September 5, 2000

Mr. M. Wadley
Senior Vice President and Chief Nuclear Officer
Nuclear Management Company
700 First Street
Hudson, WI 54016

SUBJECT: MONTICELLO NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-263/2000006(DRP)

Dear Mr. Wadley:

On August 15, 2000, the NRC completed a baseline inspection at your Monticello Nuclear Power Plant. The results of this inspection were discussed on August 15, 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 one issue that was evaluated under the risk significance determination process and was determined to be of very low safety significance (Green). This issue has been entered into your corrective action program and is discussed in the summary of findings and in the body of the attached report.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS)*. *ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Roger D. Lanksbury, Chief
Reactor Projects Branch 5

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

Enclosure: Inspection Report 50-263-00-06(DRP)

cc w/encl: Site General Manager, Monticello
Plant Manager, Monticello
J. Malcolm, Commissioner, Minnesota
Department of Health

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No: License No:	50-263 DPR-22
Report No:	50-263-00-06(DRP)
Licensee:	Nuclear Management Corporation, LLC
Facility:	Monticello Nuclear Power Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	July 1, through August 15, 2000
Inspectors:	Stephen Burton, Senior Resident Inspector Daniel Kimble, Resident Inspector Karla Stoedter, Regional Inspector Paul Pelke, Regional Inspector Michael Kunowski, Regional Inspector
Approved by:	Roger D. Lanksbury, Chief Reactor Projects Branch 5 Division of Reactor Projects

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.

SUMMARY OF FINDINGS

Monticello Nuclear Power Plant
NRC Inspection Report 50-263-00-06(DRP)

Inspection Report 50-263-00-06(DRP), on 07/01-08/15/2000; Nuclear Management Corporation, LLC; Monticello Nuclear Power Plant;, Operator Workarounds.

The inspection was conducted by resident inspectors and regional projects inspectors. The report covers a 6½-week period of resident inspection. This inspection identified one Green issue. The significance of issues is identified by their color (Green, White, Yellow, Red) and was determined by the Significance Determination Process.

GREEN. The inspectors identified that the licensee may be unable to implement compensatory actions for operator workarounds associated with the inboard residual heat removal to waste surge tank valve due to the inaccessibility of plant areas during accident conditions.

The risk significance of this issue was determined to be very low because Emergency Operating Procedures provided alternate actions that would be taken in the event that the compensatory actions for valve operation were unsuccessful. The alternate actions would have assured that core cooling would have been maintained. (Section 1R16)

Report Details

Summary of Plant Status: Monticello operated at or near full power for the entire inspection period.

1. REACTOR SAFETY

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

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial walkdown of the following redundant equipment trains to verify operability and proper equipment lineup while the counterpart train was disabled due to planned maintenance. The systems were selected due to the increase in core damage frequency caused by rendering one train out-of-service for maintenance.

- 12 Control Rod Drive (CRD) Pump and associated components while the 11 CRD Pump was out-of-service for the performance of emergent maintenance work.

- 11 Emergency Diesel Generator (EDG) while the 12 EDG was unavailable.

- 11 Emergency Service Water System (ESW) while the 12 ESW was out-of-service for maintenance.

The inspectors verified the position of critical portions of the redundant equipment and looked for any discrepancies between the existing equipment lineup and the required lineup. The documents reviewed included:

- Operations Manual B.1.3, “CRD Hydraulic System”

- Updated Safety Analysis Report (USAR), Revision 17, Section 3.5.3, “Control Rod Drive System”

- Fire Watch Patrol Form for Work Order (WO) 0002881

- Post-Maintenance Test Form for WO 0002881

- Surveillance Test 0187-02, Revision 32, “12 Emergency Diesel Generator/ 12 Emergency Service Water Pump System Test”

- Equipment Isolations:
  - 00-02772, Version 1, “Replace 11 CRD Pump Bearings/Rebuild as necessary”
  - 00-02881, Version 1, “Repair or Replace 12 ESW Pump Due to Degrading DP [differential pressure]”

- Piping and Instrument Diagrams (P&IDs):
  - P&ID M-112, Revision BF, “RHR [residual heat removal] Service Water and Emergency Service Water System”
  - P&ID M-811, Revision C, “Service Water and Make-up Water Intake Structure”
  - P&ID M-118, Revision AU, “Control Rod Hydraulic System”

- WOs:
  - WO 0002772, “Replace 11 CRD Pump Bearings/Rebuild as necessary”
  - WO 0002881, “Repair or Replace 12 ESW Pump Due to Degrading DP”

b. Issues and Findings

There were no findings identified during this inspection.

1R05 Fire Zone Walkdown

a. Inspection Scope

The inspectors walked down the following risk significant areas looking for any fire protection issues. The inspectors selected areas containing systems, structures, or components that the licensee identified as important to reactor safety.

- Fire Zone 7-A (125V [volt] Division I Battery Room)

- Fire Zone 7-B (250V Division I Battery Room)

- Fire Zone 31-B (EFT [emergency filtration train] Building - 1st Floor (Division II))

- Fire Zone 1-C (RCIC [reactor core isolation cooling] Room)

- Fire Zone 1-E (HPCI [high pressure coolant injection] Room - Reactor Building Elevation 896')

The inspectors reviewed the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, and barriers to fire propagation. The documents reviewed included:

- Monticello Fire Strategies:
  - A.3-07-A, Revision 2, “Fire Zone 7-A, 125V Division I Battery Room”
  - A.3-07-B, Revision 4, “Fire Zone 7-B, 250V Division I Battery Room”

- A.3-31-B, Revision 4, “Fire Zone 31-B, “EFT Building 1st Floor (Division II)”
- A.3-01-C, Revision 2, “Fire Zone 1-C, RCIC Room”
- A.3-01-E, Revision 3, “Fire Zone 1-E, HPCI Room - Reactor Building Elevation 896'”

• Procedures:
  - Administrative Work Instruction (AWI) 4AWI-08.01.01, Revision 14, “Fire Prevention Practices”
  - 0271, Revision 24, “Fire Hose Station and Yard Hydrant Hose House Equipment Inspection”
  - 0274, Revision 16, “Fire Hose Hydrostatic Test Interior Hose Stations,” completed July 6, 1999
  - 0275-1, Revision 8, “Fire Barrier Penetration Seal Visual Inspection”
  - 0275-2, Revision 15, “Fire Barrier Wall, Damper, and Floor Inspection”

• Drawings:
  - NX-16991-14, Revision A, “Monticello Nuclear Generating Plant Fire Hazards Analysis Plan View, Administration Building, Elevation 928'-0"”
  - NX-16991-15, Revision A, “Monticello Nuclear Generating Plant Fire Hazards Analysis Plan View, Administration Building, Elevation 939'-0"”
  - NX-16991-47, Revision A, “Monticello Nuclear Generating Plant Fire Penetration Seal Locations”

• Technical Manual NX-16991, “Monticello Updated Fire Hazards Analysis”

• Quadrex Corporation Report QUAD-5-80-009, Revision 7, “Specifications for Installation of Electrical and Mechanical Penetration Seals at the Monticello Nuclear Generating Plant”

b. Issues and Findings

There were no findings identified during this inspection.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed the performance of a training crew during a simulator exam scenario and evaluated licensed operator performance in mitigating the consequences of events. The scenario included a stuck open reactor relief valve with a broken tail pipe. The transient resulted in a reactor scram complicated by containment pressure and temperature control problems. Areas observed by the inspectors included: clarity and formality of communications, timeliness of actions, prioritization of activities, procedural adequacy and implementation, control board manipulations, managerial oversight, emergency plan execution, and group dynamics. Documents reviewed by the inspectors included:

- Monticello Simulator Scenario RQ-SS-04E, Revision 6, "SORV [solenoid operated relief valve] With Tailpipe Break"

- Alarm Response Procedures:
  - C.6-006-B-18, Revision 4, "Cond Service Wtr Low Pressure"
  - C.6-007-B-22, Revision 1, "CW [circulating water] Pump Pit Flood CW Pump Trip"

- Abnormal Operating Procedures:
  - B.01.04-05.03.B.3, Revision 8, "Jet Pump Failure"
  - C.4-B.06.02.04.A, Revision 5, "Stator Cooling Water Failure"
  - C.4-B.06.04.A, Revision 7, "Decreased Circulating Water"
  - C.4-I, Revision 2, "Plant Flooding"
  - C.4-B.04.01.A, Revision 9, "Primary Containment Isolation - Group 1"
  - C.4-B.04.01.B, Revision 17, "Primary Containment Isolation - Group 2"
  - C.4-B.04.01.C, Revision 8, "Primary Containment Isolation - Group 3"
  - C.4-A, Revision 16, "Reactor Scram"
  - C.4.B.03.03.A, Revision 9, "Stuck Open Relief Valve"

- Emergency Operating Procedures (EOPs):
  - C.5.1 - 1100, Revision 6, "RPV [Reactor Pressure Vessel] Control"
  - C.5.1 - 1200, Revision 8, "Primary Containment Control"
  - C.5-3205, Revision 0, "Terminate and Prevent"
  - C.5-3202, Revision 3, "Containment Spray"

b. <u>Issues and Findings</u>

There were no findings identified during this inspection.

1R12 <u>Maintenance Rule Implementation</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) to ensure rule requirements were met for the selected systems. The following systems were selected based on their being designated as risk significant under the Maintenance Rule, or their being in the increased monitoring (Maintenance Rule category a(1)) group:

- 480-Volt Alternating Current (VAC)

- High Pressure Coolant Injection System

- Standby Liquid Control System

The inspectors verified the licensee's categorization of specific issues, including evaluation of the performance criteria and, when applicable, goal-setting established for the systems listed above. 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 condition reports listed below, and current equipment performance status. The documents reviewed included:

- NUMARC [Nuclear Management and Resources Council] 93-01, Revision 2, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

- Regulatory Guide 1.1.6, Revision 1, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

- Engineering Work Instruction (EWI) 05.02.01, Revision 3, "Monticello Maintenance Rule Program Document"

- Monticello Maintenance Rule Periodic Assessment Report, 4th Quarter - 1999;

- Operations Manual B.3.5, "Standby Liquid Control [SBLC] System"

- USAR, Revision 17, Section 6.6, "Standby Liquid Control System"

- Technical Specification (TS) 3/4.4, "Standby Liquid Control System"

- Condition Reports (CRs):
  - CR 20002786, "Worn Pin in Safety Related Circuit During Major Maintenance as a Result of an Improperly Installed Snap Ring"
  - CR 20002876, "Broken Cable Strands Found in [breaker] B2215 During Performance of Preventive Maintenance"
  - CR 20000208, "HPCI CV-2065 air accumulator Check Valve failed leak rate test"
  - CR 20000556, "During calibration of LS-23-90, HPCI STM SUPPLY DRN HI LVL B/P, the switch failed to trip during initial cal attempt"
  - CR 20002110, "HPCI overspeed reset time not in accordance with tech manual"
  - CR 20002541, "TIS-7268, HPCI Lube Oil High Temp Alarm, configuration not per the tech manual, NX-8292-54"
  - CR 19990881, "Unplanned LCO [Limiting Condition for Operation] due to Water Spraying on #12 SBLC Pump from [sightglass] FG-2630 Leakage"
  - CR 19992912, "Unplanned LCO Entry due to Leak on 12 SBLC Accumulator Requiring 12 SBLC to be Declared Inoperable"
  - CR 20001447, "SBLC RV-11-39B had Continuous Leakage after Pumps were Shutdown"
  - CR 20003088, "Alarm Response Procedure for 5-B-15, STANDBY LIQUID HI/LO TEMP, was not followed correctly on January 7, 1999"
  - CR 20003090, "CR 19990881 was not coded as a Maintenance Rule Functional Failure although it was counted as such in the Maintenance Rule"

- Monticello Maintenance Rule Program System Basis Document:
  - "High Pressure Coolant Injection System - B.3.2," Revision 1

- “480 VAC Station Auxiliary - B.9.7,” Revision 2
- “Standby Liquid Control System - B.3.5,” Revision 1

- P&IDs:
  - P&ID M-124, Revision Y, “High Pressure Coolant Injection System (Water Side)"
  - P&ID M-127, Revision V, “Standby Liquid Control System”

- WOs:
  - WO 0001371, “SBLC PI [pressure indicator] Power Supply Not Functioning Properly”
  - WO 0001061, “Indicated [SBLC Tank] Level Lower Than Actual”
  - WO 0001098, “Indicated [SBLC Tank] Level Off Scale High”
  - WO 0000691, “Blown Fuses on SBLC System”

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 or preventive maintenance activities on selected systems. The inspectors selected the following risk significant systems undergoing scheduled or emergent maintenance:

- Observations of selected portions of the 11 CRD Pump bearing replacement work-in-progress and associated post-maintenance testing.

- Observations of selected work-in-progress and associated post-maintenance testing portions for maintenance of the Division 1, 250-VDC [Volt Direct Current] battery to correct degraded conditions.

- Observations of selected work-in-progress and associated post-maintenance testing portions for maintenance to correct degraded conditions on 12 ESW pump.

The inspectors also reviewed the licensee’s evaluation of plant risk, risk management, scheduling, and configuration control for these activities in coordination with other scheduled risk significant work. The inspectors verified that the licensee’s control of activities considered assessment of baseline and cumulative risk, management of plant configuration, and control of maintenance. The documents reviewed included:

- TS 3/4.13, "Fire Detection and Protection Systems"

- Operations Manual B.1.3, “CRD Hydraulic System”

- Operations Manual B.8.5, “Fire Protection"

- Preventative Maintenance Procedure 4077PM, Revision 1, “CRD Pump Rebuild”

- Vendor Technical Manual NX-16923, "Crispin Air Valves Multiplex MF6"

- Equipment Isolations:
  - 00-02772, Version 1, “Replace 11 CRD Pump Bearings / Rebuild as necessary”
  - 00-02778, Version 1, “Repair Diesel Fire Pump Air Vent Valve, AV-1937”

- WOs:
  - WO 0002772, “Replace 11 CRD Pump Bearings / Rebuild as necessary”
  - WO 0002778, “Repair Diesel Fire Pump Air Vent Valve, AV-1937”
  - WO0002690, “Install Temporary Cell Charger on #13 Battery”

- P&IDs:
  - P&ID M-118, Revision AU, “Control Rod Hydraulic System”
  - P&ID M-812, Revision RQ, “Screen Wash, Fire & Intake Structure”

- USAR, Revision 17:
  - Section 3.5.3, “Control Rod Drive System”
  - Section 10.3.1, “Fire Protection System”

- CR 20002076, “#13 Battery 250VDC Monthly Operability Check Found Low Voltage on Cell #81 (Procedure 0193-01)”

b. <u>Issues and Findings</u>

There were no findings identified during this inspection.

1R15 <u>Operability Evaluations</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the technical adequacy of operability evaluations to determine the impact on TSs, the significance of the evaluations, and that adequate justifications of operability were documented. Operability evaluations were selected based upon the relationship of the safety-related system, structure, or component to risk. The operability evaluation and associated references reviewed were:

- CR 20002448, “'A' RHRSW [residual heat removal service water] loop Inoperability not immediately considered and CR not initiated in response to unexpected LCO entry 9/17/99”

- P&IDs:
  - P&ID M-112, Revision BF, “RHR Service Water and Emergency Service Water Systems”
  - P&ID M-811, Revision CA, “Service Water System and Make-Up Intake Structure”

b. Issues and Findings

There were no findings identified during this inspection.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed operator workaround (OWA) 6, “Motor Operated Valve (MO) 2032 is Inoperable and Must be Operated Manually” and Non-Transient OWA 8, “Closing Breakers B-4211(MO-2032) Requires an Hourly Fire Watch.” The inspectors reviewed each workaround’s potential to impact the operators’ ability to implement emergency or abnormal operating procedures.

The inspectors also performed a semiannual review of the cumulative effects of operator Workarounds. The inspectors reviewed the cumulative effects of Workarounds on the reliability, availability, and potential for improper operation of the system. The inspectors also evaluated the Workarounds to determine it they could increase the possibility of an initiating event, affect multiple mitigating systems, or impact the operators’ ability to respond to accidents or transients. The documents reviewed included:

- Monticello Operational Challenges List, dated June 28, 2000
- 4AWI-04.01.01, Revision 25, “General Plant Operational Practices”
- USAR, Section 14.9.7.3, Revision 17, “Reactor Building Accessibility”
- Operations Manual:
  - C5.1-2002, Revision 2, “Reactor Pressure Vessel Lowdown”
  - C.5.1-1200, Revision 6, “Primary Containment Control”
  - C.5-3402, Revision 0, “Draining Torus Water to Radwaste”
  - B.3.4-01, Revision 2, “Residual Heat Removal System”
  - B.3.4-02, Revision 9, “Residual Heat Removal System Description of Equipment”
- P&ID M-121, Revision BE, “Residual Heat Removal System”

b. Issues and Findings

Operations personnel normally operated MO-2032 to control torus water level by manipulating a hand-switch located in the control room. The licensee identified that MO-2032 could inadvertently open during a fire which could drain the RHR system and jeopardize safe shutdown capabilities, an Appendix R concern. To remedy this, the licensee established Non-Transient OWA 8, which maintained MO-2032 in the closed position with the associated breaker left open. As a result, an operator was required to manually close the breaker for MO-2032 before operations personnel could operate the valve from the control room.

In June 2000, operations personnel identified that the motor operator for MO-2032 was not performing as expected. Troubleshooting activities resulted in removing the motor operator and establishing OWA 6. Therefore, operations personnel were no longer able to open and close MO-2032 from the control room as directed by the EOPs. To maintain the ability to control torus water level, operations personnel implemented a compensatory measure which consisted of dispatching an operator to the torus area to locally operate MO-2032 as needed.

During discussions with licensed operators, the inspectors identified that the area surrounding MO-2032 and its associated breaker may be inaccessible during accident conditions due to radiological concerns. As a result, the compensatory measure for OWA 6 might not have been implemented and operators would have been required to initiate an emergency depressurization of the reactor vessel. Because accessability to the area had not been considered when the OWA were established, the inspectors determined that the licensee had not adequately evaluated the ability to perform compensatory actions associated with both OWA under accident conditions.

Emergency operating procedures required operators to secure systems taking suction external to the containment when torus water level was high. The inspectors determined that the inability to implement compensatory actions for MO-2032 was more than minor since this issue could become a more significant safety concern if left uncorrected. Because high torus water level may affect the availability of mitigating systems with suction sources outside containment, the ability to implement compensatory actions for MO-2032 were degraded. The inspectors and the senior reactor analysts evaluated this finding using Appendix A of Inspection Manual Chapter 0609, “Significance Determination Process.”

During the Significance Determination Process Phase 1 review, the inspectors determined that the finding did not constitute a design deficiency or an actual loss of safety function. The inspectors also determined that if the licensee was unable to perform the compensatory measures for this operator workaround, alternative actions described in the EOPs assured that core cooling was maintained. Specifically, the EOPs directed operations personnel to perform an emergency depressurization if torus water level could not be maintained, ensuring that the low pressure mitigation systems could maintain core cooling. Based upon this information, the inspectors determined that this finding was within the licensee response band (Green). These findings were documented in the licensee’s corrective action program as CRs 20002471 and 20001144.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors selected the following post maintenance activities for review. Activities were selected based upon the structure, system, or component's ability to impact risk.

- Observation of selected portions of the 11 CRD Pump bearing replacement work-in-progress and associated post-maintenance testing.

- Observation of the 11 Diesel Fire Pump post-maintenance testing following repair of the pump's associated discharge line auto-vent valve (AV-1937).

The inspectors observed the performance of post-maintenance testing activities which included, but were not limited to, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use and compliance, control of temporary modifications or jumpers required for test performance, documentation of test data, TS applicability, system restoration, and evaluation of test data. The inspectors verified that maintenance and post-maintenance testing activities were adequate and would detect deficiencies prior to returning equipment to service. The post-maintenance testing observed and related documents reviewed included:

- TS 3/4.13, "Fire Detection and Protection Systems"

- Operations Manual B.1.3, "CRD Hydraulic System"

- Operations Manual B.8.5, "Fire Protection"

- Post-Maintenance Testing Vibration Measurement Report for 11 CRD Pump dated 7/25/00

- American Society of Mechanical Engineers/American National Standards Institute OWA-1988, Part 6, "In service Testing of Pumps in Light-Water Reactor Power Plants"

- Preventative Maintenance Procedure 4077PM, Revision 1, "CRD Pump Rebuild"

- WOs:
    - WO 0002772, Revision 1, "Replace 11 CRD Pump Bearings/Rebuild as necessary"
    - WO 0002778, Revision 0, "Repair Diesel Fire Pump Air Vent Valve"

- Equipment Isolations:
    - 00-02772, Version 1, "Replace 11 CRD Pump Bearings/Rebuild as necessary"
    - 00-02778, "Repair Diesel Fire Pump Air Vent Valve"

- P&IDs:
  - P&ID M-118, Revision AU, “Control Rod Hydraulic System”
  - P&ID M-812, Revision RQ, “Screen Wash, Fire & Intake Structure”

- USAR, Revision 17:
  - Section 3.5.3, “Control Rod Drive System”
  - Section 10.3.1, “Fire Protection System”

b. Issues and Findings

There were no findings identified during this inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors selected the following surveillance test activities for review. Activities were selected based upon risk significance and the impact upon risk that an unidentified performance degradation of the structure, system, or component could have if unresolved for long periods of time.

- Surveillance Test Procedure 0060, Revision 21, “RCIC Hi Flow Sensor Test and Calibration Procedure”; and Surveillance Test Procedure 0255-08-IA-1, Revision 44, “RCIC System Tests with Reactor Pressure at Rated Conditions”

- Surveillance Test Procedure 0255-11-III-3, Revision 20, “13 Emergency Service Water Pump and Valve Operability Test”

- Surveillance Test Procedure 0006, Revision 18, ”Scram Discharge Volume Hi Level Scram Test and Calibration Procedure”

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, TS applicability, impact of testing relative to performance indicator reporting, and evaluation of test data. The following documents were reviewed:

- P&IDs:
  - P&ID M-112, “RHR Service Water and Emergency Service Water Systems”
  - P&ID M-811, “Service Water System and Makeup Intake Structure”

- TSs:
  - 3/4.1.A, “Reactor Protection System”
  - Table 3.2.3, “Instrumentation That Initiates a Rod Block”
  - Table 4.2.1, “Minimum Test and Calibration Frequencies for Core Cooling, Rod Block, and Isolation Instrumentation”

- Operations Manual:
  - B.8.1.4, “Emergency Service Water System”
  - B.1.3, “Control Rod Drive Hydraulic System”

- Surveillance Testing:
  - Surveillance Procedure 0006, Revision 18, “Scram Discharge Volume High Level Scram Test and Calibration Procedure”

- Calculations:
  - Calculation CA-97-085, “Scram Discharge Volume Calculation Levels for Technical Specification Limits”
  - Calculation CA-97-093, “Magneto Scram Discharge Volume Set point”
  - Calculation CA-97-094, “FBI Scram Discharge Volume Set point Calculation”

- Safety Review Item 97-001, “Resolution of Scram Discharge Volume High Level Alarm Setting”

- CR 19991998, “Alarm CO5-B-30 Discharge Volume Tank Not Drained Came in During Performance of Test not Identified in Procedure”

- USAR, Section 7.6, “Reactor Protection System”

b. Issues and Findings

There were no findings identified during this inspection.

1EP6 Drill Evaluation

a. Inspection Scope

The resident inspectors reviewed a simulator-based training evolution to evaluate drill conduct and the adequacy of the licensee’s critique of performance to identify weaknesses and deficiencies. The inspectors selected simulator scenarios that the licensee had scheduled as providing input to the Drill/Exercise Performance Indicator. The inspector observed, when applicable, the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critiques. Observations were compared to the licensee’s observations and corrective action program entries. The inspectors verified that there were no discrepancies between observed performance and performance indicator reported statistics. The simulator scenario observed resulted in an unusual event and alert classifications. Documents reviewed included:

- Monticello Simulator Scenario RQ-SS-04E, Revision 6, "SORV With Tailpipe Break"

- Operations Manual A.2 - 101, Revision 25, "Classification of Emergencies"

- Monticello Forms:
  - 5790-102-02, Revision 24, "Monticello Emergency Notification Report"
  - 3195, Revision 19, "Event Notification Worksheet"
  - 3695, Revision 2, "EP [emergency preparedness] Performance Record"

b. Issues and Findings

There were no findings identified during this inspection.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Cornerstone: Barrier Integrity

.1 Reactor Coolant System Identified Leak Rate

a. Inspection Scope

The inspectors verified the accuracy and completeness of the “Reactor Coolant System Identified Leak Rate” performance indicator data submitted by the licensee for January 1 through June 30, 2000. The inspectors reviewed data reported to the NRC since the last verification. The review was accomplished, in part, through evaluation of the TS requirements, plant records, procedural reviews, and reactor coolant sample data. The procedures evaluated and documents reviewed included:

- Monticello Performance Indicator Data Summary Report Q2/2000

- Nuclear Energy Institute (NEI) 99-02, Revision 0, “Regulatory Assessment Performance Indicator Guideline”

- 4AWI-04.08.11, Revision 1, “NRC Performance Indicator Reporting”

- Worksheet 3530-07, Revision 0, "Performance Indicator Drywell Equipment Drain Sump”
  - January 2000 - March 2000
  - April 2000 - June 2000

- Monticello Operations Daily Log - Part J, Revision 76

- Worksheet 3530-12, Revision 0, "NRC Performance Indicator Drywell Equipment Drain Leakage”

b. Issues and Findings

There were no findings identified during this inspection.

4OA3 Event Follow-up

Cornerstones: Mitigating Systems and Barrier Integrity

.1 (Closed) Licensee Event Report (LER) 50-263/2000-010: Missed Technical Specification Required Surveillance Test

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-010, "Technical Specification Surveillance Requirement for Containment Isolation Valve Not Performed." The inspectors reviewed the following references:

- CR 20002445, "Operability Status of MO-2026 & MO-2027 May Not Be Consistent with T.S. Requirements"

- TSs:
  - Section 3.7.D.2 and basis
  - Section 4.7.D.2 and basis
  - Section 5.2.1.2.2 and basis
  - Section 5.2.2.5.3 and basis
  - Section 5.2.3.6.2 and basis
  - Section 5.2.4.3 and basis
  - Basis Section 3.2
  - Definitions for power operation, operable, operating, and primary containment integrity

- USAR:
  - Section 5.2, "Primary Containment"
  - Table 5.2-3b, "Primary Containment Automatic Isolation Valves"

- Generic Letters (GLs):
  - GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance"
  - GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves"

- Monticello Response to GL 89-10 and GL 96-05

b. Issues and Findings

On June 8, 2000, the licensee determined that the reactor head motor-operated spray inboard and outboard automatic containment isolation valves, which were maintained closed during reactor power operations, should have been declared inoperable while the valves were closed. With the valves inoperable, TS 4.7.D.2 required that the position of at least one fully closed valve in each line having an inoperable automatic containment isolation valve be verified and recorded daily. This surveillance requirement had not been met for the two spray valves. The cause of this failure was an erroneous determination by the licensee regarding the applicability of GL 96-05 to these valves.

The licensee’s analysis of the condition indicated that the valves had not been open when primary containment integrity was required and concluded that the event had no effect on the health and safety of the public. Because the valves were maintained closed when required to be operable, primary containment integrity was assured. Therefore, the inspectors concluded that the failure to perform the TS required surveillance constituted a violation of minor significance that was not subject to enforcement action in accordance with Section IV of the Enforcement Policy. The licensee had entered this issue into their corrective action program as CR 20002445.

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 August 15, 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

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

50-263/2000-010	LER	Technical Specification Surveillance Requirement for Containment Isolation Valve Not Performed (4OA3)

Discussed

None

LIST OF ACRONYMS USED

AWI	Administrative Work Instruction
CR	Condition Report
CRD	Control Rod Drive
CW	Circulating Water
DP	Differential Pressure
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
EOP	Emergency Operating Procedure
EP	Emergency Preparedness
ESW	Emergency Service Water System
EWI	Engineering Work Instruction
GO	Generic Letter
HPCI	High Pressure Coolant Injection
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MO	Motor-Operated Valve
NEI	Nuclear Energy Institute
NUMARC	Nuclear Management and Resources Council
OWA	Operator Workaround
P&ID	Piping and Instrument Diagram
PI	Pressure Indicator
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPV	Reactor Pressure Vessel
SBLC	Standby Liquid Control
SORV	Solenoid Operated Relief Valve
TS	Technical Specification
USAR	Updated Safety Analysis Report
V	Volt
VAC	Volt Alternating Current
VDC	Volt Direct Current
WO	Work Order