November 30, 2000

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 50-263-00-08(DRP)

Dear Mr. Hammer:

On November 14, 2000, the NRC completed a baseline inspection at your Monticello Nuclear Power Plant. The results of this inspection were discussed on November 15, 2000, 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 50-263-00-08(DRP)

See Attached Distribution

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R. Nelson, President
  Minnesota Pollution Control Agency
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer
  Wright County Government Center
Commissioner, Minnesota Department of Commerce
A. Neblett, Assistant Attorney General

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263-00-08(DRP)
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Power Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	October 1 through November 14, 2000
Inspectors:	Stephen Burton, Senior Resident Inspector Daniel Kimble, Resident Inspector Michael Bielby, 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 50-263-00-08(DRP)

IR 05000263-00-08, on 10/01-11/14/2000; Nuclear Management Company, LLC; Monticello Nuclear Power Plant; Resident Operations Report.

The inspection was conducted by resident inspectors 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 operated at 100 percent power for the duration of the 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 High Pressure Coolant Injection System and associated 250 Vdc and 125 Vdc battery systems to verify operability and proper equipment lineup while the Reactor Core Isolation Cooling system was disabled due to planned maintenance. The system was selected due to the relative increase in core damage frequency caused by rendering one train of high pressure injection 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:
  - Section B.2.3, "RCIC System"
  - Section B.3.2, "HPCI System"
  - Section B.9.9, "250 Vdc System"
  - Section B.9.10, "125 Vdc System"

- Equipment Isolations:
  - 00-01545, Version 1, "MO-2078 Preventative Maintenance"
  - 00-01555, Version 1, "MO-3502 Preventative Maintenance"
  - 00-03576, Version 1, "Packing Leak on RCIC Valve CV-2079"

- Piping and Instrument Diagrams (P&IDs):
  - M-124, Revision Y, "HPCI System [Water Side]"
  - M-123, Revision AF, "HPCI System [Steam Side]"
  - M-123-1, Revision B, "HPCI Hydraulic Control & Lubrication System"
  - M-125, Revision AK, "RCIC System [Steam Side]"
  - M-126, Revision Y, "RCIC System [Water Side]"

- Technical Specifications (TSs):
  - Section 3/4.9, "Auxiliary Electrical Systems," and Basis
  - Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis

- Updated Safety Analysis Report (USAR), Revision 18
  - Section 6.2.4, "HPCI System"
  - Section 8.5, "DC [Direct Current] Power Supply Systems"
  - Section 10.2.5, "RCIC System"

- Work Orders (WOs):
  - 0001545, "MO-2078 Preventative Maintenance & Post-Maintenance Testing"
  - 0003442, "Repair Minor Oil Leaks on RCIC Turbine"

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 1-D, (Reactor Building 896' Tank Room)

- Fire Zone 2-B, (East CRD (Control Rod Drive) HCU (Hydraulic Control Unit Area)

- Fire Zone 2-C, (West CRD HCU Area)

- Fire Zone 3-C, (Reactor Vessel Instrument Rack Area [962' Elevation])

- Fire Zone 4-A, (Reactor Building [985' Elevation South])

- Fire Zone 4-B, (RBCCW [Reactor Building Closed Cooling Water] Heat Exchanger Area)

- Fire Zone 13-B, (Reactor Feed Pump and Lube Oil Reservoir Room)

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:

- Technical Manual NX-16991, "Monticello Updated Fire Hazards Analysis"

- Monticello Fire Strategies:
  - A.3-02-B, Revision 4, "East HCU Area"
  - A.3-02-C, Revision 3, "West HCU Area"
  - A.3-04-B, Revision 2, "RBCCW Heat Exchanger Area"
  - A.3-01-D, Revision 3, "Reactor Building Elevation 896' Tank Room"

- A.3-03-C, Revision 3, "Reactor Vessel Instrument Rack Area"
- A.3-04-A, Revision 3, "Reactor Building 985' Elevation South"
- A.3-13-B, Revision 5, "Reactor Feed Pump and Lube Oil Reservoir Room"

- Procedures and Administrative Work Instructions (AWIs):
  - 4AWI-08.01.01, Revision 14, "Fire Prevention Practices"
  - 4AWI-08.01.02, Revision 4, "Combustion Source Use Permit"
  - 0271, Revision 24, "Fire Hose Station and Yard Hydrant Hose House Equipment Inspection"
  - 0275-2, Revision 14, "Fire Barrier Wall, Damper, and Floor Inspection"
  - 0274, Revision 16, "Fire Hose Hydrostatic Test Interior Hose Stations"
  - 0275-1, Revision 8, "Fire Barrier Penetration Seal Visual Inspection"
  - 0275-2, Revision 15, "Fire Barrier Wall, Damper, and Floor Inspection"

- Drawings:
  - FHA-18, Revision 1, "Fire Hazards Analysis Section View A-A"
  - FHA-19, Revision 0, "Fire Hazards Analysis Section View B-B"
  - NX-16991-47, Revision A, "Fire Penetration Seal Locations"

- IPEEE [Individual Plant Evaluation External Events], NSPLMI-95001, Revision 1, Appendix B, "Internal Fires"

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

There were no findings identified during this inspection.

1R11 <u>Licensed Operator Requalification Program</u>

a. <u>Inspection Scope</u>

The inspectors observed the performance of a training crew during a simulator exam scenario and evaluated licensed operator performance in responding to accident sequences. The scenario included a failure to scram scenario complicated by the unavailability of HPCI and RCIC resulting in a reactor depressurization. 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 included:

- Monticello Simulator Scenario RQ-SS-14E, Revision 8, "ATWS [Anticipated Transient Without Scram] With Loss of High Pressure Injection"

- Plant Procedures:
  - Operator Work Instruction, OWI-01.03, Revision 5, “Operations Communication Standard”
  - A.2-101, Revision 26, “Classification of Emergencies”
  - 4AWI-04.08.02, Revision 6, “10 CFR 50.72 and 10 CFR 73.71 Immediate Notifications”
  - 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 17, "Reactor Scram"

- Emergency Operating Procedures:
  - C.5-1100, Revision 6, "RPV [Reactor Pressure Vessel] Control"
  - C.5-1200, Revision 8, "Primary Containment Control"
  - C.5-3205, Revision 0, "Terminate and Prevent"
  - C.5-2007, Revision 9, “Failure To Scram”

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:

- Control Rod Drive System

- Control Rod Drive Hydraulic System

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.1.2, "Control Rod Drive System"
  - Section B.1.3, "Control Rod Drive Hydraulic System"
  - Section B.5.05, "Reactor Manual Control System"

- TS, Section 3/4.3, "Control Rod System," and Basis

- Monticello Maintenance Rule Program System Basis Document:
  - Section B.1.2/3, Revision 0, "Control Rod Drive System"
  - Section B.1.2/3, Revision 2, "Control Rod Drive Hydraulic System"
  - Section B.5.5, Revision 2, "Reactor Manual Control System"

- P&IDs:
  - M-108, Revision BK, "Condensate and Demineralized Water Storage System"
  - M-118, Revision AU, "Control Rod Hydraulic System"
  - M-119, Revision S, "Control Rod Hydraulic System"

- Elementary Wiring Diagram, NX-7866-74, "Reactor Manual Control System"

- USAR, Revision 18:
  - Section 7.2.1, "Reactor Manual Control System"
  - Section 3.5, "Reactivity Control Mechanical Characteristics"

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:

- Selected portions of RCIC pump and valve maintenance work-in-progress

- Selected portions of the Condensate Storage Tank (CST) 13-2 penetration Furmanite repair

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:

- Operations Manual, Section B.2.3, "RCIC System"

- USAR, Revision 18:
  - Section 10.2.5, "RCIC System"
  - Section 6.2, "Emergency Core Cooling System"

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

- Equipment Isolations:
  - 00-01545, Version 1, "MO-2078 Preventative Maintenance"
  - 00-01555, Version 1, "MO-3502 Preventative Maintenance"
  - 00-03576, Version 1, "Packing Leak on RCIC Valve CV-2079"

- P&IDs:
  - M-125, Revision AK, "RCIC System [Steam Side]"
  - M-126, Revision Y, "RCIC System [Water Side]"
  - M-108, Revision BK, "Condensate & Demineralized Water Storage System"

- WOs:
  - 0001545, "MO-2078 Preventative Maintenance & Post-Maintenance Testing"
  - 0003442, "Repair Minor Oil Leaks on RCIC Turbine"
  - 0003699, "Installation of CST-13-2 Flange Seal"

- Maintenance and Administrative Procedures and Forms:
  - 4916-12PM, "Lubrication - CRD, HPCI, and RCIC Rooms and Torus Area"
  - 8190, Revision 5, "Leak Sealing"
  - 4AWI-01.03.03, Revision 3, "Color Coded P&ID Q-List Extension"
  - 3034, Revision 18, "Jumper Bypass Form"
  - 3278, Revision 1, "10 CFR 50.59 Applicability Screening"

- Furmanite Engineering Procedure N-2000224, Revision 0, Project ML-12878, "Monticello Enclosure Installation"

b. Issues and Findings

There were no findings identified during this inspection.

1R15 Operability Evaluations

.1 Residual Heat Removal Service Water (RHRSW) Header Cross-Tie

a. Inspection Scope

The inspectors reviewed the technical adequacy of an operability evaluation associated with an open RHRSW Header Cross-Tie valve, RHRSW-32. The inspectors reviewed applicable plant information to ascertain compliance with the TSs and to ensure that adequate justification was provided in the evaluation for leaving the valve in the open position. The operability evaluation was selected based upon the relationship of the safety-related system, structure, or component to risk. The documents reviewed included:

- Condition Reports (CRs):
  - 20002516, "Both Loops of RHRSW Declared Inoperable Due to Low Standby System Pressure"
  - 19992809, "Check Valve RHRSW-1-1 Fails to Close During Surveillance Test 0255-05-IA-1"
  - 20002448, "'A' RHRSW Loop Inoperability not Immediately Considered and CR not Initiated in Response to Unexpected LCO [Limiting Conditions for Operation] Entry on 9/17/99"
  - 20003502, "While Performing Isolation per WO 0001607, Temporarily Lost Differential Pressure Across Both RHR [Residual Heat Removal] Heat Exchangers"
  - 20003535, "Potential Single Failure Vulnerability of RHRSW System in Standby with RHRSW-32 Cross-Tie Valve Open"

- 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”

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

- TS, 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"

- Monticello Design Change 82M068, "RHRSW Cross-Tie"

b. Issues and Findings

During review of licensee Condition Report 20002448, inspectors identified a potential vulnerability associated with RHRSW system operation with RHRSW-32 open. Specifically, the inspectors identified two occasions where operation with RHRSW-32 open had resulted in system depressurization. The inspectors were concerned that a failure in one system while operating with RHRSW-32 open could permit the depressurization, draining, and voiding of both trains of RHRSW. The inspectors also postulated that both RHRSW systems could be rendered inoperable in the event that RHRSW pumps were restarted with an existing void.

The inspectors also identified that the two conditions which had resulted in system depressurization were attributed to potentially deficient design or plant operating conditions. One of the depressurization events occurred after RHRSW pump maintenance had been preformed and the pump returned to service. When the RHRSW pump was un-isolated, the discharge check valve was found slightly open and the system depressurized via back-flow through the valve and pump suction. The licensee identified that the discharge check valves may be located too close to the pump discharge and that this condition may have been the cause of the depressurization. Specifically, the check valves are located within 10 pipe diameters from the pump discharge, and good engineering practice would locate these valves greater than 10 diameters downstream. The licensee referenced EPRI-NMAC (Electric Power Research Institute - Nuclear Maintenance Applications Center) document NP-5479, Revision 1, “NMAC Application Guide for Check Valves in Nuclear Power Plants,” which indicated that check valves located from 1 to 5 pipe diameters from flow disturbances would not be held firmly against the stop.

The second occurrence happened when two check valves preventing back-flow through the RHRSW keep fill system failed to close. System keep fill is normally provided to RHRSW by normal service water via the check valves that failed to seat. Upon inspection the licensee discovered that the failure of the two in-line valves to close was caused by silt deposits. Inspector interviews indicated that valve inspection frequency coupled with environmental conditions and valve design may be the cause of the silt related failures. Because the purpose of the service water to RHRSW interconnection is to minimize the impact of water-hammer on the RHRSW system, inspectors believed this function to be potentially compromised. These conditions further validated the inspectors belief that the RHRSW system was susceptible to a draining and water-hammer event.

Upon further review, the inspectors’ identified that the licensee had installed the RHRSW header cross-tie as Design Change 82M068 in the 1980s. The purpose of the modification was to permit both RHRSW headers to be pressurized by a single RHRSW pump in the event that only one pump was available due to vital ac power limitations. This would permit proper differential pressure to be maintained across RHR/RHRSW heat exchangers in both trains, thus ensuring that any leakage across the heat exchanger tubes would be maintained in the conservative direction, from RHRSW into the RHR system.

During the review of the design change the inspectors identified that the safety evaluation for the cross-tie modification package permitted normal operation with RHRSW-32 in the closed position. The licensee had been operating with RHRSW-32 in the open position for an indeterminate amount of time. Operation in the open configuration was contrary to the requirements of the safety evaluation and design change package. Subsequently, the licensee was unable to produce a safety evaluation that allowed the procedure to be modified to permit operation with RHRSW-32 open. The licensee closed RHRSW-32 and restored operation to conform to the requirements of the design change. The licensee was at the time controlling the position and operation of the valve under several temporary procedure changes. The licensee issued a Condition Report 20003535 to document this issue and to facilitate evaluation.

Because a water-hammer event had the potential to render both trains of RHRSW inoperable the inspectors evaluated the issue as a potential candidate for the significance determination process. The inspectors concluded that the postulated event had a credible impact on safety because the RHRSW system is risk significant, the system was included in the licensee’s maintenance rule scoping, and the licensee’s probabilistic risk model indicated an increase in risk when both systems were rendered inoperable. Because RHRSW is the heat sink for post-accident long term core cooling the inspectors also concluded that the condition could affect the operability, availability, reliability, or the function of a system or train in a mitigating system. Therefore, the issue was classified as one that affects a cornerstone and warrants assessment by the significance determination process (SDP).

This item is considered unresolved pending the licensee’s evaluation of Condition Report 20003535, "Potential Single Failure Vulnerability of RHRSW System in Standby with RHRSW-32 Cross-Tie Valve Open," and completion of the associated significance determination process. (Unresolved Item (URI) 50-263/00-08-01)

.2 HPCI Minimum Flow Valve

a. Inspection Scope

The inspectors reviewed the technical adequacy of operability evaluations associated with the HPCI minimum flow valve to determine the impact on TSs and to ensure that adequate justifications were documented. The operability evaluation was selected based upon the relationship of the safety-related system, structure, or component to risk. The documents reviewed included:

- Condition Reports (CRs):
  - 20003071, "Evaluation of Allowable Leak Rate for HPCI Minimum Flow Air Accumulator Check Valve, AI-611, May Not Have Been Bounding"- 20000149, "B RHR Minimum Flow Accumulator and CV-1729/CGCS Air Supply Check Valve Leakage Was Greater Than Allowed by 0255-17-ID-5"

    - 20000549, "A RHR Minimum Flow Air Accumulator Check Valves Fail ASME Section XI Leak Testing"
    - 20000208, "HPCI CV-2065 Air Accumulator Check Valve Failed Leak Rate Test 0255-06-ID-3"

- P&IDs:
    - M-124, Revision Y, "HPCI System [Water Side]"
    - M-123, Revision AF, "HPCI System [Steam Side]"
    - M-123-1, Revision B, "HPCI Hydraulic Control & Lubrication System"

- Operations Manual, Section B.3.2, "HPCI System"

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

- USAR, Revision 18, Section 6.2.4, "HPCI System"

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.

- Post-maintenance testing of control room ventilation radiation monitor RM-9021B following troubleshooting

- Post-maintenance testing of control rod 34-19 following replacement of the associated hydraulic control unit

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:

- Operations Manual:
    - Section B.8.13, "Control Room Heating and Ventilation and Emergency Filtration Train"
    - Section B.1.2, "Control Rod Drives"

- USAR, Revision 18:

- Section 6.7, "Main Control Room, Emergency Filtration Train Building, and Technical Support Center Habitability"
- Section 3.5.3, "Control Rod Drive System"

• TSs:
  - Section 3/4.2.I, "Instrumentation for Control Room Habitability," and Basis
  - Section 3/4.17, "Control Room Habitability," and Basis
  - Section 3/4.3, "Control Rod System," and Basis

• WOs:
  - 0003963, "RM-9021B Control Room Rad Monitor Does Not Operate as Expected During Test"
  - 0003407, "Hydraulic Control Unit for CRD 34-19"

• CR 20004288, "Control Room Radiation Monitor RM-9021B Found Inoperable During Performance of 0460-A, Causing Unexpected LCO"

• Procedures and Forms:
  - 0460-A, Revision 7, "Control Room Air Intake Radiation Monitor Monthly Test"
  - 3069, Revision 8, "Post-maintenance Cover Sheet for WO 0003963"
  - 0081, Revision 31, "Control Rod Drive Scram Insertion Time Test"
  - 4010PM, Revision 9, "HCU Water Accumulator Replacement"
  - 3006, Revision 8, "Stores Requisition"
  - 3186-G-01-03, Revision 4*, "Quality Control Inspection"

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

• Surveillance Test Procedure 0062/0063, Revision 13, "RCIC Steam Line High Area Temperature Test and Calibration Procedure," and Surveillance Test Procedure 0030/0031, Revision 8, "ECCS [Emergency Core Cooling System] High Drywell Pressure Sensor"

• Surveillance Test Procedure 0141, Revision 16, "Reactor Building To Torus Vacuum Breaker Operability Test"

• Surveillance Test Procedure 0255-11-III-3, Revision 21, "13 Emergency Service Water Pump Flow Test"

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-7833-21-1, Revision AC, "Core Spray System"
  - NX-7905-46-3, Revision T, "Residual Heat Removal System"
  - M-143 [NH-36258], Revision AY, "Primary Containment & Atmospheric Control System"
  - M-811, Revision CB, "Service Water System and Make-Up Intake Structure"
  - M-112, Revision BF, "RHR Service Water & Emergency Service Water Systems"

- Surveillance Test Procedures:
  - 0062/0063, Revision 13, "RCIC Steam Line High Area Temperature Test and Calibration Procedure"
  - 0030/0031, Revision 8, "ECCS High Drywell Pressure Sensor"
  - 0141, Revision 16, "Reactor Building To Torus Vacuum Breaker Operability Test"
  - 0255-11-III-3, Revision 21, "13 Emergency Service Water Pump Flow Test"

- AWI 09.04.01, Revision 7, "Inservice Testing Program"

- Calculations:
  - CA-94-106, Revision 0, "Determination of Drywell High Pressure Instrument Setpoints, PS-10-101A, B, C, D," dated April 24, 1995
  - CA-98-044, Revision 6, "Static O-Ring EQ Calculation," dated June 5, 1998
  - CA-94-111, Revision 0, "Determination of RCIC Area High Temperature Instrument Setpoints, TS-13-79 thru 82, A, B, C, D," dated April 28, 1995

- Operations Manual:
  - Section B.4.1, "Containment Systems"
  - Section B.8.1.4, “Emergency Service Water System”
  - Section B.8.13, “Control Room Heating and Ventilation and Emergency Filtration Train”

- TSs:
  - Section 3/4.2, "Protective Instrumentation," and Basis
  - Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis
  - Section 3/4.7, "Containment Systems," and Basis

- USAR, Revision 18:
  - Section 6.7, "Main Control Room, Emergency Filtration Train Building, and Technical Support Center Habitability"
  - Section 10.4.4, "Emergency Service Water System"

- Northern States Power letter to the NRC dated April 25, 1972, discussing HPCI steam line high area temperature switch

- Monticello Significant Operating Event No. 69, "Failure to Obtain Calibration Data Required to Evaluate Performance of the HPCI & RCIC High Area Temperature Steam Line Isolation Switches"

- ASME/ANSI OMa-1988, Part 10, "Operation And Maintenance of Nuclear Power Plants"

- NUREG 1482, Revision 4/95, "Guidelines For Inservice Testing At Nuclear Power Plants"

- 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. Issues and Findings

There were no findings identified during this inspection.

1EP6 Drill Evaluation

a. Inspection Scope

The resident inspectors reviewed an annual emergency preparedness exercise to evaluate drill conduct and the adequacy of the licensee's critique of performance to identify weaknesses and deficiencies. The inspectors selected an exercise that the licensee had scheduled for the purpose of providing input to the Drill/Exercise Performance Indicator. The inspectors observed, when applicable, the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critiques. Inspector 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:

- Operations Manual A.2 - 101, Revision 26, "Classification of Emergencies"

- Procedures and Forms:
  - A.2-809, Revision 2, "EOF [Emergency Operations Facility] Security"
  - MTCP.06.02, Revision 9, "Site Drill & Exercise Manual"

- 5790-001-01, Revision 21, "Emergency Response Organization"
- 5790-803-01, Revision 11, "EOF Reclassification Call-List"
- 5790-102-02, Revision 25, "Monticello Emergency Notification Report"
- 5790-104-04, Revision 72, "Emergency Dall List Alert/Site Area/General"
- 3195, Revision 20, "Event Notification Worksheet"
- 3695, Revision 2, "EP [emergency preparedness] Performance Record"

b. Issues and Findings

There were no findings identified during this inspection.

3. SAFEGUARDS

Cornerstone: Physical Protection

PP4. Security Plan Changes (IP 71130.04)

a. Inspection Scope

The inspector reviewed Revision 16 of the Monticello Security Force Training and Qualification Plan which was submitted by licensee letter, dated August 30, 2000, to verify that the change did not decrease the effectiveness of the security plan. The security plan was submitted in accordance with 10 CFR 50.54(p).

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Cornerstone: Mitigating Systems

Safety System Functional Failures

a. Inspection Scope

The inspectors verified the performance indicator data for Safety System Functional Failures from January 1, through September 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 Q3/2000

- Nuclear Energy Institute 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline"

- 4AWI-04.08.11, Revision 1, "NRC Performance Indicator Reporting"

- Monticello Operations Daily Log - Part J, Revision 76

b. Issues and Findings

There were no findings identified during this inspection.

4OA3 Event Follow-up

Cornerstones: Mitigating Systems and Barrier Integrity

(Closed) LER 50-263/2000-008: Primary Containment Isolation of TIP (Transversing In-core Probe) Ball Valves Does Not Function Independently of Normal Controls.

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-008, "Primary Containment Isolation of TIP Ball Valves Does Not Function Independently of Normal Controls." The inspectors reviewed the following references:

- CR 20001096, "Primary Containment Isolation of TIP Ball Valves Does Not Function Independently of Normal Controls"

- TS, Section 3/4.7, "Containment Systems," and Basis

- USAR, Revision 18
  - Section 5.2, "Primary Containment"
  - Table 5.2-3b, "Primary Containment Automatic Isolation Valves"

- Safety Review Item 00-014, Revision 0, "Justification For Performing Periodic TIP Scans And Ball Valve Surveillances"

- Calculation CA-00-062, Dated 4/3/00, "Radiological Consequence of TIP Ball Valve Failure Following a DBA-LOCA [Design Basis Accident-Loss of Coolant Accident]"

b. Issues and Findings

Licensee engineering personnel identified a design deficiency associated with the plant's TIP system. Specifically, non-safety related relays and probe position transducers were identified as being used in TIP guide tube ball isolation valve control circuitry. A failure in one of these non-safety related components could have led to the spurious opening of the TIP ball valves following a design basis accident.

The licensee disabled the TIP ball valves in the closed position by de-energizing power to the valve drive mechanisms to prevent spurious opening under accident conditions. Additionally, the licensee performed a calculation to evaluate the radiological consequences associated with the spurious opening of the TIP ball valves and the potential leakage path associated with the TIP system when in use. Because the calculated leakage was well below the containment limiting value of L_a and the 10 CFR 100 limit, the inspectors concluded that this issue was of very low risk significance. The licensee entered this issue into their corrective action program as CR 20001096.

4OA6 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Hammer and other members of licensee management on November 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 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

50-263/00-08-01	URI	Potential vulnerability associated with the RHRSW header cross-tie (1R15)

Closed

50-263/2000-008	LER	Primary Containment Isolation of TIP Ball Valves Does Not Function Independently of Normal Controls (4OA3)

Discussed

None

LIST OF ACRONYMS USED

AWI	Administrative Work Instruction
CR	Condition Report
CRD	Control Rod Drive
CRV	Control Room Ventilation
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
EOF	Emergency Operations Facility
EP	Emergency Preparedness
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NUMARC	Nuclear Management and Resources Council
P&ID	Piping and Instrument Diagram
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SDP	Significance Determination Process
TIP	Traversing In-core Probe
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
VDC	Volt Direct Current
WO	Work Order