April 28, 2000

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

SUBJECT: MONTICELLO INSPECTION REPORT 50-263/2000001(DRP)

Dear Mr. Wadley:

On April 1, 2000, the NRC completed an inspection at the Monticello reactor facility. The enclosed report presents the results of that inspection.

During the 8-week period covered by this inspection, activities at the Monticello facility were characterized by the effective conduct of operations. However, several examples of errors associated with control and integration of evolutions indicated a weakness in the plant staff's ability to effectively coordinate work activities during a refueling outage. Examples included two unplanned engineered safety feature actuations. Although your staff also identified these deficiencies and were reviewing the issues to identify areas for improvement these examples highlighted the need for increased attention in this area. During this inspection we also identified a non-cited violation for failure to comply with administrative safety tag requirements.

Based on the results of this inspection, the NRC has determined that one violation of NRC requirements occurred. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy. If you contest this violation or the severity level of the NCV, 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 Monticello.

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 No. 50-263
License No. DPR-22

Enclosure: Inspection Report 50-263/2000001(DRP)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263/2000001(DRP)
Licensee:	Northern States Power Company
Facility:	Monticello Nuclear Generating Station
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	February 4 through April 1, 2000
Inspectors:	S. Burton, Senior Resident Inspector D. Wrona, Resident Inspector P. Pelke, Regional Inspector R. Jickling, Regional Inspector
Approved by:	Roger D. Lanksbury, Chief Reactor Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Monticello Nuclear Generating Station
NRC Inspection Report 50-263/2000001(DRP)

This inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers an 8-week period of resident and regional inspector inspection.

Operations

- The inspectors identified that a residual heat removal valve locking device had been inadvertently removed. The removal of the device occurred, in part, due to the licensee's practice of using common keyed locks for locked valves as well as for miscellaneous equipment control. The inspectors concluded that this was a weakness in the licensee's control of locked valves. (Section O1.2)

- Operations personnel conducted startup activities in a controlled manner. Material condition deficiencies that were identified during startup were promptly resolved and the startup secured for deficiencies that affected the startup. (Section O1.3)

- During testing of the standby liquid control system, operators failed to comply with administrative procedures for the control of safety tagged equipment. The test resulted in the closure of a valve that had been safety tagged open to establish a vent path for other work that was scheduled to be done in parallel with the test. The failure to comply with the administrative procedure was a non-cited violation. The licensee established a team to review safety tagging practices following this safety tagging error, the second that had occurred during the refueling outage. (Section O1.4)

Maintenance

- A long-standing material condition issue with the residual heat removal service water valve resulted in unexpected flow oscillations during system testing. The licensee obtained air-operated valve diagnostic equipment and engineering support from another facility in order to gather data on this problem. The licensee submitted the data collected to the valve vendor and planned on using the vendors analysis and recommendations to resolve this issue. (Section M1.3)

Engineering

- Operations personnel were slow to recognize that the work order for replacing 11 control rod drive mechanisms required revision to ensure technical specification surveillance tests were performed prior to the conduct of maintenance. Engineering personnel were proactive in their response to the emergent issue. (Section E.1)

Plant Support

- The licensee implemented acceptable corrective actions to address previously identified emergency preparedness issues. The inspectors determined that the licensee's emergency preparedness staff adequately identified and implemented corrective actions to resolve NRC-identified concerns. (Sections P8.1 and P8.2)

Report Details

Summary of Plant Status

The inspection period began with the unit shutdown for refueling outage number 19. Reactor startup was commenced on February 27, 2000, and the outage officially ended when the unit connected to the electrical distribution grid on February 29. Full power operation was achieved on March 3. The unit remained at approximately 100 percent power operation for the remainder of the inspection period with two exceptions. On March 5, power was reduced to approximately 75 percent for control rod pattern adjustments. On March 26, power was reduced to approximately 65 percent for reactor recirculation flow and main turbine testing.

I. Operations

O1 Conduct of Operations

O1.1 General Comments

a. Inspection Scope (71707)

The inspectors observed various aspects of plant operations, including use of Technical Specifications (TSs), plant procedures, and the Updated Safety Analysis Report (USAR); communications; management oversight; proper system configuration and configuration control; operations committee; and operator performance during refueling activities, reactor startup, routine plant operations, and plant power changes. The inspectors also performed a walkdown of the drywell prior to closure and portions of the core spray system.

b. Observations and Findings

The conduct of operations was characterized by good procedural compliance, evaluations of risk for work activities, proper three-part communications, and safety-conscious performance. Evolutions such as surveillance tests, reactor startup, and plant power changes were well-controlled, deliberate, and were performed in accordance with procedures. Shift turnover briefings were comprehensive and were typically attended by the plant manager and the general superintendent of operations. Material condition was good. Minor material condition and housekeeping discrepancies were brought to the attention of the licensee and promptly corrected, and condition reports were issued, when appropriate. Containment isolation valves were observed to be properly aligned.

O1.2 Drywell Inspection

a. Inspection Scope (71707)

The inspectors conducted an inspection of the drywell prior to reactor startup. As part of this assessment, the inspectors also reviewed the following documents:

- Procedure [administrative work instruction] 4AWI-04.04.01, Revision 16, “Equipment Isolation”

- Procedure 2154-26, Revision 38, “Drywell Prestart Valve Checklist”

- Procedure 1401-1, Revision 20, “Locked Valve Alignment”

b. Observations and Findings

The material condition of equipment in the drywell was good. Minor discrepancies were communicated to the licensee and promptly corrected. Components were properly aligned, but one valve was found unlocked. The inspectors observed that the locking device on [residual heat removal] RHR 6-2, “12 LPCI (low pressure coolant injection) Injection Isolation to 12 Recirculation Pump Discharge,” was missing. The inspectors verified that the corresponding valve in the alternate system, valve RHR 6-1, “11 LPCI Injection Isolation to 11 Recirculation Pump Discharge,” was locked open. The inspectors questioned shift management regarding the status of RHR 6-2. The licensee determined that the locking device for RHR 6-2 was inadvertently removed by a plant helper during drywell cleanup activities. The licensee also indicated that this condition would have been discovered during performance of activities contained in Procedure 1401-1, “Locked Valve Alignment,” which was scheduled to occur subsequent to the inspectors’ walkdown of the drywell.

The inspectors found that the licensee used the same keyed locks for both equipment control and locked valve control. The inspectors considered the use of locks that utilize the same key for locked valves as well as other equipment to be a weakness in the licensee’s control of locked valves. This issue is in the licensee’s corrective action program as condition report (CR) 20000929.

c. Conclusions

The inspectors identified that a residual heat removal valve locking device had been inadvertently removed. The removal of the device occurred, in part, due to the licensee’s practice of using common keyed locks for locked valves as well as for miscellaneous equipment control. The inspectors concluded that this was a weakness in the licensee’s control of locked valves.

O1.3 Reactor Startup

a. Inspection Scope (71707)

The inspectors observed portions of the reactor startup from the refueling outage. Operations Manual C.1, Revision 24, “Startup Procedure,” was also reviewed.

b. Observations and Findings

Plant operators conducted the startup in a controlled manner, used peer checking during critical tasks, such as reactivity additions, and followed the instructions contained within the approved procedures used during evolutions. A shift supervisor and a nuclear engineer were in the control room during the startup.

The inspectors observed that a caution for Step V.A.3 of Procedure C.1 stated, “A LPCI LCO [limiting condition for operation] will be required if RHR System is not removed from Shutdown Cooling Mode prior to exceeding 212 degrees Fahrenheit (°F) Reactor coolant temperature." Technical Specifications required LPCI to be operable prior to going above 212 °F when irradiated fuel was in the vessel. The inspectors were concerned that this procedure allowed operators to change operating modes into a mode requiring entry into an LCO. The inspectors interviewed several shift personnel and the General Superintendent of Operations (GSO) to determine if this was an acceptable practice. Interviews indicated that licensee management would not allow a mode change into a condition where an LCO would become active. The inspectors concluded that although the C.1 procedure allowed this condition, adequate training barriers existed to preclude a mode change into an active LCO. Licensee management initiated CR 20001469 to further evaluate a possible revision to Procedure C.1.

Conservative decision-making was observed when several issues developed during the startup. These issues included possible indication of a steam leak on a feedwater heater, and a connection in a breaker for the bus duct blower which appeared to be overheating. The licensee appropriately investigated and resolved the issues prior to proceeding further in the startup.

c. Conclusions

Operations personnel conducted startup activities in a controlled manner. Material condition deficiencies that were identified during startup were promptly resolved and the startup secured for deficiencies that affected the startup.

O1.4 Use of Safety Tags on Hand Switches

a. Inspection Scope (71707)

The inspectors reviewed the isolation associated with [work order] WO 9908594, "HWC-16 [hydrogen water chemistry] Body-to-Bonnet Leak," and discussed the isolation with the licensee. Administrative Work Instruction 4AWI-04.04.01, Revision 15, "Equipment Isolation," was also reviewed.

b. Observations and Findings

The inspectors observed that a safety tag was installed on the valve hand switch of valve CV-2791, "Sample Line Outboard Isolation Valve," to keep it open in support of maintenance. During testing the valve repositioned closed, contrary to the position required by the safety tag. The inspectors discussed this issue with a shift manager. The shift manager stated that he had evaluated the closure of the valve and the safety tag prior to the conduct of the test and determined that it was acceptable to proceed. He had reasoned that the valve would only be shut for a short period of time, and that he had verified that maintenance work had not yet commenced for WO 9908594.

An isolation had been established for maintenance on the valve that included tagging the hand switch of valve CV-2791 to keep it open as a vent/drain path; however, the isolation instructions provided for WO 9908594 did not ensure a valve required to be open for a vent/drain path remained open under all conditions. The inspectors concluded that the tagging was incomplete because tagging the hand switch did not ensure that the valve would remain open if the air was removed from the control valve.

Step 4.2.1.6 of 4AWI-04.04.01 stated that "safety tagged equipment shall not be operated until the safety tag has been properly released and removal is authorized by the shift supervisor." Section 4.8 of 4AWI-04.04.01 also provided instructions for the process to be used for temporary removal of safety tags. Technical Specification 6.5 required, in part, that detailed written procedures be followed for corrective maintenance of plant equipment. Contrary to the above, on February 8, 2000, safety tagged equipment (CV-2791) was operated as part of testing prior to the safety tag being properly released and removal authorized by the shift supervisor. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy (NCV 50-263/2000001-01(DRP)).

This issue was in the licensee's corrective action program as CR 20000635. The inspectors discussed this issue with the GSO. The GSO stated that, due to this issue and the safety tagging issue discussed in Section M1.2 of Inspection Report 50/263-99009(DRP), a team was formed to review the isolation process.

c. Conclusions

During testing of the SBLC system, operators failed to comply with administrative procedures for the control of safety tagged equipment. The test resulted in the closure of a valve that had been safety tagged open to establish a vent path for other work that was scheduled to be done in parallel with the test. The failure to comply with the administrative procedure was an NCV. The licensee established a team to review safety tagging practices following this safety tagging error, the second that had occurred during the refueling outage.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments on Maintenance and Surveillance Test Activities

a. Inspection Scope (61726, 62707)

The inspectors observed portions or all of the following maintenance and surveillance test activities:

- WO 9908669, "Replace CRD [control rod drive] Pump Minimum Flow Orifices"
- WO 0000945, "Install Flood Barrier at Lower 4KV"
- Procedure 1371, Revision 4, "Drywell Prestart Inspection"
- Procedure 0036-02, Revision 19, "ECCS [Emergency Core Cooling System] Automatic Initiation Test, Including Loss of Auxiliary Power"
- Procedure 4851-30PM, Revision 5, "LPCI Swing Bus Relay Calibration"
- Procedure 0255-20-IIC-1, Revision 12, "Reactor Coolant Pressure Boundary Leakage Test"

- Procedure 0075, Revision 9, “Control Rod Drive Coupling Test”

- Procedure 0074, Revision 26, “Control Rod Drive Exercise”

b. Observations and Findings

In general, the inspectors found that the activities specified in maintenance and surveillance test procedures were performed in a professional and thorough manner, and completed in accordance with the applicable procedures. Workers that were interviewed were knowledgeable of their assigned tasks. When applicable, appropriate radiological work permits were followed. The inspectors observed supervisory and engineering department personnel involvement in the activities. Adequate foreign material exclusion controls were observed to be in effect. Personnel generally demonstrated effective three-part communications, self-checking, and peer-checking.

Issues, identified by the licensee or the inspectors, were promptly corrected and entered into the corrective action program. One example was when the licensee obtained a controlled copy of the emergency diesel generator prestart checklist prior to the incorporation of a temporary change for a modification to the DROOP controls for the generator governor. The procedure was reviewed as controlled by the shift supervisor and the checklist begun using the wrong revision of the procedure. However, prior to the checklist being signed off, operators questioned the appropriateness of the procedure used to perform it and the checklist was redone using the correct revision of the procedure. The licensee entered this issue into the corrective action program.

M1.2 Integration of Various Processes Associated with Maintenance

a. Inspection Scope (61726, 62707)

The inspectors performed routine tours and reviewed condition reports to assess how the licensee integrated various work processes. The processes included the use of temporary information tags and work request tags as they pertained to maintenance activities, and how testing and maintenance were coordinated with current plant configurations. The inspectors also reviewed the following documents:

- Event Report 36666 associated with an engineered safety feature (ESF) actuation during standby liquid control refueling testing

- Event Report 36670 associated with an ESF actuation during main steam isolation valve testing

- Licensee Event Report (LER) 50-263/00-003, “Procedural Inadequacy Results in Two Automatic Closures of Recirculation Sample Containment Isolation Valve”

- Procedure 0086, Revision 21, “SBLC [Standby Liquid Control] Refueling Tests”

- Procedure 0255-07-IA-2, Revision 10, “Main Steam Isolation Valve [MSIV] Functional Checks Test”

- WO 9908594, “HWC-16 Body-to-Bonnet Leak”

- 4AWI-04.04.01, Revision 15, “Equipment Isolation”

b. Observations and Findings

The inspectors observed that the licensee had identified occurrences where work and testing were initiated without considering current plant configurations, resulting in unexpected outcomes. Examples included:

- The licensee documented in CR 20000207, “During Performance of WO 9906418, ECCS Div. 2 Analog Trip System Relay Replacement, ADS [automatic depressurization system] Timer Initiated Unexpectedly,” that the ADS timer actuated unexpectedly during maintenance. The inspectors concluded that the licensee did not consider the current plant configuration when performing the activities specified in the work order.

- The licensee appropriately notified the NRC of two unplanned ESF actuations in accordance with 10 CFR 50.72(b)(2)(ii) when reactor water sample isolation valves unexpectedly closed during maintenance activities.

  - During the performance of SBLC testing, in accordance with instructions contained in Procedure 0086, operators initiated the SBLC system from the control room. When the SBLC system was initiated, an interlock appropriately caused the Group 3 primary containment isolation valves to close. The Group 3 actuation caused reactor water sample isolation valve CV-2791 to close resulting in an unplanned ESF actuation. The licensee documented this issue in their corrective action program as CR 20000635.

  - The licensee documented in CR 20000584, “CV-2790 & CV-2791 Not Closing During Performance of 0153-02 due to Jumpers Hung for 0137-23OCD Jumper Bypass #00-24,” that valves CV-2790 and CV-2791 did not close as expected due to jumpers installed to support other maintenance activities.

The inspectors considered these issues as examples, along with other examples identified in Sections M1.1 and E1.1, of not properly coordinating and integrating activities during the refueling outage. Although these examples were of low risk significance, the inspectors determined that, based on the number of observations in this area, additional attention in this area was warranted by the licensee. The licensee also identified that coordination and integration of activities had not met expectations and had generated a condition report to track the identification of areas of improvement.

c. Conclusions

Several examples where the licensee failed to properly control evolutions and integrate activities were identified during the refueling outage. Though two of the examples included unplanned ESF actuations, all of the examples were of low risk significance. Each example was entered in the licensee’s corrective action program. The licensee also wrote a condition report to track the identification of areas for improvement in controlling and integrating activities during outages.

M1.3 Residual Heat Removal Service Water (RHRSW) 11 RHR Heat Exchanger RHRSW Outlet Valve CV-1728 Diagnostic Test

a. Inspection Scope (62707)

On March 14, 2000, the inspectors observed activities specified in portions of Procedure 0255-05-IA-1, Revision 39, “RHR Service Water - Valve Operability Test,” and WO 000643. The purpose of the WO was to monitor valve CV-1728 using air-operated valve diagnostic equipment during operation of the RHRSW system. The position of the valve had oscillated during previous surveillance testing.

b. Observations and Findings

Normal RHRSW system flow was 3500 gallons per minute (gpm) for one-pump operation and 7000 gpm for two-pump operation. Operations Manual B.08.01.03-05, Revision 14, “RHRSW System Operation,” had a general precaution that stated when placing both “A” Division RHRSW pumps in service, or shutting down one of the two operating “A” Division RHRSW pumps, operate the flow controller to minimize the time in flow range of 4500 to 6000 gpm. This precaution would minimize the time spent in the operating range where CV-1728 experienced flow instabilities. The precaution for the “B” Division was the same but specified a flow range of 4500 to 6500 gpm where instabilities occur. The inspectors identified that no basis for the precautions were provided in the procedure.

During the performance of the diagnostic test on March 14, the control room operator observed flow oscillations between 6700 and 7200 gpm with CV-1728 at the 75 percent open position and both RHRSW pumps running. This was outside the band specified in the procedure precaution. The operator indicated that when these oscillations occurred the heat exchanger differential pressure remained about 100 psid, well above the 20 psid design limit, indicating that the system design basis was maintained.

The licensee obtained air-operated valve diagnostic equipment and qualified engineering support from Prairie Island Nuclear Station to assist with troubleshooting of the flow oscillation problem. The diagnostic equipment measured the voltage signal from the transducer, the valve position, the closing air pressure, and the opening air pressure on CV-1728. Test data was subsequently provided to the vendor for their analysis. The licensee planned on using the vendor’s analysis and recommendations to resolve this long-standing material condition issue. The inspectors noted that procedural guidance was insufficient when the valve oscillated at flow rates other than expected. Subsequent to the discussion with the inspectors, the system engineer initiated CR 20001211 to address the procedural discrepancies.

c. Conclusions

A long-standing material condition issue with the residual heat removal service water valve resulted in unexpected flow oscillations during system testing. The licensee obtained air-operated valve diagnostic equipment and engineering support from another facility in order to gather data on this problem. The licensee submitted the data collected to the valve vendor and planned on using the vendors analysis and recommendations to resolve this issue.

M1.4 High Pressure Coolant Injection (HPCI) System Notification Retracted

On November 7, 1999, a ventilation cooling coil to one of the two HPCI room coolers developed a leak that required the licensee to isolate the room cooler. During a review of this condition, the licensee determined that HPCI was inoperable, due to the failure of the support equipment, and made a four-hour, non-emergency report to the NRC Operations Center (Event Report 36410). Subsequent analysis performed by the licensee demonstrated that HPCI would remain operable in the event that all room coolers were out-of-service. Based on the analysis, on December 7, 1999, the licensee retracted the previous notification. The inspectors reviewed the retraction and had no concerns.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) Inspection Followup Item (IFI) 50-263/99007-01(DRP): "Reactor core isolation cooling (RCIC) operability, preconditioning, and maintenance rule."

The inspectors identified an issue associated with RCIC operability when a gasket became unwound and entered portions of the governor valve. Issues included possible preconditioning of RCIC bearings by adding oil to the bearing housing prior to running the system and maintenance rule aspects due to unavailability during troubleshooting and maintenance.

The inspectors reviewed CR 19993134, which documented that the RCIC governor valve design would have prevented the unwound gasket from interfering with opening of the valve and thus RCIC would have continued to perform its intended function. The licensee also determined that adding oil to the bearing would not enhance operating characteristics and, in fact, they discontinued the practice due to the potential of foreign material entering the system while adding oil. The licensee addressed maintenance rule aspects of this issue in CR 19993265. Upon further review, the inspectors had no concerns.

M8.2 (Closed) LER 50-263/00-002: "Personal Error Results in Failure to Comply with Requirements of Section XI Operability Test for Emergency Filtration Treatment Service Water Pump." The licensee identified that a water level gauge used during the performance of surveillance testing for the 14 emergency service water (ESW) pump did not meet the test instrumentation requirements of Section XI of the American Society of Mechanical Engineers (ASME) Code. The licensee appropriately considered the surveillance test invalid and considered the 14 ESW pump inoperable. The licensee also identified that:

- Surveillance Test Procedure 0255-11-III-4 was not completed within the required periodicity, since the most recently performed test was considered invalid.

- From January 18, 2000, when the 14 ESW pump operability was questioned until the surveillance test was satisfactorily completed on February 3, periods of time existed when the 13 ESW pump had been inoperable due to maintenance and testing. Therefore, refueling operations had been conducted when both divisions of control room ventilation were not operable as required by TS 3.17.A.

The licensee subsequently demonstrated through testing that the 14 ESW pump was operable, without adjustment. Because the system was capable of performing its intended function, the risk significance of this issue was low. Additionally, the gauge used during the initial performance of the surveillance test was 3.03 times the measured reference value vice the 3.0 allowed by the ASME Code.

Technical Specification Section 4.0.C required, in part, that discontinued surveillance tests be resumed less than one test interval before establishing plant conditions requiring operability of the associated system or component. Contrary to the above, Surveillance Test Procedure 0255-11-III-4 was not performed within one test interval before establishing plant conditions (refueling operations) requiring operability of the ESW system and is considered a violation of minor significance that is not subject to formal enforcement action. This licensee identified item was entered into the licensee's corrective action program as CR 20000579, "Both Divisions of Control Room Ventilation May Have Been Inoperable During Refueling. Tech Spec 3.17.A Not Met." This LER is closed.

M8.3 (Closed) LER 50-263/00-003: "Procedural Inadequacy Results in Two Automatic Closures of Recirculation Sample Containment Isolation Valve."

This issue is discussed in Section M1.2 of this report. This LER is closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Refueling Technical Specifications Not Fully Evaluated

a. Inspection Scope

The inspectors reviewed WO 9904127 for the replacement of 11 control rod drive mechanisms. The inspectors also reviewed the associated technical specifications for refueling, TS 3.10.A, and for control rod drive accumulators, TS 3.3.D.2.

b. Observations and Findings

On January 25, 2000, during review of refueling activities, the inspectors noted that TS 3.3.D.2 stated that "In the refuel mode, a rod accumulator may be inoperable provided that the one rod out refuel interlock for the associated drive is operable," and that TS 3.10.A stated, in part, that "the reactor mode switch shall be locked in the 'Refuel' position during core alterations and the refueling interlocks shall be operable." The inspectors interviewed shift management and engineering personnel to determine the method used to ensure that the refueling interlocks were operable for both requirements and found that the surveillance performed for TS 3.10.A was considered adequate for both TSs. The surveillance test for TS 3.10.A required a weekly test of the refueling interlock using any rod as the test sample. The inspectors were concerned because maintenance was planned for removal and repair of 11 control rod drive mechanisms from under the vessel in a fully loaded core and TS 3.3.D.2 implied that a surveillance test would be required for each rod where the associated drive was inoperable.

The inspectors reviewed the TS change request and the basis for Section 3.3.D.2 that was submitted in 1989 for TS Amendment 63, which resulted in the wording identified above. The inspectors identified that the basis was to permit the replacement of control rod drives from below the vessel. Specifically, testing of the refueling interlocks associated with each drive to be withdrawn and replaced ensured that equipment malfunctions, such as inoperable position indication instrumentation or relaying, for the rod drive to be changed would not result in inadvertent bypassing of the refueling interlock for other rods.

The licensee was again questioned about the determination associated with refueling interlocks on January 29 and 30. The licensee had not initiated a condition report, nor modified procedures associated with the planned maintenance to perform individual refueling interlock verifications. On January 30, the inspectors' inquiries were made to the shift manager, who determined that this issue warranted further review with respect to the TS requirements and that an interpretation of the requirements was required. The shift manager initiated CR 20000504 to investigate this issue.

Subsequently, the licensee determined that testing of the refueling interlock was required and the licensee completed the testing prior to the associated control rod drive maintenance. The engineering staff was proactive in that they did not limit their review to the issues raised by the inspectors and further clarified TS requirements associated with rod maintenance.

c. Conclusions

Operations personnel were slow to recognize that the work order for replacing 11 control rod drive mechanisms required revision to ensure technical specification surveillance tests were performed prior to the conduct of maintenance. Engineering personnel were proactive in their response to the emergent issue.

E8 Miscellaneous Engineering Issues (92700, 92903)

E8.1 Year 2000 Leap-day Rollover

The inspectors reviewed the licensees' preparations for leap-day computer monitoring. The inspectors found that the computer services personnel were actively involved with associated issues and had established procedures to monitor leap-day rollover. The inspectors were in the control room for the February 28 to February 29 roll-over. Prior to midnight, the shift manager informed operators that the date rollover had the potential to affect computers due to computational errors associated with the leap year. The shift manager directed operators to pause startup activities and monitor computer processes during the roll-over. No computer problems were evident to control room personnel during the transition.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

During routine tours of the plant and observations of plant activities, the inspectors found that access doors to locked high radiation areas were properly secured, areas were properly posted, and personnel demonstrated proper radiological work practices. The inspectors reviewed various survey data and radiation work permit (RWP) use and found that personnel were logged onto the correct RWP for the work being performed. Personnel logged into RWPs were wearing proper protective clothing and kept radiation protection personnel informed of activities as required by the RWP. Additionally, the inspectors found surveys to be timely and accurate.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments (71750)

The inspectors observed the licensee implement proper physical security measures associated with the integrity of protected area barriers, personnel and package access, and personnel searches. Lighting was adequate in all areas. The security guards were knowledgeable of requirements and performed the required inspections. The NRC inspectors noted no deficiencies.

F2 Status of Fire Protection Facilities and Equipment

F2.1 General Comments (71750)

During normal resident inspection activities, routine observations were conducted in the area of fire protection. Fire extinguishers and fire hoses were properly stored and inspected by licensee personnel. No notable degradation of equipment was noted.

P8 Miscellaneous Emergency Preparedness Issues (92904)

P8.1 (Closed) IFI No. 50-263/99011-01: During the June 22, 1999, emergency preparedness (EP) exercise, the NRC identified that the radiation protection support supervisor (RPSS) was uncertain as to the cause of a significant difference between the dose projections from an unmonitored release path and the field team measured dose readings. The licensee indicated that the dose assessment procedure would be evaluated for clarification and whether additional dose assessment training for RPSS-assigned personnel was warranted. Condition report 19992312, dated August 5, 1999, was initiated to provide training for radiological emergency coordinators (REC), RPSS, and for MIDAS (dose assessment program) operators and assistants. The MIDAS training was conducted to ensure users understood the capabilities and limitations of MIDAS when used for an unmonitored release. The completed action included presentation of material in EP seminars regarding operation characteristics of MIDAS. No additional changes were made to the training program because the information was covered in existing training lesson plans. An attachment was added to the EP seminars lesson plan and was provided to RECs and RPSSs in December 1999. The licensee also planned to provide MIDAS operators the attachment information during their continuing EP training in March and April 2000. This item is closed.

P8.2 (Closed) IFI No. 50-263/97016-01: During the November 19, 1997, EP exercise, the NRC identified that the emergency manager was uncertain of the need to declare a General Emergency (GE) and delayed the declaration due to inconsistencies between the Alert and GE Classification Guidelines for loss of fission product barriers. Condition report 19980029, dated January 6, 1998, was initiated to evaluate the guidance in Emergency Classification Guideline 28 on what constitutes failure of fuel cladding and to revise the guidance, as necessary, to ensure that clear requirements for classification were provided. A team was formed to review and recommend changes to Guideline 28. A draft procedure change was being reviewed for comment which included a containment monitor response curve which could be used in determining severe core damage. This item is closed.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on April 4, 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

B. Day, Plant Manager
J. Grubb, General Superintendent, Engineering
M. Hammer, Site Manager
K. Jepson, Superintendent, Chemistry & Environmental Protection
E. Reilly, General Superintendent, Maintenance
J. Rootes, Acting Manager, Quality Services
C. Schibonski, General Superintendent, Safety Assessment
E. Sopkin, General Superintendent, Operations
L. Wilkerson, Superintendent, Security
J. Windschill, General Superintendent, Radiation Services

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

50-263/2000001-01	NCV	Failure to remove safety tag prior to testing as required by procedures

Closed

50-263/2000001-01	NCV	Failure to remove safety tag prior to testing as required by procedures
50-263/99007-01	IFI	Reactor core isolation cooling (RCIC) operability, preconditioning, and maintenance rule
50-263/2000-002	LER	Personal Error Results in Failure to Comply with Requirements of Section XI Operability Test for Emergency Filtration Treatment Service Water Pump
50-263/2000-003	LER	Procedural Inadequacy Results in Two Automatic Closures of Recirculation Sample Containment Isolation Valve
50-263/99011-01	IFI	Evaluation of unmonitored release provisions of the meteorological information and dispersion system dose assessment procedure
50-263/97016-01	IFI	Clarification needed to emergency action level guideline 28

Discussed

None

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
ALARA	As Low As is Reasonably Achievable
AOV	Air-Operated Valve
AWI	Administrative Work Instruction
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EP	Emergency Preparedness
ESF	Engineered Safety Feature
ESW	Emergency Service Water
gpm	gallons per minute
GSO	General Superintendent of Operations
HPCI	High Pressure Coolant Injection
HWC	Hydrogen Water Chemistry
IFI	Inspection Followup Item
IP	Inspection Procedure
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
mrem/hr	millirem per hour
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSP	Northern States Power
PERR	Public Electronic Reading Room
psid	pounds per square inch differential
RCIC	Reactor Core Isolation Cooling
REC	Radiological Emergency Coordinator
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPSS	Radiation Protection Support Supervisor
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SBLC	Standby Liquid Control
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
VIO	Violation
WO	Work Order