U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263/99008(DRP)
Licensee:	Northern States Power Company
Facility:	Monticello Nuclear Generating Station
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	November 5 through December 16, 1999
Inspectors:	S. Burton, Senior Resident Inspector D. Wrona, Resident Inspector
Approved by:	Roger D. Lanksbury, Chief Reactor Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Monticello Nuclear Generating Station
NRC Inspection Report 50-263/99008(DRP)

This inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection.

Operations

- Operators demonstrated an increased level of alarm awareness, application of management expectations, and command and control during routine control room evolutions. (Section O1.1)

- The licensee appropriately declared the high pressure coolant injection (HPCI) system inoperable, entered a 14-day limiting condition for operation, isolated and depressurized the HPCI steam line, and made a 4-hour non-emergency notification when they identified that a HPCI steam line support was loose. (Section O1.3)

Maintenance

- Procedures that required changes as a result of the recent reactor power level increase (rerate), which remained outstanding after the rerate, were properly controlled to prohibit use or were identified as having no impact on plant operations. (Section M1.3)

- The licensee appropriately made a 4-hour non-emergency report to the NRC when they discovered a problem with a reactor core isolation cooling (RCIC) system flow indicator that could have affected RCIC system operability. When troubleshooting revealed that the problem was due to instrument drift, which would not have prevented the RCIC system from performing its function, the licensee retracted the 4-hour notification. (Section M1.4)

Engineering

- The standby liquid control system relief valve remained operable with some valve seat leakage. The failure to document the amount of leakage that was acceptable was considered a weakness. (Section E1.1)

Report Details

Summary of Plant Status

Reactor power was at 100 percent at the beginning of the inspection period and remained there until November 12, 1999. Reactor power coastdown prior to the refueling outage, scheduled for January 6, 2000, began at 12:30 p.m. on November 12, when all rods were fully withdrawn and recirculation flow set at the maximum allowed value. Between November 29 and December 3, power was reduced on several occasions by approximately 10-15 percent to support control rod testing. At the end of the inspection period, coastdown continued with reactor power at approximately 86 percent.

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; and operator performance during routine plant operations and plant power changes.

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 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. Plant equipment material condition was good and minor discrepancies were brought to the attention of the licensee and corrected. Containment isolation valves were observed to be properly aligned. Specific events and noteworthy observations are detailed below.

- Operators demonstrated good annunciator response during routine operations by announcing alarms as expected or unexpected, and reviewing annunciator response procedures.
- The inspectors reviewed Equipment Isolation 99-80360, "Instrument Air System 13 Air Compressor K-1C." Hold and Secure Cards (equipment out-of-service tags) were properly removed and the equipment was in the correct lineup. No concerns were identified.

- Operators used three-part communications. Shift supervision enforced management's expectations concerning the use of three-part communications when operators failed to meet the expectations.

- The licensee modified the format of the 7:00 a.m. preshift briefing to minimize the number of personnel in the control room. Information provided by the maintenance and security personnel was duplicated in the licensee's 8:00 a.m. staff meeting. Maintenance and security personnel were no longer required to attend the 7:00 a.m. control room briefing, thereby, reducing potential distractions and minimizing the duplication of information.

c. Conclusions

Operators demonstrated an increased level of alarm awareness, application of management expectations, and command and control during routine control room evolutions.

O1.2 Inspection of New Fuel

a. Inspection Scope (71707)

The inspectors observed operators and engineers perform activities associated with the receipt, movement, and inspection of new fuel. The following documents were also reviewed:

- Procedure 9015, Revision 19, "Procedure for Inspection of New Fuel"

- General Electric Procedure 246-GP-54, Revision 8, "Customer Site Handling and Inspection of GE New Fuel Bundles, Channels and Channel Fasteners"

- Operations Manual D.1-05, Revision 0, "Accountability"

- Operations Manual D.2-05, Revision 6, "Reactor Core Components Handling Equipment"

- Administrative Work Instruction [AWI] 4AWI-02.03.03, Revision 10, "Work Procedure Preparation"

b. Observations and Findings

The new fuel receipt inspections were carried out in a controlled and organized manner. Senior reactor operators (SROs) were present and provided appropriate oversight, nuclear engineers performing fuel inspections were knowledgeable of assigned tasks, and operators who were responsible for rigging the new fuel bundle boxes were aware that the load was considered a "heavy load" and ensured that the approved load path was followed. The fuel inspectors and the SROs ensured that the new fuel assemblies were free of foreign material. When performing operations near the spent fuel pool, operators ensured that foreign material exclusion requirements were followed. The torque wrench used for securing the channel fasteners was in calibration. Licensee management personnel observed portions of the fuel inspections and identified areas for improvement.

The inspectors noted that the licensee was inspecting fuel with sharp objects, such as badge clips, electronic dosimeters, and belt buckles, on their person. The inspectors inquired if fuel inspectors were required to remove or tape over sharp objects. Subsequently, the licensee removed or taped over sharp objects and contacted the fuel vendor to determine if protecting the fuel from sharp objects was a requirement. The fuel vendor informed the licensee that removing or taping over sharp objects was a good practice.

The inspectors observed that the "work copy" of Procedure 9015, located at the job site, did not match the "official copy." The official copy was marked to include the performance of Sections B and C, fuel channel inspections and installation of the fuel channel on the fuel bundle. Although the Sections B and C activities were performed, the work copy was not similarly annotated. The inspectors were concerned that fuel inspectors had not identified that the working copy of the procedure was improperly annotated. The inspectors brought this to the attention of the SRO in charge of the fuel handling and the issue was promptly resolved. A similar issue is discussed in Section M1.2 of this report. The inspectors discussed both of these issues with licensee management.

Technical Specification 6.5 required an Operations Committee (OC) review of fuel handling procedures. The inspectors observed that Procedure 9015 was reviewed by the OC, but noted that a contractor-provided procedure which was being used for the fuel inspections was not reviewed by the OC. The inspectors questioned the nuclear engineering department personnel associated with the lack of an OC review. The licensee concluded that an OC review of the vendor-supplied fuel inspection procedure was not required because Procedure 9015, the controlling procedure, which referenced the vendor procedure within the performance steps, had been reviewed. The inspectors had no further concerns with the licensee's conclusion.

The inspectors were also concerned about controls on reference documents, particularly with vendor-provided procedures that were used to perform quality inspections. The inspectors reviewed 4AWI-02.03.03, and observed that it imposed two requirements on reference documents. First, 4AWI-02.03.03 required that "documents required to properly perform the procedure" be included in the "reference section," and second, if the reference document was not a controlled document then the revision number must also be included. Contrary to the above, Procedure 9015 only specified to inspect fuel and channels in accordance with the vendor-supplied procedure. It did not list the vendor-supplied procedure in the reference section and it did not include the associated revision number. This example of a failure to follow procedures constitutes a violation of minor significance and is not subject to formal enforcement action.

c. Conclusions

Two minor errors associated with the review and performance of the "Procedure for Inspection of New Fuel" were identified and demonstrated a weakness relative to procedural use and attention to detail. The errors involved a failure to specifically identify a reference document used to perform the work and a discontinuity between the authorizations on the working and official copies of the procedure.

O1.3 High Pressure Coolant Injection (HPCI) Safety Restraint

a. Inspection Scope (71707)

The inspectors assessed the actions initiated when the licensee discovered a HPCI steam line support (SR-708) slightly pulled away from a wall. The following documents were reviewed as part of this assessment:

- NRC Event Report 36494, "Inoperability of HPCI due to a Loose Pipe Support"
- Condition Report (CR) 19993640, "HPCI Support SR-708 Baseplate Loose"
- Work Order (WO) 9908401 "Attempt to Reset Base Plate Fasteners on SR-708," performed on December 8, 1999

b. Observations and Findings

On December 7, 1999, the licensee discovered that the baseplate for SR-708 was pulled away from the wall, and promptly performed an evaluation to determine the effect of the failed support. Shift supervision appropriately declared the HPCI system inoperable based on the results of the evaluation, entered a 14-day limiting condition for operation (LCO) in accordance with TSs, isolated and depressurized the HPCI steam line, and made a 4-hour non-emergency notification in accordance with 10 CFR 50.72(b)(2)(iii)(D). The licensee entered this item into the corrective action program as CR 19993640, and wrote WO 9908401 to govern the repair of the support.

The licensee considered the possibility that SR-708 was damaged due to the main steam line flooding event discussed in Section O1.5 of Inspection Report 50/263-99003(DRP), but concluded that this was not the case, based on a walkdown of the HPCI steam line and its associated supports conducted after the main steam line flooding event. The inspectors reviewed the inspections performed by the licensee after the main steamline flooding event which were documented in CR 99001163, "Reactor Water Level Above Main Steam Lines Following SCRAM 107." The inspections supported the licensee's conclusion. The licensee identified that the failure could have been caused by a large dynamic load, such as water hammer, or from a smaller cyclic load caused by pipe vibrations, which had been observed while the HPCI system was in standby. The licensee concluded that the failure was not caused by a large dynamic load because the baseplate was not deformed, the concrete near the baseplate was not damaged, other pipe supports and insulation showed no sign of damage, and no water hammer event on the HPCI steam line had been reported. The inspectors questioned the general superintendent of engineering concerning what was planned to ensure the support would not continue to pull away from the wall due to the steam line vibration. The general superintendent of engineering stated that they planned to redesign SR-708 and periodically monitor the support to ensure operability until the design modifications were completed. The licensee also planned to evaluate other systems to see if they are susceptible to similar failures. The inspectors identified no concerns with the licensee's conclusions and the planned corrective actions.

The support was brought back into compliance with code requirements and an American Society of Mechanical Engineers (ASME) Section XI preservice inspection was performed following the repairs. The HPCI system was declared operable on December 8, 1999.

c. Conclusions

The licensee appropriately declared the HPCI system inoperable, entered a 14-day LCO, isolated and depressurized the HPCI steam line, and made a 4-hour non-emergency notification when they identified that a HPCI steam line support was loose.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments on Maintenance Surveillance Test Activities

a. Inspection Scope (61726, 62707)

The inspectors observed or reviewed the performance of all or portions of the activities contained in the following maintenance and surveillance test procedures.

- Procedure 7264, Revision 3, "Containment Oxygen Analyzer (Hayes) Instrument Calibration," performed on November 15, 1999
- Surveillance Test Procedure 1136, Revision 20, "RHR [Residual Heat Removal] Heat Exchanger Efficiency Test," performed on November 17, 1999
- Surveillance Test Procedure 1054, Revision 8, "Control Rod Drive Normal Drive Timing Test," performed on November 29, 1999
- WO 9907820, "Replace Astrigal Screws [Door-11, Turbine Building to Diesel Generator Room]," performed on November 30, 1999
- Surveillance Test Procedure 0000-A, Revision 71, "Operations Daily Log - Part A," performed on December 6, 1999
- Surveillance Test Procedure 0000-B, Revision 75, "Operations Daily Log - Part B," performed on December 6, 1999
- Surveillance Test Procedure 0000-D, Revision 64, "Operations Daily Log - Part D," performed on December 6, 1999
- WO 9908401, "Attempt to Reset Base Plate Fasteners on SR-708," performed on December 8, 1999
- Procedure 7170, Revision 1, "Instrument Air System Instrument Maintenance Procedure," performed on December 10, 1999

b. Observations and Findings

In general, the inspectors observed that maintenance and surveillance test activities were performed in a professional and thorough manner and completed in accordance with the instructions contained within referenced procedures. The 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 and adequate foreign material exclusion controls. Personnel generally demonstrated effective three-part communications, self-checking, and peer-checking. Specific observations of maintenance activities are outlined below.

M1.2 Residual Heat Removal Service Water (RHRSW) Pump Motor Cooling Cleaning

The inspectors observed activities specified in portions of Procedure 4058-4PM, Revision 1, "RHRSW Pump 12 and 14 Motor Cooler Chemical Cleaning and Pressure Test," performed on November 12, 1999.

The "reason for performing" section of the "work copy" indicated that all steps were to be completed, but all steps for the 12 pump were marked as not applicable. The inspectors brought this to the attention of shift supervision, who promptly resolved the issue by clarifying the "reason for performing" section to identify that only the 14 RHRSW pump was to be cleaned and tested. The "reason for performing" section of the "official copy" was correctly marked to identify which pump was to be tested. This issue, similar to an issue identified in Section O1.2 of this report, indicated a lack of questioning attitude on the part of the personnel performing the work.

M1.3 Procedure Changes Identified Due to Licensed Reactor Power Level Increase (Rerate)

a. Inspection Scope (61726, 62707)

Based on the finding that a procedure requiring a manual scram was not updated in a timely manner, which was discussed in Section O3.1 of Inspection Report 50/263-99007(DRP), the inspectors reviewed the remaining procedures that were identified by the licensee as requiring changes due to the reactor power level rerate.

b. Observations and Findings

The licensee provided the inspectors the following list of procedures which were identified as requiring changes:

- Surveillance Test Procedure 0007B, Revision 4, "Condenser Low Vacuum Scram Instruments Test and Calibration Procedure (<600 psig [pounds per square inch - gauge])"
- Surveillance Test Procedure 0442, Revision 3, "Special Jet Pump Operability Test"
- Engineering Work Instruction EWI-08.09.01, Revision 3, "System Engineering Group Trending Program"

- Operations Manual B.1.4, "Reactor Recirculation System"

The licensee determined that the above procedures did not have to be changed prior to reactor power level rerate. Surveillance Test Procedure 0007B was removed from service and was procedurally controlled from being performed until updated. Surveillance Test Procedure 0442 and Operations Manual Section B.1.4 required data gathering to confirm or re-define the jet pump differential pressure limits. The system engineer was tracking and evaluating the associated data. Procedure EWI-08.09.01 referenced the old thermal power limits; however, this EWI was used by system engineers "to have available a measure of thermal power for historical purposes," and did not have an impact on plant operations. The inspectors reviewed the procedures and identified no concerns with the licensee's determination that the above procedures did not have to be changed prior to reactor power level rerate. The procedures were properly controlled to ensure that they did not have an impact on plant operations or they were not inappropriately used prior to being updated.

c. Conclusions

Procedures that required changes as a result of licensed reactor power level increase (rerate), which remained outstanding after the increase, were properly controlled to prohibit use or were identified as having no impact on plant procedures.

M1.4 Reactor Core Isolation Cooling (RCIC) Flow Indicator

a. Inspection Scope (62707)

The inspectors reviewed a 4-hour non-emergency report and its associated retraction regarding the RCIC system and discussed this issue with the system engineer.

b. Observations and Findings

On November 17, 1999, the licensee identified that the RCIC flow indicator failed downscale, less than zero. A flow indicator reading of less than zero could indicate a problem that would prevent RCIC from performing its intended function. The licensee declared RCIC inoperable, entered a 14-day LCO per TS 3.5.D.2, initiated troubleshooting, and made a 4-hour non-emergency report, NRC Event Report 36445, in accordance with 10 CFR 50.72(b)(2)(iii). During troubleshooting, the licensee determined that the apparent downscale reading was due to flow transmitter instrument drift. Although the amount of instrument drift caused the instrument to indicate downscale, it was insufficient to prevent the system from performing its intended function. The licensee recalibrated the flow transmitter, performed post-maintenance testing, and declared RCIC operable. On November 24, 1999, the licensee retracted the 4-hour non-emergency report, based on the fact that the instrument drift would not have prevented RCIC from performing its intended function. The inspectors had no concerns with the licensee's conclusions.

c. Conclusions

The licensee appropriately made a 4-hour non-emergency report to the NRC when they discovered a problem with a RCIC flow indicator that could have affected RCIC system operability. When troubleshooting revealed that the problem was due to instrument drift, which would not have prevented the RCIC system from performing its function, the licensee retracted the 4-hour notification.

M8 Miscellaneous Maintenance Issues (92700)

M8.1 (Closed) Licensee Event Report (LER) 50-263/99-008: Loss of Speed Control for Reactor Core Isolation Cooling System Turbine Due to Loose Gasket Material Preventing Closure of Governor Valve.

This issue was discussed in Section M1.3 of Inspection Report 50/263-99007(DRP). No new issues were identified following the inspectors' review of the LER.

III. Engineering

E1.1 Standby Liquid Control System (SBLC) Relief Valve Leakage

a. Inspection Scope (37551)

On November 2, 1999, during the performance of activities specified in Surveillance Test Procedure 0085, "Standby Liquid Control System," the licensee noted that the pump discharge relief valves were weeping (slightly leaking). The inspectors reviewed the licensee's procedure and operability determination for the continued operation of SBLC with weeping relief valves.

b. Observations and Findings

The SBLC system is required to mitigate an ATWS (Anticipated Transient Without a Scram), which per the licensee's risk analysis contributed greater than 11 percent to core damage frequency. During an ATWS condition, reactor pressure would be higher than normal, and the inspectors were concerned that a SBLC relief valve whose setpoint may have drifted low could render the system inoperable.

Step 40 in Procedure 0085 directed activities to determine if any leakage past the pump discharge relief valve existed. The procedure-specified basis for this step stated that "leakage from relief valves is indicative of a relief valve which began to lift or lifted, which was most likely caused by a drifting or low setpoint." The licensee performed an operability evaluation during the first observed occurrence of relief valve weeping on September 3, 1998, and again on April 8, 1999. The operability evaluations for the weeping relief valves attributed the cause of the weeping to pulsations from the positive displacement SBLC pump exceeding the relief valve setpoint. The operability evaluation contained data and calculations to support this conclusion. Additionally, one evaluation noted that a TS amendment request to allow the setpoint to be increased had been submitted and when approved and implemented would prevent this condition. Amendment 106 to TSs was issued on October 12, 1999, which incorporated a higher SBLC relief valve setpoint. The licensee intended to reset the SBLC system relief valves setpoints during the next refueling outage.

The inspectors reviewed past documentation and noted that during subsequent performance of activities specified in Procedure 0085, the relief valves failed the leakage test outlined in Step 40 and condition reports were initiated. The operability determinations associated with these condition reports referenced the prior operability determinations noted above. The inspectors were concerned that testing did not take into account an increase in the amount of leakage from the relief valve. Because the operability evaluations did not assess changes in leakage/weeping rates, nor was the procedure revised to provide a method for quantifying a change in relief valve leakage, the inspectors could not determine how the purpose of Step 40 was met and that the relief valves were being evaluated for adverse setpoint drift.

The inspectors interviewed system engineering and operations personnel to determine how relief valve setpoint drift was being evaluated. The system engineer was quantifying the change in leakage rates based upon observation and memory. The inspectors were concerned that visual quantification of flow may not be able to assess an increase in the leakage rate which may be indicative of valve setpoint drift. The licensee had entered this condition into their corrective action program as CR 99003287 for further review.

c. Conclusions

The standby liquid control system relief valve remained operable with some valve seat leakage. The failure to document the amount of leakage that was acceptable was considered a weakness.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Control of Potentially Contaminated Liquid

a. Inspection Scope (71707)

On November 5, 1999, the inspectors observed water leaking from a ventilation cooler while performing a routine tour of the HPCI tank room. The cooler, which was undergoing repairs and located above a radiologically controlled contamination area, was dripping water into the contamination area and the water was flowing across the contamination area boundary. The inspectors also observed that a mop head had been placed onto and across the boundary to minimize the flow of water from the contamination area into the clean area. The inspectors followed up on the radiological controls of this issue.

b. Observations and Findings

Inspectors interviewed radiation protection technicians and reviewed documentation associated with the above observation. Radiation protection technicians indicated that the mop head had been placed inside the boundary to prevent water from traversing the boundary. The mop head had migrated across the boundary due to the force of the water leaking from the cooler. Further interviews and a review of survey records indicated that radiation protection technicians had surveyed the water that had crossed the contamination boundary prior to initiating cleanup, which occurred before the inspectors' observations, and again after the inspectors identified that water had crossed the contamination boundary. The surveys indicated that the level of contamination was below procedural requirements and 10 CFR Part 20 limits. The use of the mop head as a barrier to minimize the potential spread of contamination, without securing it to ensure that it remained within the contaminated area, was contrary to Procedure 4AWI-08.04.03, Revision 7, "Radioactive Material Control." Procedure 4AWI-08.04.03 required that "items which can cross a radioactive boundary are secured in a manner to prevent movement back and forth across the boundary." The inspectors reviewed the documentation associated with the event and verified that surveys had been performed which properly documented contamination levels for pre- and post-leak cleanup activities. The failure to follow Procedure 4AWI-08.04.03 constituted a violation of minor significance of T.S. 6.5.D, "Plant Operating Procedures - Radiological," and is not subject to formal enforcement action.

The inspectors also identified that a CR for the potential spread of contamination had not been written, contrary to 4AWI-10.01.03, Revision 11, "Condition Report Process." The failure to follow Procedure 4AWI-10.01.03 constituted a violation of minor significance of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and is not subject to formal enforcement action. The licensee subsequently entered this condition into their corrective action program as CR 99003342, "Mop Head Used to Collect Leak in 896' Tank Room Positioned Partially Across Contamination Barrier."

R1.2 Radiation Protection Issues with Fuel Inspections

a. Inspection Scope (71750)

The inspectors assessed the licensee's performance in the area of radiation protection during the new fuel inspections discussed in Section O1.2 of this report.

b. Observations and Findings

On November 9, 1999, the inspectors observed that personnel were not following the instructions contained in Radiation Work Permit (RWP) 700. The RWP instructions required personnel to wear rubber gloves and cotton liners when handling fuel over the spent fuel pool. Contrary to this, operators were only wearing surgical gloves when handling new fuel assemblies over the spent fuel pool in preparation for placing them into the pool. The radiation protection specialist was notified and personnel were directed to wear rubber gloves with cotton liners when working over the spent fuel pool. The radiation protection staff subsequently changed the RWP to allow the use of surgical gloves since the operators were handling new, uncontaminated fuel. This example of a failure to follow the RWP protective clothing requirements was contrary to Procedure 4AWI-08.04.01, Revision 10, "Radiation Protection Plan," and as such constituted a violation of minor significance of T.S. 6.5.D, "Plant Operating Procedures - Radiological," and is not subject to formal enforcement action.

Subsequently, the inspectors identified that a CR had not been written to document this issue. The inspectors discussed this with the general superintendent of radiation services, after which CR 19993400, "WORK CONTROL - Workers were observed not following requirements on RWP," was written. The initial failure to follow Procedure 4AWI-10.01.03, "Condition Report Process," constituted a violation of minor significance of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and is not subject to formal enforcement action.

c. Conclusions

The radiation work permit for placement of new fuel into the spent fuel pool was required to be modified to conform with actual practices when it was noted that the protective clothing identified within the radiation work permit was not being used.

P4 Staff Knowledge and Performance in EP (71750)

P4.1 Emergency Communicator Performance

During the retraction of a previously made non-emergency notification, discussed in Section O1.3 of this report, the inspectors observed that the shift emergency communicator was unfamiliar with the method to contact the NRC operations center on the dedicated emergency notification system (ENS) phone lines. The communicator thought that it was a direct line and did not require him to dial a number. When he realized it was not a direct line, he was unable to contact the NRC operations center using the ENS telephone line. The inspectors observed that 4 AWI-04.08.02, Revision 4, "10 CFR 50.72 and 10 CFR 50.73 Immediate Notifications," contained detailed instruction on how to contact the NRC operations center via the back-up/commercial telephone lines. However, the procedure for the primary/dedicated ENS phone line stated "use a dedicated line ... to notify the NRC," and did not include the specific number to dial. The inspectors were concerned that the emergency communicator was unfamiliar with the use of the ENS phones and that procedures used for notifications did not contain as detailed instructions for the ENS phones as it did for the backup method. The inspectors discussed these observations with the superintendent of emergency preparedness, who stated that use of the phones was covered in training and planned to discuss this issue with the emergency communicators.

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. The NRC inspectors noted no deficiencies with the performance of security activities.

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.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management, after the conclusion of the inspection, on December 17, 1999. 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
L. Wilkerson, Manager Quality Services
C. Schibonski, General Superintendent Safety Assessment
E. Sopkin, General Superintendent Operations
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 Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902:	Followup-Maintenance

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Closed

50-263/99-008	LER	Loss of speed control for reactor core isolation cooling system turbine due to loose gasket material preventing closure of governor valve

Discussed

None

LIST OF ACRONYMS USED

ATWS	Anticipated Transient Without SCRAM
AWI	Administrative Work Instruction
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
ENS	Emergency Notification System
EWI	Engineering Work Instruction
HPCI	High Pressure Coolant Injection
IFI	Inspection Followup Item
IP	Inspection Procedure
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSP	Northern States Power
OC	Operations Committee
PDR	Public Document Room
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RMA	Radioactive Materials Area
RWP	Radiation Work Permit
SBLC	Standby Liquid Control System
SRO	Senior Reactor Operator
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
WO	Work Order