January 16, 2001

Mr. M. Hammer
Site General Manager
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-263-00-09(DRP)

Dear Mr. Hammer:

On December 31, 2000, the NRC completed a baseline inspection at your Monticello Nuclear Power Plant. The results of this inspection were discussed on January 4, 2001, with you 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. No findings were identified in any of the cornerstones of safety during our inspection.

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 System (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/

Bruce L. Burgess, Chief
Reactor Projects Branch 2

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

Enclosure: Inspection Report 50-263-00-09(DRP)

See Attached Distribution

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263-00-09(DRP)
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Power Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	November 15 through December 31, 2000
Inspectors:	Stephen Burton, Senior Resident Inspector Daniel Kimble, Resident Inspector Karla Stoedter, Regional Inspector Paul Pelke, Regional Inspector Gary Pirtle, Regional Inspector
Approved by:	Bruce L. Burgess, Chief Reactor Projects Branch 2 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

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

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. 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 50-263-00-09(DRP)

IR 05000263-00-09(DRP), on 11/15-12/31/2000; Nuclear Management Company, LLC; Monticello Nuclear Power Plant; Resident Operations Report.

The inspection was conducted by resident and regional projects inspectors. The report covers a 6½-week period of resident inspection.

No findings were identified in any of the cornerstones of reactor safety.

Report Details

Summary of Plant Status: The unit began the inspection period operating at full power. On December 9, 2000, at approximately 0900 the unit began a planned power reduction to 75 percent, which was achieved at approximately 1235 that day. Following the completion of scheduled maintenance and surveillance activities, which included main steam isolation valve (MSIV) exercising and a rod pattern adjustment, power escalation was commenced at approximately 1609 on December 9, 2000, with full power being reached at approximately 0610 on December 10, 2000. The unit remained at full power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. Cold weather protection, such as heat tracing, was verified to be in operation where applicable. Documents reviewed included:

- Updated Safety Analysis Report (USAR), Revision 18:
  - Section 5.3.4, "Reactor Building Heating and Ventilating Systems"
  - Section 10.3.2, "Plant Heating, Ventilating, and Air Conditioning Systems"
- Administrative Procedure 1151, Revision 39, "Winter Checklist"
- Safety Review Item (SRI) 97-010, Revision 0, "Revision of USAR Sections 5.3.4 and 10.3.2.2.5: Clarification of the Reactor and Radwaste Building Air Supply Description and the Reactor Building Ventilation Stack Description"

b. Issues and Findings

There were no findings identified during this inspection.

1R04 Equipment Alignment

a. Inspection Scope

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

- Division I residual heat removal service water (RHRSW) system while the Division II RHRSW system was out-of-service for planned maintenance

- Division II core spray system while the Division I core spray system was out-of-service for planned 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:

- Design Basis Document, Section B.8.1.3, Revision 2, “Design Basis Document for RHR Service Water"

- Operations Manual:
  - Section B.8.1.3, “RHR Service Water System"
  - Section B.3.1, "Core Spray Cooling System"

- Technical Specifications, Section 3/4.5, “Core and Containment Spray/Cooling Systems," and Basis

- USAR, Revision 18:
  - Section 10.4.2, “Residual Heat Removal Service Water System"
  - Section 6.2.2, "Core Spray System"

- Piping and Instrument Diagrams (P&IDs):
  - M-811, Revision CB, "Service Water System and Make-Up Intake Structure"
  - M-112, Revision BF, "RHR Service Water & Emergency Service Water Systems"
  - M-122, Revision AH, "Core Spray System"

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 (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:

- Automatic Depressurization System

- Reactor Manual Control

- Plant Protection System/ATWS (Anticipated Transient without SCRAM)

The inspectors verified the licensee's categorization of specific issues, including evaluation of the performance criteria. 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 05.02.01, Revision 3, "Monticello Maintenance Rule Program Document"

- Monticello Maintenance Rule Periodic Assessment Report, 1st Quarter - 2000

- Operations Manual:
  - Section B.3.3, "Reactor Pressure Relief"
  - Section B.5.6, “Plant Protection System”

- Technical Specifications:
  - Section 3/4.2, "Protective Instrumentation," and Basis
  - Section 3/4.5, "Core and Containment Cooling Systems," and Basis
  - Section 3/4.6, "Primary System Boundary," and Basis

- Maintenance Rule Program System Basis Document:
  - Section B.3.3, "Reactor Pressure Relief"
  - Section B.5.6, Revision 1, "ATWS - Alternate Rod Insertion & Recirc Pump Trip 9ARI/RPT)"
  - Section B.5.5, Revision 2, “Reactor Manual Control System”

- P&IDs:
  - NX-16162-1; Revisions B (sheet 1), C (sheets 6-8 & 10-11), and D (sheets 3-5 & 9); "ATWS System Elementary Diagram"

- USAR, Revision 18:
  - Section 4.4, "Reactor Pressure Relief"
  - Section 6.2, "Automatic Depressurization System"
  - Section 7.6.1, "Reactor Protection System"
  - Section 7.6.2, "ATWS System"

- Surveillance Procedures:
  - 0037/0038, "APRS [Automatic Pressure Relief System] Low Pressure Core Cooling Pumps Discharge Pressure Interlock Instruments Test," dated November 11, 1999, February 24, 2000, and May 11, 2000.

- Condition Reports:
  - 19990158, "Safety Relief Valve (SRV) Bellows Weld and Second Stage Seat Indications Found During Refurbishment Activities"
  - 19990808, "G SRV Declared Inoperable Due to Questionable Integrity of Bellows Leak Detection System"
  - 19991142, "Generic Letter 96-06 Operability Evaluation of SRV Operator Solenoid Valves"
  - 19991280, "Received G SRV Bellows Leaking Alarm"
  - 19991300, "Reactor Scram 108, Manual Scram Inserted After Isolation of Air Ejectors on High Pressure"
  - 19991633, "A Small Leak was Found in the G SRV Bellows"
  - 19993792, "As Left Leakage Criterion not met for SRV Bonnet Leak Check"
  - 20000593, "During Calibration Found PS [Pressure Switch] - 7354 SRV C Bellows Leak Detection Out of As-Found"
  - 20000669, "As-Built Configuration for SRV Discharge Line Support Does Not Match Drawing"
  - 20001196, "Inadequate Support of Conduit for SRV Tailpipe Thermocouple"
  - 20003675, "SRV Inlet Flange Studs are 6-thread Series, ANSI B16.5 Requires that they be 8-thread Studs"
  - 20004649, “Definition of What Constitutes a Maintenance Rule Functional Failure Needs to be Clarified and Applied to Past Events for RMC [Reactor Manual Control] System”
  - 20004822, “10CFR50.59 Screening Incorrectly Determined That a 10CFR50.59 Evaluation was not Required for a Jumper/Bypass”

- Work Orders:
  - 9803122, "Safety Relief Valve Bellows Monitor Light/Alarm Cleared Early"
  - 9904980, "A SRV Bellows Solenoid Valve Leaking"
  - 9905224, "Troubleshoot G Bellows Leak Detection System"
  - 9905565, "SRV Bellows Leak Detection System Leaks"
  - 0000784, "Repair SRV Bellows Leak Detection Leaks"

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 observed the following risk significant systems undergoing scheduled or emergent maintenance:

- "B" Train EFT [Emergency Filtration Train] Relay Replacement

- 14 RHRSW Pump Discharge Check Valve

- 12 RHRSW Pump Discharge Check Valve

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:

- Design Basis Document, Section B.8.1.3, Revision 2, "Design Basis Document for RHR Service Water"

- Operations Manual:
  - Section B.8.13, "Control Room Heating and Ventilation and Emergency Filtration Train"
  - Section B.8.1.3, "RHR Service Water System"

- USAR, Revision 18:
  - Section 6.7, "Main Control Room, Emergency Filtration Train Building, and Technical Support Center Habitability"
  - Section 10.4.2, "Residual Heat Removal Service Water System"

- Technical Specifications:
  - Section 3/4.2.I, "Instrumentation for Control Room Habitability," and Basis
  - Section 3/4.17, "Control Room Habitability," and Basis
  - Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis
  - Section 3/4.13.H.2, “Fire Detection and Protection Systems - Alternate Shutdown System”

- Work Orders:
  - 0003971, "Replace "B" EFT Relays”
  - 0005110, “Investigate/Repair Leaking Check Valve”
  - 0004989, “14 RHRSW Pump Check Valve”

- Procedures and Forms:
  - 3630, “Alteration Package - RHRSW Pump Discharge Check Valve Sleeve Installation,” Number 00A059
  - 3630, “Alteration Package - RHRSW Check Valve Disk Fit Up,” Number 98A044
  - 4001-11-01, “Swing Check Valve Inspection,” RHRSW-1-4 for WO 0004989
  - 3590, Revision 1, “Service Water Component Inspection”
  - 0255-05-IA-1, Revision 39, “RHR Service Water Pump and Valve Tests”
  - MMP-011, Revision 1, “Check Valve Disassembly/Inspection”

- P&IDs:
  - M-811, Revision CB, "Service Water System and Make-Up Intake Structure"
  - M-112, Revision BF, "RHR Service Water & Emergency Service Water Systems"

- Calculation CA-98-180, “Temporary Lifting of RHRSW Valves”

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 the following operability evaluations to determine the impact on Technical Specifications, and the significance of the evaluations, and to ensure that adequate justifications were documented.

- RHRSW pump discharge check valves

- Lube Oil Discrepancy Found on Diesel Fire Pump

Operability evaluations were selected based upon the relationship of the safety-related system, structure, or component to risk. The documents reviewed included:

- Condition Reports:
  - 20004709, “ ‘B’ RHRSW Loop Pressure Decreased to Zero on Shutdown of 12 RHRSW Pump Making Loop Inoperable and Unplanned LCO Entry”
  - 20005106, “Lube Oil Addition Discrepancy Between Procedure and Equipt. Oil Identification Name Tag”

- P&IDs:
  - M-811, Revision CB, "Service Water System and Make-Up Intake Structure"
  - M-112, Revision BF, "RHR Service Water & Emergency Service Water Systems"

- Design Basis Document, Section B.8.1.3, Revision 2, "Design Basis Document for RHR Service Water"

- Alteration #94A030, "Diesel Fire Pump Engine Oil"

- Operations Manual
  - Section B.8.1.3, "RHR Service Water System"
  - Section B.8.5, "Fire Protection"

- Technical Specifications, Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis

- USAR, Revision 18, Section 10.4.2, "Residual Heat Removal Service Water System"

- NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants"

- NRC Information Notice 91-50, Supplement 1, "Water Hammer Events Since 1991"

b. Issues and Findings

There were no findings identified during this inspection.

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.

- 13 Service Water Pump Re-packing

- Post-maintenance testing to evaluate restoration of the service water to RHRSW keep fill system after unexpected failure and RHRSW depressurization.

- Division II Core Spray Injection Valve Maintenance

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 documents reviewed included:

- Design Basis Document, Section B.8.1.3, Revision 2, "Design Basis Document for RHR Service Water"

- NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants"

- NRC Information Notice 91-50, Supplement 1, "Water Hammer Events Since 1991"

- Operations Manual, Section B.8.1.3, "RHR Service Water System"

- USAR, Revision 18, Section 10.4.2, "Residual Heat Removal Service Water System"

- Technical Specifications:
  - Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis
  - Section 3/4.5.C, "Containment Spray/Cooling System," and Basis
  - Section 3/4.13.H, "Alternate Shutdown System," and Basis

- Work Orders:
  - 0004209, "Re-pack 13 Service Water Pump"
  - 0004968, "Verify Operation of SW-21-2 and SW-22-2"
  - 0001507, “12 Core Spray Injection Outboard”
  - 0001509, “12 Core Spray Injection Inboard”

- P&IDs:
  - M-813, Revision R, "Miscellaneous Piping: Circulating Water System"
  - M-811, Revision CB, "Service Water System and Make-Up Intake Structure"
  - M-112, Revision BF, "RHR Service Water & Emergency Service Water Systems"

- Condition Reports:
  - 20004731, " ‘B’ RHRSW Keep Fill System Unable to Maintain Standby Pressure Above Setpoint of Alarm CO3-B-19"
  - 20004674, “ ‘B’ RHRSW to RHR HX Differential Pressure Unexpectedly Decreased During Restoration of SW Auto Strainer. Unexpected Alarm CO3-B-19"
  - 20004709, “ ‘B’ RHRSW Loop Pressure Decreased to Zero on Shutdown of 12 RHRSW Pump Making Loop Inoperable and Unplanned LCO Entry”

- Procedures and Forms:
  - 3069, Revision 8, "Post-Maintenance Testing Activities Control Cover Sheet"
  - 4001-13, Revision 2, "Pump Packing Checklist"
  - 4900-01-PM, Revision 15, “PM For Limitorque motor Operated Valves”

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

There were no findings identified during this inspection.

1R22 <u>Surveillance Testing</u>

a. <u>Inspection Scope</u>

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 a structure, system, or component could have if left unresolved for long periods of time.

- ATWS Trip Unit Test and Calibration
- Safeguards Bus Voltage Protection Tests and Calibrations
- Torus Level Instrumentation Semi-Annual Calibrations
- Core Spray System Instrumentation Calibrations

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:

- Drawings:
  - NX-16162-1; Revisions B (sheet 1), C (sheets 6-8 & 10-11), and D (sheets 3-5 & 9); "ATWS System Elementary Diagram"
  - NF-36397, Revision Y, "Meter and Relay Diagram, 4160V Buses"
  - NE-36399-9, Revision N, "Essential Bus Transfer Circuits - Division I"
  - NE-36399-9B, Revision A, "Essential Bus Transfer Circuits - Division II"
  - NX-7833-21; Revisions AC (sheet 1), Q (sheet 2), H (sheets 3 and 5), E (sheet 4), and K (sheet 4A); "Core Spray System Elementary Diagram"
  - M-122, Revision AH, "Core Spray System P&ID"
  - M-114-1, Revision X, “Service Condensate System - Radwaste Building"

- Surveillance Test Procedures:
  - 0278-A, Revision 8, "ATWS-Recirc Trips for Reactor Pressure and Level Trip Unit Test and Calibration"
  - 0278-B, Revision 9, "ATWS-Recirc Trip for Reactor Pressure and Level Trip Unit Test and Calibration"
  - 0301, Revision 26, "Safeguard Bus Voltage Protection Relay Unit Functional Test, Degraded Voltage Protection"
  - 0302, Revision 13, "Safeguard Bus Degraded Voltage Protection - Relay Unit Calibration"
  - 0303, Revision 26, "Safeguard Bus Voltage Protection Relay Unit Functional Test, Loss of Voltage Protection"

- 7120, Revision 7, "Core Spray System Instrument Maintenance"

- Calculations:
  - CA-95-023, Revision 0, "ATWS Low Low Level Setpoint"
  - CA-95-019, Revision 0, "Determination of ATWS High Reactor Pressure Instrument Setpoints"
  - CA-92-220, Revision 0, "Degraded Voltage Setpoint Calculation"

- Instrument Calibration Worksheets (Card 64s):
  - FT-14-40A, "11 Containment Spray Pump Flow"
  - FI-7189, "Division I RHR/RHRSW/Containment Spray Systems Test Instrument"
  - PT-14-38A, "11 Containment Spray Pump Discharge Pressure"
  - PI-14-48A, "Division I Containment Spray Pump Pressure"
  - FT-14-40B, "12 Containment Spray Pump Flow"
  - FY-4104, "Containment Spray Pump Loop B Flow Isolation"
  - FI-7188, "Division II RHR/RHRSW/Containment Spray Systems Test Instrument"
  - PT-14-38B, "12 Containment Spray Pump Discharge Pressure"
  - PI-14-48B, "Division II Containment Spray Pump Pressure"

- Operations Manual:
  - Section B.5.6, "Plant Protection System"
  - Section B.3.1, "Core Spray Cooling System"

- Technical Specifications:
  - Section 3/4.2, "Protective Instrumentation," and Basis
  - Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis

- USAR, Revision 18:
  - Section 7.6.1, "Reactor Protection System"
  - Section 7.6.2, "ATWS System"
  - Section 6.2.2, "Core Spray System"

- NRC Inspection Manual, Part 9900, "Technical Guidance On Operable/Operability"

- NRC Generic Letter 91-18, Revision 1, "Resolution Of Degraded And Non-conforming Conditions"

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

There were no findings identified during this inspection.

1R23 Temporary Plant Modification

a. Inspection Scope

The inspectors reviewed Jumper Bypass 99-029, "Install Blanking Plate Downstream of X-10-4 (Recirculation Pump Seal Vent to ORW [Open Radwaste])." The inspectors reviewed the safety screening, design documents, USAR, and applicable TS to determine that the temporary modification was consistent with modification documents, drawings and procedures. The inspectors also reviewed the actual impact of the temporary modification on the permanent and interfacing systems. Other documents reviewed included:

- Administrative Work Instructions (AWIs):
  - 4AWI-04.04.03, Revision 13, "Bypass Control"
  - 4AWI-05.06.02, Revision 3, "10 CFR 50.59 Applicability Screening"

- Safety Screening for Jumper Bypass 99-029

- USAR, Revision 18, Chapter 15, "USAR Drawings"

- Daily Plant and Equipment Status dated December 4, 2000

- P&ID M-117, Sheet 2, Revision J, "Recirc Loops, Pumps, and Motors"

- ANSI/ASME B31.1, "Power Piping," 1983 Edition

- ASME Code Section XI, "Inservice Inspection," Program Document

- TS 3/4.1.5, "Inservice Inspection and Testing"

- Condition Reports:
  - 20001751, "Results of Jumper Bypass Audit"
  - 20002614, "Jumper bypass calculations were revised but not approved prior to bypass installation"
  - 20003140, "Testing complete block of jumper bypass form not completed when bypass device was installed"
  - 20004822, "10CFR50.59 screening incorrectly determined that a 10CFR50.59 evaluation was not required for a jumper/bypass"

- Work Orders:
  - 9601382, "Valve Leaks by Seat"
  - 9601393, "Install Pipe Cap Downstream from XR-10-3"
  - 9602908, "Replace XR-10-3"
  - 9905650, "Flange Upstream of FI [Flow Instrument] -4200B Leaking"
  - 9908298, "Replace XR-10-4 and XR-10-2"
  - 0000558, "Install New Packing in XR-10-4 and XR-10-2"
  - 0000657, "Verify Flange Studs are Sufficiently Torqued"
  - 0000672, "Replace Gasket Downstream of XR-10-4"
  - 0000976, "Repair Valves XR-10-4 and XR-10-2"

b. Issues and Findings

There were no issues identified during this inspection.

3. SAFEGUARDS

Cornerstone: Physical Protection

PP4. Security Plan Changes (IP 71130.04)

a. Inspection Scope

The inspector reviewed Revision 50 of the Monticello Security Plan and Revision 10 of the Monticello Safeguards Contingency Plan, which were submitted by licensee letter, dated August 30, 2000, to verify that the changes did not decrease the effectiveness of the security plans. The security plans were submitted in accordance with 10 CFR 50.54(p).

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES

4OA3 Event Follow-up

Cornerstone: Mitigating Systems, Barrier Integrity, Emergency Preparedness

.1 (Closed) LER 50-263/2000-014: Missed Standby Liquid Control (SLC) System Surveillance Test

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-014, "Missed Standby Liquid Control System Surveillance Test." The inspectors reviewed the following references:

- Condition Reports:
  - 20003610, "SLC Test not Recycled Monthly - Not Consistent with TS Requirements"
  - 19981545, "The Standby Liquid Control Pumps are not Tested as Described in the USAR"
- Technical Specifications, Section 3/4.4, "Standby Liquid Control System," and Basis
- USAR, Revision 18. Section 6.6.4, "SLC Inspection and Testing"

b. Issues and Findings

On September 20, 2000, the licensee determined that the SLC surveillance procedure did not include the steps necessary to recycle demineralized water to the test tank monthly as required by TS 4.4.A.1. The inspectors reviewed the licensee's assessment and concluded that this issue had minimal impact on safety and that the licensee had taken adequate corrective actions. Subsequently, the inspectors determined 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 20003610.

.2 (Closed) LER 50-263/2000-009: "Procedural Inadequacy Results in Mismatch of Feedwater Flow Instruments and Process Computer Calibration Causing Operation above Licensed Power."

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-009, "Procedural Inadequacy Results in Mismatch of Feedwater Flow Instruments and Process Computer Calibration Causing Operation Above Licensed Power." The inspectors reviewed the following references:

- Condition Report 20001464, "Mismatch of Feedwater Flow Instruments and Process Computer Calibrations Result in High Core Thermal Power"
- Memorandum dated August 22, 1980, from E. L. Jordan, NRC, "Discussion of Licensed Power Level"
- Monticello Nuclear Generating Plant, Operating License Number DPR-22, Section 2.C.1, "Maximum Power Level"

b. Issues and Findings

On April 4, 2000, the licensee identified a small mismatch between the feedwater flow transmitter calibration values and the corresponding values provided by the plant process computer. The licensee determined that the mismatch occurred in 1998 while implementing an increase in the maximum licensed thermal power level. Specifically, the span of the feedwater flow transmitters was changed to allow for the increased maximum licensed thermal power. However, the process computer feedwater flow calibration constants were not changed. Due to this error, the licensee operated the reactor up to 100.1 percent of the maximum licensed thermal power of 1775 megawatts thermal for 154 days and up to 100.2 percent for 162 days.

Section 2.C.1 of Monticello Nuclear Generating Plant Operating License Number DPR-22 authorizes the Nuclear Management Company to operate the Monticello facility at steady state reactor core power levels not in excess of 1775 megawatts thermal. Because no thermal limits were exceeded during the increased power operation and because the licensee maintains the nuclear instrument trip setpoints, at a minimum, 1 percent conservative, TS limits were bounded during plant operation with a maximum error of 0.2 percent. And, when added to the uncertainty associated with the General Electric online heat balance calorimetric used by the licensee to measure actual plant power, the maximum 0.2 percent excess described in this LER still remains below the 102 percent maximum power value assumed in USAR safety analyses. Based upon these observations, the inspectors concluded that the operation of the Monticello facility at steady state reactor core power levels in excess of 1775 megawatts thermal for approximately 316 days constituted a violation of minor significance that was not subject to formal enforcement action in accordance with Section IV of the Enforcement Policy. The licensee had entered this issue into their corrective action program as CR 20001464.

.3 (Closed) LER 50-263/2000-011: "Service Water Radiation Monitor Alarm Setpoint Non-conservative With Circulating Water Pumps Shutdown."

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-011, "Service Water Radiation Monitor Alarm Setpoint Non-conservative With Circulating Water Pumps Shutdown." The inspectors reviewed the following references:

- CR 20001070, "Service Water Monitor Alarm Setpoint Does Not Meet Tech Spec 3.8.A.1.d When No Circulating Water Pumps Are In Operation," and associated action requests

- Technical Specifications, Section 3.8.A.1.d, "Radioactive Effluents - Liquid Effluents" and Basis

- Monticello Offsite Dose Calculation Manual (ODCM)

b. Issues and Findings

During a review of the Monticello ODCM, the licensee discovered that the setpoint for the service water discharge radiation monitor was set non-conservative during certain modes of plant operation. The inspectors reviewed the licensee's assessment that the event had no effect on the health and safety of the public. Subsequently, the inspectors determined that this failure to meet the requirements of TS 3.8.A.1.d was of very low safety significance and 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 20001070.

4OA6 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Hammer and other members of licensee management on January 4, 2001. 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 General 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

50-263/2000-014	LER	Missed Standby Liquid Control (SLC) System Surveillance Test (4OA3)
50-263/2000-009	LER	Procedural Inadequacy Results in Mismatch of Feedwater Flow Instruments and Process Computer Calibration Causing Operation above Licensed Power (4OA3)
50-263/2000-011	LER	Service Water Radiation Monitor Alarm Setpoint Non-conservative With Circulating Water Pumps Shutdown (4AO3)

Discussed

None

LIST OF ACRONYMS USED

ATWS	Anticipated Transient Without SCRAM
AWI	Administrative Work Instruction
DRP	Division of Reactor Projects
EFT	Emergency Filtration Train
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MSIV	Main Steam Isolation Valve
NUMARC	Nuclear Management and Resources Council
P&ID	Piping and Instrument Diagram
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SLC	Standby Liquid Control
SRV	Safety Relief Valve
USAR	Updated Safety Analysis Report