 through September 2, 1989

Inspectors: P. L. Hartmann

J. E. Hard

P. B. Moore

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

9/15/89
Date

Inspection Summary

Inspection on July 18 through September 2, 1989 (Report No. 50-263/89019(DRP)
Areas Inspected: A routine, unannounced inspection by the resident inspectors of previous inspection items; operational safety verification; maintenance; surveillance; and onsite followup of operational events. Planning and work performance related to the outage was also inspected.
Results: During this inspection period, the licensee coasted down to 46 percent power before shutting the plant down for a refueling outage. The outage is expected to last for 79 days and this report period covers activities through day 15 of the outage. During this inspection period, fourteen open items were closed out. The licensee experienced an ESF actuation, two small flammable incidents, and one spill of contaminated liquid. An open item was created to track the improvement needed to the General Employee Training, to include more emphatic instructions regarding liquid spills in the plant.

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DETAILS

1. Persons Contacted

*D. D. Antony, Plant Manager
M. H. Clarity, Assistant to the Plant Manager
B. D. Day, General Superintendent, Engineering and Rad. Prot.
D. E. Nevinski, General Superintendent, Operations
R. L. Scheinost, General Superintendent, Quality, Security and Admin.
L. L. Nolan, Superintendent, Nuclear Technical Services
S. A. Engelke, Superintendent, Nuclear Technical Services
S. J. Hammer, Superintendent, Operations Engineering
G. T. Bart, Quality Supervisor

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 resident inspectors' exit meeting.

2. Licensee Action on Previous Inspection Findings (92701 92702)

a. (Closed) Generic Letter 85-13 - Transmittal of NUREG-1154 Regarding the Davis-Besse Loss of Main and Auxiliary Feedwater Event. The licensee received NUREG-1154 and in response to the NUREG and Information Notice 85-50, Complete Loss of Main and Auxiliary Feedwater at a PWR Designed by Babcock and Wilcox, formed a task force to review the event. The result was a listing of NUREG findings and conclusions. In addition, a response was generated for specific applicability to Monticello. These responses commented on the finding and identified why present plant actions were acceptable to prevent a similar situation or provide a recommendation to incorporate lessons learned into the plant's operating or maintenance practices. Recommendations were added to the plant experience tracking system. The inspector reviewed the February 12, 1986, Task Force Recommendations letter and concluded that the licensee has reviewed the NUREG in detail and considered its applicability to the plant.

b. (Closed) Unresolved Item (263/87015-05) - Improvements in Instrument Valve Identification and Lineups. The licensee provided a written commitment dated March 28, 1988, in response to inspector concerns and findings. The details of the issue are contained within Inspection Reports No. 263/87012, 263/87015 and 263/88006. This written commitment was revised by letter dated November 30, 1988. Following the review, the inspector concluded that the corrective actions discussed within the written commitments were completed. In particular, the inspector noted that the startup checklist and labeling established and implemented now provides for control of process type valves which were previously uncontrolled.

c. (Closed) Open Item (263/88006-01) - Recurrence of Material Condition Problems of Small Valves. The inspector discussed the inspector concerns for this open item with the General Superintendent of Maintenance. The licensee has, in addition to the corrective actions committed to this concern (see Item a. above, this paragraph), instituted training of operations and maintenance personnel to enhance identification of material deficiencies. The responsibility for material condition of valves which were not previously contained on any valve checklist are now assigned to the I&C Department. The valve checklist titled Plant Prestart Checklist Process Instrumentation, Form 2161, identifies and documents positioning of these valves.

d. (Closed) Violation (263/88006-02) - Fire Door Found Open. On March 14, 1988, the resident inspector found Fire Door No. 105 open with the automatic door closing device unattached from the door. In response, the licensee immediately closed the door and repaired the closing mechanism. The licensee submitted LER 263/88002 in which the corrective actions are identical to those provided by the violation response. The corrective actions appear adequate and are discussed in detail in Paragraph 6.e of this report.

e. (Open) Unresolved Item (263/88017-04) - Design Concerns - Low Pressure Coolant Injection (LPCI) Power Supply. On October 5, 1988, the licensee, following NRC inquiry, informed Region III that a failure of a 125 VDC bus, leaving only one core spray pump capable of injecting into the reactor vessel, was not accounted for in the current Monticello Safety Analysis Report and was beyond the original design basis for the plant. The licensee also informed Region III of its intent to correct this design deficiency.

After further review of single failure analysis, the licensee determined for fault analysis purposes that the break could be assumed in the core spray piping. With a loss of a 125 VDC power supply, the result could alter the previous predicted worst case condition. This information was provided by letter dated October 4, 1988.

In response, Region III issued a Confirmatory Action Letter (CAL-RIII-88028) which confirms the following licensee corrective actions.

(1) Provide adequate procedures and training to the operators to make them aware of this situation and actions necessary to manually restore power to the deenergized LPCI valves in the event of a loss of 125 VDC power.

(2) During the next scheduled refueling outage (August 1989) make modifications to assure that a failure of one division of 125 VDC power will not disable all of the LPCI functions.

(3) Submit by October 7, 1988, to the Office of Nuclear Reactor Regulation and to the NRC Region III office, a report describing the current swing bus configuration, including a justification for continued operation of the Monticello facility.

Regarding these actions, the inspector verified that the necessary training and procedures were adequate and implemented. Thus, Item (1) above is closed.

Item (2), a procedure for modification to the LPCI 125 VDC power supply, has been recently written. The modification is scheduled for completion during the current ongoing refueling outage. This modification will be reviewed and inspected by DRS personnel prior to plant startup.

Item (3), required submittals to the NRC. The licensee submitted letters on October 4 and 7 1988, in response to these submittals. NRR by letter dated November 28, 1988, assigned TAC No. 71092, and requested additional information which was submitted in part by the licensee on December 19, 1988. There have been additional correspondence and meetings. NRR will review the modification and will close this TAC item.

f. (Closed) Unresolved Item (263/88021-01) - Containment Temperature Detector Concerns. The inspector verified that computer locations of several temperature detectors had been changed to the correct location. The inspector reviewed the Volume "F" memo regarding the change of multichart recorder TR-23-115 alarm point and found the basis of the change and its review adequate. This change is temporary and the licensee plans to relocate several temperature detectors during this current outage to provide less localized heating indication (i.e., away from steam piping). This action should result in the ability to return the control board annunciator alarm point to 150 degrees F.

The inspector also questioned several operators regarding what actions would be taken in response to rising or alarming drywell temperature indications. The operators all were acutely aware of the Emergency Operating Procedure entry condition for the continuation of a 150 degrees F drywell temperature. Operators would utilize the SPDS drywell temperature validation function to determine average drywell temperature. With SPDS unavailable, operators would utilize various computer points for drywell temperature indications. In the absence of both these comprehensive temperature indications, operators would rely on the temperature indication of the multichart recorder. The licensee appears to be actively pursuing returning this alarm function to 150 degrees F. The inspector concerns of this unresolved item have been adequately addressed.

g. (Open) Open Item 50-263/89-019-01 - IE Information Notice 89-50, Inadequate Emergency Diesel Generator Fuel Supply. The licensee reviewed their Emergency Diesel Generator fuel supply and calculated

the amount of fuel needed to meet the USAR requirements using the guidance in Regulatory Guide 1.137. The licensee concluded the following:

- The minimum oil storage tank level given in the Technical Specifications, based on having fuel to run one diesel generator at full load (2500 KW) for seven days, is not adequate based on the licensees calculations.

- Each diesel generator does have a local fuel capacity to operate at full load (2500 KW) for at least eight hours as specified in the USAR.

The licensee has implemented corrective actions in the form of temporary changes to all affected procedures and issued a Volume F memo to operations regarding the proper tank levels.

The licensee has not submitted the Technical Specification or USAR changes as of this report period. This item will be closed upon the issuance of the revised procedures incorporating the change and the approval of the Technical Specification and USAR changes.

3. Operational Safety Verification (71707)

a. Routine Inspection

The inspector observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the inspection period. The inspector verified the operability of selected emergency systems, reviewed tagout records, and verified proper return to service of affected components. Tours of the radwaste building, reactor building, intake structure, and turbine building were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under Technical Specifications, the Code of Federal Regulations, and the licensees administrative procedures.

The resident inspector observed training activities conducted in-plant for Non-Licensed Operators on breaker installation and removal.

b. Electronic Pressure Regulator (EPR) Malfunction

On August 4, the licensee experienced a small pressure transient in the RCS when the control valves to the main turbine began cycling due to faulty signals from the plants EPR system. Unlike other plants that use an EHC system for the control of system pressure, Monticello utilizes a combination of the EPR and a Mechanical

Pressure Regulation (MPR) System. During normal plant operations, the EPR functions to maintain system pressure, while the MPR is kept in standby at a slightly higher pressure setpoint.

When the EPR started cycling the control valves, the operators increased its setpoint, raising it out of the pressure range which allowed the MPR to function at the setpoint that was previously being maintained by the EPR. This action terminated the control valve fluctuations. The cycling of the control valves resulted in a minor pressure transient that led to a small flux transient due to void collapse in the reactor core. Some of the APRMs registered this increase. The plant Nuclear Engineer reviewed the data associated with the event and determined that the transient was small and of very short duration. The plant at the time was operating at approximately 50% power and the core was well below its thermal limits and remained so during the transient. In addition, the limitations of the Pre-conditioning Interim Operating Management Recommendation, which governs the rate of power increase to irradiated fuel, were not violated. The inspector reviewed the data and analyses with the Nuclear Engineer.

c. Fire Brigade Actuation

On August 26, at 1455, the licensee sounded the fire alarm and announced over the plant page system for the Fire Brigade to proceed to the condenser room in the turbine building. The resident inspector was on site in the turbine building at that time and was able to observe the actions taken by the plant personnel.

The licensee is currently replacing four of their feedwater heaters that are contained in the condenser. There is a great deal of cutting and grinding in process in the area. The cutting is being accomplished with acetylene torches. There are three fire watches stationed in the condenser room to observe the activities and maintain an active fire watch.

The acetylene is stored outside of the condenser room and the supply is via hoses to the cutting torches. The hoses are sleeved in plastic tubing that is secured to the fittings in the condenser. This was done to minimize contamination. One of the fittings leaked and pressurized the plastic tubing with acetylene. Sparks from grinding activities landed in and around the plastic tubing and ignited it causing a loud pop.

The fire brigade responded to the area in approximately five minutes. A non-licensed operator had already dressed out and entered the area. There was no need for the fire brigade. No one was injured and no damage occurred.

d. ESF Actuations

- Reactor Building Vent Isolation

On August 28, at 0943, the licensee experienced an ESF actuation in the form of an isolation of the reactor building vent on an inadvertent Group 2 isolation. A shift chemist needed to obtain historical data from the reactor building vent Wide Range Gas Monitor (WRGM). This is normally obtained through the plant process computer which had been tagged out for replacement through a modification. Under these circumstances, the technician can obtain the information directly from the WRGM. This action caused the monitor to spike to a Hi Hi condition, which generated the Group 2 isolation signal.

Operations personnel stated that this has been known to happen in the past and plans have been made to rectify this condition. The inspector intends to followup on the licensees actions following the review of the LER describing this event.

- Emergency Diesel Generator Autostart

On August 30, at 1110 hours, an autostart of the No. 11 Diesel Generator occurred due to the placement of a jumper bypass. The jumper was intended to bypass the reactor Lo-Lo level trip function in order to replace reactor level differential pressure transmitters and adjust trip units for Division II of RPS and ECCS water level systems. Incorrect contacts were specified on the jumper bypass form. Placement of jumpers on these contacts caused a false reactor Lo-Lo signal and the subsequent start of the #11 Diesel Generator.

e. Spill in Torus Room

On August 25, early in the day, operations personnel proceeded to drain a portion of the RHR system in preparation for leak rate testing. The drain hoses were placed into floor drains in the Torus room. At approximately 2045 hours, two maintenance personnel returning from a break noticed that some of the water from the drain hose was backing up and not flowing down the drain. This condition was reported to two operators who entered the area, but not to anyone else. During the next hour, the maintenance personnel contaminated themselves by coming into contact with the water while they worked on a large snubber. By 2200, the maintenance personnel had reached access control, where the Friskall detected between 100-1000 counts per minute (considered low level) contamination on the individuals.

The two operators who had entered the area at 2100 had been sent to restore the system. After the control room began filling the system, these operators cracked two valves located near the floor of the Torus room until flow was verified. At approximately 2200, one of the operators proceeded to the catwalk above the Torus to vent a third valve. When the water reached this third valve, a faulty connection resulted in water being sprayed. The operator on the catwalk did not think that the water was highly radioactive and decided to try to stop the leak even though they were not wearing protective clothing. This operator also attempted to warn the other

operator standing below, but was too late. After securing the leaks, which spilled approximately 10 gallons of water, the operators notified the Radiation Protection Office to report that there had been a leak and that they were potentially contaminated. When the operators reached the access control point, the Friskall detected between 100-1000 counts per minute (considered low level) contamination on the individuals.

The Radiation Protection Coordinator (RPC) initiated followup surveys in the Torus area, but communications regarding the location of the spill broke down and the area of the highest contamination was not discovered until later Saturday morning. Two more contaminations occurred when a third maintenance person walked through the area and a plant helper who was working on the decontamination contacted unposted contamination. Approximately twenty people were contaminated during the incident.

The resident inspector was contacted immediately upon his arrival at the plant on Saturday and briefed by the Superintendent of Radiation Protection regarding the investigation into the incident at that point. The superintendent expressed concern that these contaminations indicated a breakdown in the areas of their program designed to contain spills and minimize contaminations. The inspector reviewed the licensees investigation, which is contained in Radiation Safety Deficiency Report (RSDR) 89-11. The analysis section of the RSDR detailed the following specific actions that led to the contaminations:

- The first maintenance personnel and/or the operators should have reported the spill to the RPC or Operations when it was first noticed. These personnel should have stayed out of the affected area and warned other personnel. This represents a casual attitude toward contamination control.

- The operators were following a Radiation Work Permit (RWP) that included notifying the RPC prior to venting the system. The RWP number was not mentioned in the conversation, nor did the RPC inquire about the purpose of the notification. The RWP required that a Radiation Protection Specialist (RPS) be in attendance for the venting, but due to the miscommunication, an RPS was not dispatched to watch the operation.

The resident also reviewed the licensees General Employee Training (GET) lesson plans and course outlines regarding spills and contaminations. The only references given to these topics appear in a section of a video for the initial training that briefly states:

"You must also report any suspected radioactive spills or leaks to the control room and the radiation protection staff immediately";

and a transparency (#21) for both the initial and rebadge training, under the heading of "Control of Contamination and Airborne Radioactivity":

- Source Minimization:
  Repairing, collecting and/or reporting leaks/spills.

The inspectors are concerned about the lackadaisical attitude of the personnel involved in the contaminations. The licensee correctly characterized and analyzed the events surrounding the contamination in RSDR 89-11. At present, the licensees procedure for dealing with leaks and spills is contained in their Operations Manual Emergency Procedure A.2-402, Contamination Control. The licensee has already taken action to write a specific procedure for radioactive leaks and spills.

The inspectors intend to perform followup on the following items:

- Closure of RSDR 89-11 open items by the licensee;
- Issuance of the procedure for radioactive leaks and spills;
- Changes made to the GET, both initial and rebadge, regarding the treatment of radioactive leaks and spills.

The satisfactory review of these items will be followed as Open Item 50-263/89-019-02.

f. Emergency Filtration Train (EFT) Inoperability

Operations personnel contacted the residents on the morning of August 30 to discuss the operability of their EFT system. For one EFT to be operable, it must have its emergency diesel generator (EDG), emergency service water (ESW), and associated controls and instrumentation operable. For the "A" EFT train, this includes the No. 11 EDG, and the No. 13 ESW pump. For the "B" train: the No. 12 EDG and the No. 14 ESW pump. Due to evolutions on equipment related to the outage work, the following events occurred: August 28 at 0850, No. 12 EDG was taken out of service; August 29 at 0317, lead lifted for No. 13 ESW pump. This action rendered both EFT trains inoperable.

Technical Specifications for the EFT is as follows:

3.17.A.2: With one control room ventilation train inoperable, restore the inoperable train to operable status within seven days or be in hot shutdown within the next 12 hours following the seven days . . . .

3.17.A.3: With both control room ventilation trains inoperable, restore at least one train to operable status within 24 hours or be in hot shutdown within the next 12 hours following the 24 hours and in cold shutdown and suspend core alterations, fuel handling and activities having the potential for draining the reactor vessel within the following 24 hours.

On August 30, between 0016 and 0625, operations personnel moved fuel around the fuel pool for fuel sipping. On the same date, between 0100 and 0820, the licensee was draining the defueled reactor vessel.

A shift manager noted that both EFTs were inoperable on August 30 and suspended the drain down activities to determine whether or not they had violated their Technical Specifications.

Discussions between the resident inspectors and operations personnel determined that the EFT system had been inoperable from August 28 at 0317. Fuel movement was suspended on August 30 at 0820. This was a period of 53 hours and 3 minutes. 3.17.A.3 states that fuel handling is not allowed after 60 hours have elapsed from the start of the inoperability period. Operations personnel were fortunate to have noticed the inoperable EFT system prior to the expiration of the limiting condition for operation. With this in mind, operations promptly revised and submitted to the Operations Committee a revision of procedure 9007B, Shift Supervisors Refueling Checklist, with an added step 16 that states: Control Room Habitability as required by Tech Spec 3.17 for fuel movement is established.

g. Security

On August 30, 1989, a corporate QA representative briefed the resident inspector regarding the results of their audit of outage-related activities. Some potential findings made by the auditors were:

1. On August 22, entry into one of the 4 kV vital switchgear areas was not properly controlled by the plant security forces while the permanent barrier had been removed for modifications.

2. On August 22, a hatchway opening to the torus may not have been adequately monitored to prevent unauthorized entry.

3. Following notification to the security department of 1. and 2. above, entries of these events were not made in the security log within 24 hours.

4. Access to the refueling floor may not have been adequately controlled during periods when the drywell and reactor heads were removed.

These subjects will be inspected by a representative of the Region III Safeguards Section during a future inspection.

4. Monthly Maintenance Observation (62703)

Station maintenance activities of 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.

The following items were considered during this review: the limiting conditions for operation were met 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; parts and materials used were properly certified; radiological controls were implemented; and, fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety-related equipment maintenance which may affect system performance.

The following maintenance activities were observed/reviewed:

- WRA 89-0187, "Perform PM No. 4926, Sections I, II, and III on MO-2071, HPCI test return to CST", PM No. 4926 - Preventive Maintenance for Signature Tested Limitorque Motor Operated Valves, Revision 2;

- WRA 89-01740, PM No. 4927 - MOV Signature Analysis Testing Procedure, Revision 1, performed on valve MO-4044A on the Combustible Gas Control System;

- PM No. 4121, RCIC pump, Revision 5;

- PM No. 7180, Diesel Generator System Instrument Maintenance procedure, Revision 6.

No violations or deviations were identified in the review of this program area.

5. Monthly Surveillance Observation (61726)

The inspectors 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 technical specifications and 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 management personnel.

The inspector reviewed/observed the following test activities:

- APRM Recirculation Flow Instrument Calibration, calibration No. 0026, Rev. 10, (Flow Bias Scram Input);

- Scram Discharge Volume High Level Scram Monthly Test and Calibration No. 0006, No. 0025, Rev. 10;

- Tests 0137-9, 0255-06-ID-2/4, Primary Containment HPCI System Exhaust Isolation Valves Local Leak Rate Test, Revision 6.

No violations or deviations were identified in the review of this program area.

6. Licensee Event Reports

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

a. (Closed) LER 263/84034 - Trip of Shutdown Cooling During Primary Containment Integrated Leak Rate Test (ILRT). During pressurization of primary containment for the ILRT, the inboard and outboard isolation valves for the RHR shutdown cooling supply line closed due to a trip of the RHR suction pressure interlock. This action isolated the suction for the inservice RHR pump and caused an automatic trip of the pump.

The interlock was bypassed to allow shutdown cooling to be put back in service. The licensee stated that the procedure for performing the ILRT would be revised to prevent recurrence of this event. The licensee had considered upgrading the RHR low pressure system and raising the pressure interlock setpoint. Upon further consideration by the licensee, the decision was made to forego the low pressure upgrade and simply revise the ILRT procedure and provide a step to bypass the RHR suction pressure interlock. The procedure was revised during the inspection period to incorporate this change, among others, in preparation for the ILRT scheduled for the fall 1989 outage.

b. (Closed) LER 263/86012 - Failure to Perform Surveillance Test on Traversing Incore Probe (TIP) Penetrations. During a review of containment fabrication drawings, the licensee discovered a group of double gasketed penetrations which were not included in the surveillance procedure for the testing of double gasketed seals. The penetrations were all individually tested and found to have zero leakage.

Corrective action included temporary changes to the double gasketed penetration surveillance schedule. A permanent change was made to Test No. 0137, Primary Containment Double Gasketed Seals Local Leak Rate Test, during this inspection period. Enforcement considerations were documented within Inspection Report No. 263/86003(DRP).

c. (Open) LER 263/86017 - ECCS Initiation During Reactor Vessel Water Level Instrumentation Installation. On June 15, 1986, with the plant in cold shutdown, an Emergency Core Cooling System (ECCS) initiation occurred during modification of the reactor water level instrumentation. The cause was administrative and procedural control inadequacies which allowed; (1) interchange of Reactor Protection System and ECCS trip units during initial installation,

and (2) connection of the trip units to the initiation relay logic prior to integrated functional testing of the sensor/trip unit subsystem.

The trip resulted in automatic start of both Emergency Diesel Generators. ECCS pump motor breakers were racked out for maintenance which prevented a start of the pumps. The Emergency Diesel Generators were secured, the trip units were restored to correct locations, and trip unit outputs were verified. Corrective action included revising an administrative procedure which governs preparation of modification procedures to require integrated functional testing of sensor/trip unit subsystems prior to connection to inservice initiation logic circuits. This action was documented as completed within the licensee tracking system. Upon attempt to verify its implementation, the inspector learned that the change was never made.

The second corrective action required a revision to the I&C technician manual to improve the control process related to the installation of new instrumentation. This change is also not accomplished. However, this action item was currently indicated as uncompleted.

When the above procedure changes are completed, this LER will be closed.

d. (Closed) LER 263/87011 - Fire Barrier Penetration Found Inoperable due to Inadequate Procedure. On April 18, 1987, while performing a normal plant inspection, a plant operator noticed a power cable running through an unplugged one-inch conduit in a fire barrier penetration seal, indicating that this penetration seal was inoperable, contrary to the requirements of Technical Specification 3.13.G.1. A fire watch was immediately established at the inoperable penetration seal. The power cable was removed and the penetration was properly sealed within three hours of discovery. The root cause of this event was determined to be an inadequate procedure, since the procedure utilized for sealing fire barrier penetrations does not address the sealing of short sections of conduit that are imbedded in these seals. An intermediate cause was an inadequate inspection procedure, since this open penetration should have been detected when the fire barrier penetration inspection procedure was performed.

The inspector verified that Procedure 8053, Electrical and Piping Penetration Sealing and Visual Inspection, was revised to provide appropriate guidance for short sections of conduit. Procedure 0275 Part 2, Fire Barrier Wall, Damper and Floor Inspection, was also verified as revised to enhance inspection requirements.

e. (Closed) LER 263/88002 - Failure to Provide Fire Watch due to Inadequate Procedures and Training. On March 14, 1988, a fire door located in an Appendix "R" fire barrier was found in the open position due to an inoperable self-closing mechanism. The condition of this mechanism had been noted previously by members of the plant

staff, but the required fire watch had not been established since it was not recognized that an inoperable self-closing mechanism would cause a fire door to be declared inoperable. The fire watch is required per Tech Spec 3.13.G.2. The self-closing mechanism was repaired shortly after the inoperable condition of this door was identified. The root cause of this event was determined to be inadequacies in procedures and training.

The inspector verified that the Daily Fire Door Inspection procedure was revised to provide more specific instructions to aid operators in identifying inoperable fire doors. A letter was also issued by the plant manager to all plant personnel to bring to their attention the importance of fire doors and to remind them of their responsibilities in the use of these doors. Also, the training department has provided further training on what constitutes an operable fire door. The inspector verified that Preventative Maintenance Procedure 4024, Fire Door Maintenance, was written and implemented to ensure that these doors are kept in a better state of repair.

f. (Closed) LER 263/88004 - Fire Barrier Found Inoperable in Turbine Building due to Inadequate Procedures. On June 30, 1988, while working in the turbine building in an area obstructed by equipment, construction personnel noticed a three-inch diameter open penetration in the floor of 931' elevation of the turbine building. This elevation is one of the plant fire area barriers that separates redundant safe shutdown equipment. This condition was considered reportable as a condition prohibited by plant Tech Spec 3.13.G, Penetration Fire Barriers. The plant staff was notified of this unsealed penetration and a temporary seal was installed immediately. A Work Request Authorization was issued and the penetration was permanently sealed. A thorough reinspection of all accessible walls and floors that comprise the plant fire area barriers was performed to determine if there were any other similar openings. No other openings were discovered as a result of this reinspection.

The inspector verified that Procedure 0275-II, Fire Barrier Wall, Damper and Floor Inspection, was revised providing more specific instructions on how thorough of an inspection is required.

g. (Closed) LER 263/88007 - Reactor Scram Caused by Level Transmitter Calibration. During operation at 100 percent power on December 16, 1988, while performing a reactor protection system reactor level transmitter calibration required by Tech Spec 4.1, a momentary hydraulic disturbance in the Division II safeguards level reference column occurred when the I&C technician opened the pressurizing valve for transmitter LT 2-3-58A. This disturbance resulted in a false high level signal from the Division II instruments that share this reference column, tripping the feedwater pumps and the main turbine. The reactor scrammed as a result of the turbine trip. The pressurizing valve was replaced and the reactor was restarted. The cause of the false high level signal was determined to be degradation of the pressurizing valve for transmitter LT 2-3-58A.

The pressurizing valve is a small metering valve that is used to slowly pressurize the transmitter when returning it to service. Later testing of the valve indicated that it opened more rapidly than an identical new valve. In addition, measurement of the valve needle stem revealed differences when compared with a new valve. The manufacturer determined that the degradation was due to wear, most likely accelerated by overtightening the valve when closing. Following the event, further guidance was provided for I&C technicians specifying the proper pressurizing valve closing method.

This guidance included precautions for overtightening and proper opening technique. This guidance was added to formal I&C training by lesson plan. The licensee also required an amendment to the technical specifications surveillance requirement for this particular surveillance. The amendment allows the licensee to perform the surveillance (Reference Revision 116, May 30, 1989) once an operating cycle versus the former requirement of quarterly.

h. (Closed) LER 263/88008 - Engineering Safeguards Feature Activation Caused by Mode Switch Misoperation. During reactor restart activities following a reactor scram on December 16, 1988, a Group 1 primary containment isolation and resultant actuation of the reactor protection system (RPS) occurred when contacts in the RUN position closed with reactor pressure less than 840 psig. This inadvertent positioning occurred when a licensed operator was attempting to move the reactor mode switch into the startup mode. When the reactor mode switch was turned from REFUEL to STARTUP by a licensed control room operator, it was moved past the detent in the STARTUP position and into the RUN position. The operator reported that he did not feel that the handswitch was in STARTUP and continued to turn it, causing it to move into the RUN position. The Group 1 isolation and RPS actuations were immediately reset and the main steam isolation valves were reopened, returning the reactor to normal hot shutdown conditions. Operators are reviewing the event as part of periodic training. The licensee has been involved in an Owners Group effort to develop a crisper acting mode switch. One is under development, which the licensee intends to install during future human factors control room design improvements.

i. (Closed) LER 263/89003 - Inadequate Work Control Causes Undetected Loss of Power to Remote Fire Alarm Panel. Beginning January 15, 1989, for approximately 60 hours, the Fire Detection System Annunciator Panel in the control room was inoperable without a patrolling fire watch in the plant. This condition is contrary to Technical Specifications Section 3.13.A. The annunciator panel was inadvertently deenergized during a modification activity. The root cause of this event was inadequate work controls.

To reduce the probability of a future occurrence, personnel involved in the preparation of the work procedure were counseled on the proper considerations for controlling work activities around sensitive equipment. The LER was presented to plant engineering personnel as

part of the regular quarterly engineering staff training. Also, the status of the alarm panel is now verified as operable hourly by control room operators.

A plant policy controlling work activities in areas containing sensitive equipment is being established. In response to a violation related to this event, the licensee committed to completing this action by April 30, 1989. This LER is closed, and the remaining corrective action will be followed by the violation (reference Inspection Report 263/89003(DRP)).

6. Refueling Outage Activities (60710)

This report period covers the activities associated with the licensees outage through September 2, or Day 15 of the outage. Below is a summary of the observations of activities in the plant during the outage.

The inspector observed several shift turnovers which are now conducted twice a day at 0700 and 1900, during the outage. The operators are working 12-hour shifts and the operations supervision is still working 8-hour shifts. All of the observed shift turnovers were thorough and complete. Conditions were in a constant state of change from shift to shift and the operators conducted the turnovers professionally. The ambient noise level, amount of distractions, and number of personnel in the control room have all increased. This does not appear to have adversely affected the ability of the operators to control the plant. The number of personnel available on shift was always adequate.

The inspectors spent an increased amount of time touring the reactor and turbine buildings observing the activities associated with the major work items of the outage. Particular attention was given to the replacement of the 11A,B and 12A,B feedwater heaters in the condenser, the defueling of the reactor vessel, chemistry sample system upgrade, control room panel upgrade, diagnostic testing of the motor operated valves, leak rate testing of primary containment isolation valves, installation of improved security barriers around the 4 KV switchgear rooms, overhaul of the low pressure sections of the turbine, and decontamination of the recirculation piping.

The inspectors attended most morning outage meetings, and plan to continue this practice throughout the outage much in the same way that the inspectors attempt to attend most of the morning meetings during power operations. The outage meetings are quite large and involve all of the plant and outage supervisors. The meetings are run by the scheduling group using a daily handout that summarizes all of the modifications and activities. The handouts contain the schedules for all of the major activities and is revised every day with the exception of the Sunday morning meeting. The meetings progress quickly and any items of discussion are deferred until after the meeting when the principals can meet to resolve their respective items. Observations of these meetings indicate that the licensee is maintaining good positive control of the activities in the plant.

Major Modifications

- 88M063, Replace RWCU Check Valve RC-6-1 and RC-6-2: Not started;

- 88Z013, Replace 11A, 11B, 12A, 12B Feedwater Heaters: Work has been proceeding on schedule. The licensee intends to remove the feedwater heaters next week and should begin installing the new feedwater heaters shortly. Erosion/corrosion in the extraction steam piping has been discovered and is being addressed.

- 88Z031, Install 4KV Area Security Barriers: Work is 50 percent complete and testing is 20 percent complete.

- 88Z042PTA, Improve Control Room Panels C263A and C264B: Not started.

- 88Z042PTB, Improve Control Room Panels C03 and C04, 15 percent completed. All of the old switches and indicators have been removed and the new panel facing has been installed on C04 and is in the process of being installed on C03.

- 89M011, Post Accident Sampling System Upgrade: Not started.

- 89M012, Install Additional Service Water Isolation and Drain Valves: Not started.

- 89M013, Recirculation Sample Line Valve CV-2790 Modification: Not started.

- 89M017, Replace Process Computer System: 20 percent complete, the computer system has been switched over to an uninterruptible power supply.

- 89M071, Core Reload for Cycle 14 Operation: 40 percent complete, the core has been off-loaded and fuel sipping is nearly complete. Rechannelling has been completed.

- 89Z003, Reactor/Turbine Building Sample System Upgrade: The installation of the new sample stations is at 30 percent and the rerouting of the sample lines is at 15 percent.

- 89Z021, Install 480V Swing Bus DC Power Supply: Not started.

Major Work Items

- Overhaul of both low pressure sections of the Main Turbine: The turbine casings revealed a substantial amount of erosion on the 14 stage section. GE analysis indicated that there was no structural degradation. The licensee intends to use Inconel clad overlay on these eroded areas. Inspection of other areas that have suffered erosion in the past before having Inconel overlaid on them demonstrated good durability.

- Inservice Inspection examination of reactor vessel shell welds: Not yet begun.

- Hydroblast cleaning of the Main Condenser: Not yet begun.

- Torus draindown, inspection, and painting: Not yet begun.

- Decontamination of Reactor Recirculation piping: Complete, very successful. The licensee achieved a piping inner surface Decontamination Factor (DF) of 15, and a general area DF of 10 to 20. This decontamination of the recirculation system piping will contribute to a substantial reduction in personnel exposure.

- Primary Containment Integrated Leak Rate Test: Not yet begun.

- Motor operated valve, check valve, manual valve and air-operated valve maintenance and PM's on the following systems: Main Condenser, Condensate and Feedwater, Combustible Gas Control System, Control Rod Drive System, Core Spray System, Circulating Water System, High Pressure Coolant Injection, Main Steam System, Reactor Building Closed Cooling Water, Reactor Core Isolation Cooling, Reactor Recirculation System, Residual Heat Removal, Reactor Water Cleanup, and Main Turbine: Proceeding as scheduled.

7. Response to Regional Request for Information

The residents responded to specific questions from the Section Chief in response to concerns raised by the Executive Director for Operations regarding the nature and effectiveness of the licensees programs for the testing and maintenance of check valves. Specifically, the requests were:

- In general, how do you view your licensee's check valve maintenance/surveillance program (ISI and IST);

- Have there been occurrences of repeat problems with a specific check valve(s) that the licensees program did not appear to correct;

- Provide a listing of other check valve problems and what actions (maint/surv) the licensee took to correct these problems.

On January 11, 1989, the licensee submitted Revision No. 9 to the ASME Code Section XI Inspection and Testing Program to the NRC. Contained within this submittal is the information detailing the licensees ISI and IST program for check valves. The licensee currently includes 157 check valves in their program. The type of testing conducted is stroke time and leak rate.

In addition, the licensee, in their response to INPO SOER 86-03, "Check Valve Failures or Degradation," stated that 115 valves in 11 different systems had been chosen for review using the guidelines in EPRI Report

NP-5479, "Application Guidelines for Check Valve in Nuclear Power Plants."

The review of check valves using the EPRI guidelines had indicated that Reactor Water Cleanup (RWCU) Valves RC-6-1 and RC-6-2 were oversized. When the plant experienced a scram due to a failed feedwater check valve on June 19, 1989, the licensee took the initiative to perform leak rate tests on these valves prior to startup following the scram. The tests revealed excessive leakage through valve RC-6-1. Subsequently, the licensee elected to isolate RC-6-1 prior to startup.

Other actions were taken following the failed feedwater check valve scram to ensure that there were no similar conditions existing in the other feedwater check valves. The discharge check valve on RFP No. 11 was not as degraded as the #12 RFP discharge, but did demonstrate similar wear on its hinge pin and arm. Inspections of check valves downstream from the RFP discharge check valves did not indicate any of the wear or degradation observed in the failed check valve.

Based upon this experience and the evidence of initiative on the part of the licensee towards check valve reliability and trouble shooting, the resident inspectors view the check valve maintenance/surveillance ISI/IST program favorably.

A review conducted of the licensees reportable events files and the Work Request Authorization database did not indicate any repetitive failures among check valves. Only three reportable occurrences that involved check valves had been noted over the life of the plant.

8. Meeting with Corporate Officials

On August 24, 1989, the senior resident inspector met with Mr. J. Howard, Chief Executive Officer, at the licensee's corporate offices in Minneapolis. Subjects discussed included:

- Monticello security program
- Corporate Quality Assurance activities
- NRC and NSP organizational changes
- Transfer of SRIs between Monticello and Prairie Island
- NSP's actions in response to the recent SALP reports
- Status of proposed Maintenance Rule
- Upcoming licensed operator requalification exams

9. Exit Interview (30703)

The inspectors met with the licensee representatives denoted in Paragraph 1 on September 5, 1989. The inspectors discussed the purpose of the inspection and the findings.

The inspectors also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any documents/processes as proprietary.