June 7, 2000

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

SUBJECT: MONTICELLO INSPECTION REPORT 50-263/2000004(DRP)

Dear Mr. Wadley:

On May 16, 2000, the NRC completed a baseline inspection at your Monticello Nuclear Power Plant. The results of this inspection were discussed on May 16, 2000, with Mr. M. Hammer and other members of your staff. The enclosed report presents the results of that inspection.

The inspection was an examination of activities conducted under your license as they relate to reactor safety, verification of performance indicators, event followup, and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

The NRC identified two issues that were evaluated under the risk significance determination process and were determined to be of very low safety significance (Green). These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached report. Of the two issues identified, one was determined to involve a violation of Technical Specification requirements. However, a violation was not cited due to the very low safety significance of the issue. If you contest this noncited violation, 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 DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Monticello Nuclear Power Plant.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room and is available on the NRC Public Electronic Reading Room (PERR) link at the NRC home-page, 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 50-263/2000004(DRP)

cc w/encl: Site General Manager, Monticello
Plant Manager, Monticello
S. Minn, Commissioner, Minnesota
Department of Public Service

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room and is available on the NRC Public Electronic Reading Room (PERR) link at the NRC home-page, 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 50-263/2000004(DRP)

cc w/encl: Site General Manager, Monticello
Plant Manager, Monticello
S. Minn, Commissioner, Minnesota
Department of Public Service

DOCUMENT NAME: G:\Mont\mont2000004rpt
To receive a copy of this document, indicate in the box:"C" = Copy without enclosure "**E**"= Copy with enclosure"**N**"= No copy

OFFICE	RIII	E	RIII	N				
NAME	Kunowski:dp		Lanksbury					
DATE	06/ /00		06/ /00					

OFFICIAL RECORD COPY

ADAMS Distribution:
CMC1
WES
CFL (Project Mgr.)
J. Caldwell, RIII w/encl
B. Clayton, RIII w/encl
SRI Monticello w/encl
DRP w/encl
DRS w/encl
RIII PRR w/encl
PUBLIC IE-01 w/encl
Docket File w/encl
GREENS
RIII_IRTS
DOCDESK
JRK1
BAH3

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263/2000004(DRP)
Licensee:	Northern States Power Company
Facility:	Monticello Nuclear Power Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	April 2, through May 16, 2000
Inspectors:	Stephen Burton, Senior Resident Inspector Thomas Fredrichs, Resident Inspector Katherine Green-Bates, Regional Inspector Laura Collins, Resident Inspector Chris Miller, Senior Resident Inspector Paul Pelke, Regional Inspector Steven Ray, Senior Resident Inspector
Approved by:	Roger D. Lanksbury, Chief Reactor Projects Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

Monticello Nuclear Power Plant
NRC Inspection Report 50-263/2000004(DRP)

The report covers a 6½-week period of resident inspections. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

- GREEN. On February 21, 2000, the licensee identified a design deficiency associated with the Standby Gas Treatment System which would allow a small portion of secondary containment atmosphere to bypass the filter train during an event. The impact on offsite dose due to the additional bypass leakage pathway was determined to be minimal. The risk significance of this issue was very low since the effectiveness of filtration systems during severe accidents was small. The event was documented in Licensee Event Report (LER) 50-263/2000-005. (Section 4OA3.1)

- GREEN. On February 28, 2000, the licensee identified that the previous practice of removing the Standby Gas Treatment System control power impacted the ability of the secondary containment isolation dampers to perform their isolation function. The licensee's failure to understand the interactions between the Standby Gas Treatment System and the secondary containment isolation dampers resulted in the damper's eight-hour limiting condition for operation being exceeded on several occasions. The risk significance of this issue was very low since the effectiveness of secondary containment isolation systems during severe accidents was small. The inspectors identified a noncited violation for failure to comply with the requirements of Technical Specification 3.7.B.1.a. (Section 4OA3.2)

Report Details

Summary of Plant Status: Monticello operated at or near full power for the entire inspection period with one exception; on May 12, 2000, power was reduced to approximately 95 percent for 1 hour to accommodate control rod pattern adjustments.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed two partial walkdowns of redundant equipment trains while the counterpart trains were disabled due to planned maintenance. These systems were selected due to the significant increase in core damage frequency caused by taking one train out-of-service for maintenance. The inspectors verified the position of critical portions of the redundant equipment and looked for discrepancies between the existing equipment lineup and the required lineup.

The inspectors also performed a complete walkdown of a system, which was selected based on its considerable impact on the plant's accident mitigation capabilities. The inspectors verified: valve lineup requirements, material condition, electrical lineups, component labeling, seismic and piping supports, support system operability, and area external interferences. The inspection activities were:

- A partial walkdown of the 'A' train of the Control Room HVAC [heating, ventilation and air conditioning] System while the 'B' train Control Room HVAC System was out-of-service for the performance of emergent maintenance work;

- A partial walkdown of the Screen Wash/Fire Pump and the Diesel Fire Pump while the Electric Fire Pump was out-of-service for planned maintenance; and

- A complete walkdown of the accessible portions of the Standby Liquid Control System. As part of this inspection, outstanding work orders and condition reports were also reviewed for operability concerns.

b. Issues and Findings

There were no findings identified during this inspection.

1R05 Fire Protection

.1 Fire Zone Walkdown

a. Inspection Scope

The inspectors walked down selected risk significant areas looking for any fire protection issues related to: the control of transient combustibles, ignition sources, fire detection equipment manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, and barriers to fire propagation. The areas walked down were:

- Fire Zone 3-B (standby liquid control area);
- Fire Zone 1-A (12 RHR [residual heat removal] & core spray pump room); and
- Fire Zone 1-B (11 RHR & core spray pump room).

b. Issues and Findings

There were no findings identified during this inspection.

.2 Annual Observation of Fire Drill

a. Inspection Scope

The inspectors performed an annual observation of a plant fire drill to evaluate the licensee's readiness to prevent and fight fires. The use of protective clothing, turnout gear, self-contained breathing apparatus, fire equipment, fire fighting techniques, communications, and fire strategies were also observed. The following documents were reviewed as part of this inspection:

- Procedure 2176, Revision 10, "Fire Drill Procedure";
- "Individual Plant Examination of External Events," NSPLMI-95001, Revision 1;
- "Monticello Nuclear Plant - Fire Hazards Analysis"; and
- "Maintenance Rule System Basis Document for Fire Protection System," Revision 2.

b. Issues and Findings

There were no findings identified during this inspection.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed the performance of a training crew during a simulator exam scenario. The scenario included a Condensate and High Pressure Coolant Injection (HPCI) system malfunction which resulted in reactor level control problems and a failure to scram when a reactor low water level condition occurred. Areas observed by the inspectors included: clarity and formality of communications, timeliness of actions, prioritization, procedural implementation, control board manipulations, managerial oversight, emergency plan execution, and group dynamics.

b. Issues and Findings

There were no findings identified during this inspection.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule requirements, including a review of scoping, goal-setting, and performance monitoring, short-term and long-term corrective actions, functional failure determinations associated with the below-listed condition reports and current equipment performance status. The systems selected for inspection were all classified as risk significant by the licensee's maintenance rule program. The systems evaluated and condition reports reviewed included:

- The Emergency Diesel Generator (EDG) system;
- The RHR System;
- The RHR Service Water System;
- The Core Spray System;
- Condition Report 20000308, "Found circulating oil pump for 12 EDG air bound";
- Condition Report 20000382, "Start failure occurred on 12 EDG on first start attempt";
- Condition Report 20000147, "Phase imbalance found on motor for [motor-operated valve] MO-1752, 12 core spray injection outboard valve";
- Evaluation of Motor Performance Parameters for MOV [motor-operated valve] 1752 (Motor Serial Number 1YFB28615A1 WJ), dated February 29, 2000;
- Condition Report 19991464, "Valve MO-2003 failed to open on demand from control room hand-switch";

- Condition Report 20000127, “During shutdown cooling on B RHR loop MO-2003 would not remain in a throttled position”; and

- Condition Report 19993612, “11 RHR Service Water Pump Did Not Meet Its Maintenance Rule Action Plan.”

b. Issues and Findings

There were no findings identified during this inspection.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed and observed emergent work, preventive maintenance activities, and associated documentation that involved risk significant systems. The inspectors also reviewed the licensee’s evaluation of plant risk, scheduling, and configuration control for these activities in coordination with other scheduled risk significant work. The work observed and documents reviewed were:

- Work Order (WO) 0001857 “Investigate/Repair Motor on Control Room Ventilation Supply Cross Connect Damper VD-9093B”;

- Electric Fire Pump P-110 in accordance with WO 0001231, and Electric Fire Pump Air Vent Valve AV-1966 in accordance with WO 0001305;

- Operations Manual Procedure B.08.05-01, Revision 4, “Fire Protection”; and

- Piping and Instrumentation Diagram M-812, “Screen Wash, Fire & Chlorination System Intake Structure.”

b. Issues and Findings

There were no findings identified during this inspection.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the technical adequacy of operability evaluations to determine the impact on Technical Specifications, significance of evaluations, and proper justification of operability. Operability evaluations were selected based upon the relationship of the safety-related system, structure, or component to risk. The operability evaluations reviewed were:

- Condition Report 20001447, “SBLC [standby liquid control] RV-11-39B had continuous leakage after pumps were shutdown”;

- Condition Report 20001536, “NRC Walkdown of SBLC System Results in Several Items to Be Addressed”; and

- Condition Report 20001944, “SBGT [standby gas treatment] Room Temperature at 103 Degrees F During SBGT Test 0253 With Outside Air Temperature at 78F. Max per Bechtel Spec 104."

b. Issues and Findings

There were no findings identified during this inspection.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed an operator workaround identified as a result of Condition Report 20001486, “In a Station Blackout Event, B4300 Will Attempt to Close With Low Control Power Available, Likely Resulting In A Blown Control Fuse.” The condition report documented that voltage for breaker control power in a station blackout would not be sufficient to properly operate the low pressure coolant injection swing bus alternate power supply breaker (B4300). The inspectors reviewed the licensee’s temporary procedure change to Operations Manual Abnormal Procedure C.4-B.9.02A, “STATION BLACKOUT,” to have operators restore the swing bus from the normal supply.

b. Issues and Findings

There were no findings identified during this inspection.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors observed the performance of post-maintenance testing activities including: integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural usage, control of temporary modifications or jumpers required for test performance, documentation of test data, technical specification applicability, system restoration, and evaluation of test data. The following post-maintenance testing on risk significant equipment was observed:

- Preventive maintenance on Electric Fire Pump P-110 in accordance with WO 0001231; and

- Preventive maintenance on “B OG [Offgas] Dilution Fan” supply breaker in accordance with WO 0002064.

b. Issues and Findings

There were no findings identified during this inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed the performance of surveillance testing activities including: reviews for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, Technical Specification applicability, impact of testing relative to performance indicator reporting, and evaluation of test data. The following surveillance testing on risk significant equipment was observed:

- Surveillance Test Procedure 0143, Revision 26, “Drywell - Torus Monthly Vacuum Breaker Check and Instrument Air System Valve Exercise”;

- Surveillance Test Procedure 0060, Revision 20, “RCIC [Reactor Core Isolation Cooling] Hi Steam Flow Sensor Test and Calibration”; and

- Surveillance Test Procedures 0397A, 0400A, 0403A, and 0405A, “SRV [Safety Relief Valve] Low-Low Set System Quarterly Tests.”

b. Issues and Findings

There were no findings identified during this inspection.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification package, safety evaluation, and installation work orders associated with the temporary modifications listed below:

- 99-89 Non-documented filtering capacitor installed in the HPCI square-rooter;

- 98-109 Temporary repair of V-EAC-14B compressor unloader solenoid return line; and

- 00-05 Input cables from LPRMs [local power range monitors] 2045B and 3613B removed from back of corresponding APRMs [average power range monitors].

b. Issues and Findings

There were no findings identified during this inspection.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Cornerstones: Mitigating Systems and Initiating Events

.1 Safety System Unavailability, High Pressure Coolant Injection System

a. Inspection Scope

The inspectors verified the performance indicator data for "Safety System Unavailability - High Pressure Coolant Injection System" from April 1, 1999, through March 31, 2000. This was accomplished, in part, through evaluation of the Limiting Conditions for Operation Log times for HPCI and required support systems, applicable work orders, condition reports, system safety tagging, and discussions with licensee personnel.

b. Issues and Findings

The inspectors identified that WO 20000636, "[Pressure Control Valve] PCV-4214 Has A Diaphragm Leak," was not evaluated for the failure of the diaphragm on past operability. Pressure Control Valve PCV-4214 was the HPCI system lubricating oil PCV. The inspectors determined that failure of the diaphragm in this valve would result in loss of lubricating oil and loss of associated pressure control capabilities. The inspectors interviewed system engineering personnel to identify the nature of the failure and obtain an assessment for fault exposure time.

System engineering personnel indicated that a visual inspection showed that the diaphragm in PCV-4214 had failed due to cracking and perforations caused by age-related degradation. The inspectors found that system engineering personnel had not formally evaluated the failure mechanism, evaluated the impact of the failure, or considered the extent of the failure with respect to other installed equipment. Because a root cause evaluation for the failure mechanism had not been performed, the licensee was unable to provide evidence that the valve would not have experienced full failure under operating conditions. The inspectors concluded that if the valve could have failed under operating conditions, fault exposure time would have to be calculated for the HPCI system.

The licensee initiated Condition Report 20001977, "Documentation of Minor PCV-4214 Diaphragm Leak Corrected During 2000 Outage," to evaluate the impact of the failure on past system operability. The verification of this performance indicator was considered an unresolved item (URI 50-263/2000004-01(DRP)), "Potential Error In Performance Indicator Reporting Data For 'Safety System Unavailability - HPCI System'," pending the results of the licensee's review of the valve PCV-4214 failure mechanisms.

.2 Unplanned Power Changes per 7000 Critical Hours

a. Inspection Scope

The inspectors verified the performance indicator data for unplanned power changes per 7000 critical hours from April 1, 1999, through March 31, 2000. This was accomplished, in part, through evaluation of control room logs, monthly operating reports, work orders, and condition reports for conditions resulting in power changes.

b. Issues and Findings

There were no findings identified during this inspection.

4OA3 Event Follow-up

Cornerstones: Mitigating Systems and Barrier Integrity

.1 Design Deficiency in Standby Gas Treatment System

a. Inspection Scope

The inspectors evaluated Licensee Event Report (LER) 50-263/2000-005, "Design Deficiency Results In Secondary Containment Leakage Pathway Which Bypasses Standby Gas Treatment System Filters."

b. Issues and Findings

The licensee identified an original design issue that impacted the ability of the SBGT system to process all radioactive material released to the secondary containment due to potential bypass pathways. The SBGT system room, which was external and attached to secondary containment, was designed to stay at a positive pressure in relation to the secondary containment during SBGT system operation. On February 21, 2000, the licensee identified that the SBGT system room did not stay at a positive pressure during system operation. As a result, any radioactive material present in the secondary containment during severe accident conditions may not have been treated by the SBGT system filters.

The LER documented the licensee's evaluation that the amount of radioactive material bypassing the SBGT system filters would not significantly impact the contribution to total dose for personnel in the control room, at the exclusion area boundary, nor at the low population zone. Additionally, the LER indicated that the doses remained well below the 10 CFR Part 50, Appendix A, General Design Criteria 19, and 10 CFR Part 100 guidelines. This determination was based upon two premises. First, the accidents impacting this issue were a loss of coolant accident, an anticipated transient without scram, and a refueling accident. Second, the licensee assumed an increased in-leakage of 125 scfm (standard cubic feet per minute) from the secondary containment into the SBGT room. The inspectors review of the Safety Analysis Report supported the licensee's conclusion that the loss of coolant accident resulted in the highest release rates which bounded the analysis for the remaining events. The inspectors were unable

to validate the basis for the second assumption because the licensee had not completed documentation of the related data.

The inspectors and senior reactor analysts (SRAs) screened this finding using the significance determination process (SDP). During the Phase 1 screening, the inspectors determined that a Phase 2 screening was required for this issue because it could affect the operability or function of the SBGT system, and impact the integrity of secondary containment. In conjunction with the SRAs, the inspectors initiated a Phase 2 screening.

During the SDP Phase 2 review, the inspectors determined that this finding was unrelated to a safety-related structure, system, or component (SSC) that was needed to prevent accidents from leading to core damage, but had important implications for the integrity of the containment. Inspection Manual Chapter MC 0609, "Significance Determination Process," Appendix H, "Containment Integrity SDP," does not specifically address this finding; therefore, a Phase 3 SDP was initiated

The inspectors and SRAs performed a Phase 3 analysis and identified that filtration systems, including the SBGT system, were within the scope of SSCs reviewed for their impact on large early release frequency. However, the risk significance of this issue was very low because the effectiveness of filtration systems in severe accidents is small. The dominant contributors to risk tend to be accident sequences in which the associated support systems (e.g., alternating current power) are unavailable, accident sequences that bypass the filtration systems (e.g., venting via the hardened vent or releases via the blowout panels), or accident sequences that would challenge the filtration systems by high aerosol loads. Therefore, even if findings related to filtration systems and secondary containment integrity exceeded 10 CFR Part 100 limits, they were not considered significant in terms of overall risk. Based upon this information the inspectors determined the event was within the licensee response band (green). In addition, the inspectors determined that a violation of requirements did not exist for this issue since the design deficiency occurred prior to the implementation of 10 CFR Part 50, Appendix B.

.2 Loss of Isolation Signal to Secondary Containment Dampers Not Recognized

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-006, "Procedural Inadequacy Results in Loss of Isolation Signal To Secondary Containment Dampers When Standby Gas Treatment System Train Control Power De-energize."

b. Issues and Findings

As part of the quarterly SBGT system fan maintenance, the licensee de-energized the SBGT system control power. The licensee entered Technical Specification 3.7.B.1.a which allowed the SBGT system to be inoperable for 7 days. On February 28, 2000, the licensee identified that de-energizing the SBGT system control power prevented the automatic closure of multiple secondary containment isolation dampers which required entry into Technical Specification 3.7.C.3. Technical Specification 3.7.C.3 stated that

actions must be taken to restore the damper(s) to an operable status or isolate the affected duct(s) within 8 hours. Failure to complete these actions required the licensee to initiate an orderly plant shutdown and achieve cold shutdown within 36 hours. Since the relationship between the SBGT system control power and the secondary containment isolation dampers was not recognized by the licensee, operations personnel failed to enter Technical Specification 3.7.C.3 or comply with the limiting conditions for operation.

Similar to the findings in Section 4OA3.1, the inspectors and SRAs performed a Phase 1 SDP. During the Phase 1 screening, the inspectors determined that a Phase 2 screening was required for this issue because the loss of secondary containment isolation ability could affect the operability or function for Secondary Containment Isolation, and impact the integrity of secondary containment. In conjunction with the SRAs, the inspectors initiated a Phase 2 screening.

During the SDP Phase 2 review, the inspectors determined that this finding was unrelated to an SSC that was needed to prevent accidents from leading to core damage, but had important implications for the integrity of the containment. Inspection Manual Chapter MC 0609, "Significance Determination Process," Appendix H, "Containment Integrity SDP," does not specifically address this finding therefore, a Phase 3 SDP was initiated

The inspectors and SRAs performed a Phase 3 analysis and identified that secondary containment isolation valves and dampers were within the scope of SSCs reviewed for their impact on large early release frequency. However, the risk significance of this issue was very low because the effectiveness of secondary containment isolation systems in severe accidents was small. The dominant contributors to risk tend to be accident sequences in which the associated support systems (e.g., alternating current power) are unavailable, accident sequences that bypass the filtration systems (e.g., venting via the hardened vent or releases via the blowout panels), or accident sequences that would challenge the filtration systems by high aerosol loads. Therefore, even if findings related to secondary containment integrity exceeded 10 CFR Part 100 limits, the findings were not considered significant in terms of overall risk. Based upon this information, the inspectors determined that the event was within the licensee response band (green). However, the inspectors determined that the failure to restore each secondary containment isolation damper to an operable status or isolate the affected duct(s) within 8 hours and the failure to initiate an orderly plant shutdown was a violation of Technical Specification 3.7.B.1.a. This Severity Level IV violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-263/2000004-02(DRP)). This issue was entered into the licensee's corrective action program as CR 20000903, "SBGT Room Found To Be At Lower Pressure Than Rx Building During WO 9908430 and Test 0151-1."

.3 Missed Fire Watch

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-007: "Procedural error results in missed periodic fire watch."

b. Issues and Findings

With the plant operating at 98 percent power, a surveillance test was performed on the HPCI System. The test required that the suppression pool (torus) level be lowered by pumping water to radwaste through motor-operated valve MO-2032. The valve used to perform the draining operation was normally maintained closed with its breaker open to prevent spurious operation in the event of a fire, a condition which could lead to subsequent draining of the torus. In order to compensate for the increased risk when the valve was used, the operating procedure requires a roving hourly fire patrol to be maintained while the valve breaker was closed. However, during the surveillance test performed on March 3, 2000, the licensee allowed approximately 2 hours and 5 minutes to elapse between fire patrols of the affected areas. The valve and breaker were returned to their normal positions at the end of the surveillance test.

The licensee determined that the cause of the error was a cognitive error on the part of a licensed operator and placed the event in its corrective action program. The licensee's corrective actions were being tracked in Condition Report 20001092, "Fire watch patrol interval of > 1 hour when required by Technical Specifications to be completed within 1 hour." The failure to perform hourly fire watch patrols during the surveillance test, as required, is considered a violation of minor significance and is not subject to formal enforcement action.

4OA5 Other

.1 Temporary Instruction 2515/144, "Performance Indicator Data Collecting and Reporting Process"

a. Inspection Scope

The inspectors reviewed the performance indicator data collecting and reporting process for the "Initiating Events - Unplanned Power Changes per 7000 Critical Hours," and for the "Mitigating Systems - Safety System Unavailability - High Pressure Coolant Injection System," performance indicators. This temporary instruction was conducted in conjunction with the performance indicator verifications performed per Inspection Procedure 71151, "Performance Indicator Verification" and documented in Section 4OA1 of this report. Included was a review of: data collecting and reporting process, indicator definitions, data reporting elements, calculational methods, definition of terms, clarifying notes used by the licensee for consistency with industry guidance document NEI [Nuclear Energy Institute] 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0. The licensee procedures reviewed were:

- Administrative Work Instruction 4AWI-04.08.11, Revision 1, "NRC Performance Indicator Reporting"; and

- Procedure SGP-03.07, Revision 0, "NRC Mitigating Safety System Data Collection and Reporting."

b. Issues and Findings

The inspectors concluded that Procedure SGP-03.07, “Safety System Unavailability - High Pressure Coolant Injection System,” contained sufficient information for personnel compiling data to adequately report the performance indicator.

The data collection methods utilized by the engineering staff for the “Unplanned Power Changes per 7000 Critical Hours” performance indicator were governed by informal Nuclear Engineering group procedures. Additionally, the safety assessment department was developing a procedure for the collection and documentation of this information. These procedures, combined with engineer knowledge, were adequate to provide data necessary for the development of the related performance indicator.

4OA6 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Hammer and other members of licensee management at the conclusion of the inspection on May 16, 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
B. Linde, Superintendent, Security
B. Sawatzke, General Superintendent, Maintenance
C. Schibonski, General Superintendent, Safety Assessment
E. Sopkin, General Superintendent, Operations
L. Wilkerson, Manger, Quality Services
J. Windschill, General Superintendent, Radiation Services

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-263/2000004-01	URI	Potential error in performance indicator reporting data for “Safety System Unavailability - HPCI System”
50-263/2000004-02	NCV	Technical Specification Actions for Secondary Containment Dampers Inoperable Not Performed Within the Time Limits of the LCO

Closed

50-263/2000-005	LER	Design deficiency results in secondary containment leakage pathway which bypass standby gas treatment [SBGT] system filters
50-263/2000-006	LER	Procedural inadequacy results in loss of isolation signal to secondary containment dampers when standby gas treatment system train control power deenergized
50-263/2000-007	LER	Procedural error results in missed periodic fire watch
50-263/2000004-02	NCV	Technical Specification Actions for Secondary Containment Dampers Inoperable Not Performed Within the Time Limits of the LCO

Discussed

None

LIST OF ACRONYMS USED

BWR	Boiling Water Reactor
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation, and Air Conditioning
LER	Licensee Event Report
PCV	Pressure Control Valve
RHR	Residual Heat Removal
SBGT	Standby Gas Treatment
SBLC	Standby Liquid Control
SDP	Significant Determination Process
SRA	Senior Reactor Analysts
SSC	Safety-Related Structure, System, or Component
URI	Unresolved Item
WO	Work Order

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC-licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	Radiation Safety	Safeguards
● Initiating Events	● Occupational	● Physical Protection
● Mitigating Systems	● Public	
● Barrier Integrity		
● Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.