December 4, 2001

Mr. J. Forbes
Site Vice-President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT
NRC INSPECTION REPORT 50-263/01-09(DRP)

Dear Mr. Forbes:

On November 14, 2001, the NRC completed an inspection at your Monticello Nuclear Generating Plant. The results of this inspection were discussed on November 16, 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.

Based on the results of this inspection, the NRC identified two issues of very low safety significance (Green) and within the licensee response band. The first finding involved inadequate post-maintenance testing procedures associated with a 480 Vac circuit breaker overhaul. The second finding involved the lock up of both feedwater regulating valves in the fully open position during a recent scram, which complicated the operating crew's response to that event.

The NRC continues to interact with the Intelligence Community and to communicate information to the Nuclear Management Company, LLC. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Bruce L. Burgess, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 50-263/01-09(DRP)

cc w/encl:
J. Purkis, Plant Manager
R. Anderson, Executive Vice President
and Chief Nuclear Officer
Nuclear Asset Manager
Site Licensing Manager
Commissioner, Minnesota Department of Health
J. Silberg, Esquire
Shaw, Pittman, Potts, and Trowbridge
R. Nelson, President
Minnesota Environmental Control Citizens
Association (MECCA)
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer
Wright County Government Center
Commissioner, Minnesota Department of Commerce
A. Neblett, Assistant Attorney General

DOCUMENT NAME: G:\mont\mon2001009 drp.wpd

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

OFFICE	RIII		RIII					
NAME	GWright/trn		BBurgess					
DATE	12/ /01		12/04/01					

OFFICIAL RECORD COPY

ADAMS Distribution:
WDR
DFT
CFL
RidsNrrDipmIipb
GEG
HBC
SXB3
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263/01-09(DRP)
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	October 1 through November 14, 2001
Inspectors:	S. Burton, Senior Resident Inspector D. Kimble, Resident Inspector R. Daley, Regional Inspector
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000263-01-09(DRP), on 10/01-11/14/2001; Nuclear Management Company, LLC; Monticello Nuclear Generating Plant; Post-Maintenance Testing; Event Response.

The inspection was conducted by resident inspectors and regional inspectors. The report covers a 6½-week period of resident inspection. The inspection identified two Green findings. The significance of all of the findings are indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

- Green. The inspectors reviewed licensee post-maintenance testing activities associated with restoration of a 480 Vac load center supply breaker following a 10-year overhaul. This finding was of very low safety significance because the loss of function of the No. 13 diesel generator was of a short duration. (Section 1R19)

- Green. The inspectors monitored the licensee's response to an inadvertent reactor scram which occurred on October 23, 2001. The lock up of both feedwater regulating valves in the fully open position complicated operating crew response to the scram. The finding was of very low safety significance because the impact on the operating crew was minimal, and operator action was available to restore system functions. (Section 4OA3)

B. Licensee-Identified Violations

None.

Report Details

Summary of Plant Status

The Unit began the inspection period operating at full power. On October 18, 2001, the Unit entered end-of-life coastdown for the current operating cycle. On October 23, 2001, the Unit scrammed from an inadvertent Group 1 isolation (Sections 1R14, 1R20.1, and 4OA3). On October 25, 2001, the reactor was restarted, and maximum power consistent with the Unit's end-of-life coastdown condition was achieved on October 26, 2001. On October 27, 2001, power was reduced to approximately 80 percent to perform a control rod pattern adjustment. Maximum attainable power (approximately 94 percent) was restored later that day. Power continued to decrease as a result of the end-of-life coastdown until the Unit was shutdown to begin Refueling Outage No. 20 on November 3, 2001. The Unit remained shutdown for refueling through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed a partial walkdown of various risk-significant equipment to verify operability and proper equipment lineup while other systems and/or components were disabled due to planned maintenance. The inspected systems and components were selected due to the increase in core damage frequency caused by rendering the following equipment out-of-service for maintenance.

- No. 11 Emergency Diesel Generator (EDG)
- High Pressure Coolant Injection (HPCI) System
- Reactor Core Isolation Cooling (RCIC) System

The inspectors verified the position of critical redundant equipment and looked for any discrepancies between the existing equipment lineup and the required lineup.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Fire Zone Inspections

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 15-A, No. 12 EDG Room
- Fire Zone 15-B, No. 11 EDG Room and Day Tank Rooms
- Fire Zone 3-A, Recirculation Motor Generator Room
- Fire Zone 4-D, Standby Gas Treatment System Room
- Fire Zone 37, Station Main and Auxiliary Transformers

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.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

The inspectors conducted an annual observation of the station's fire brigade during a drill which simulated a diesel oil fire in the No. 13 diesel generator room. The inspectors evaluated the readiness of the licensee's personnel to prevent and fight fires by verifying that: protective clothing/turnout gear was properly donned; self-contained breathing apparatus equipment was properly worn and used; fire hose lines were capable of reaching all necessary fire hazard locations, the lines were laid out without flow constrictions, the hoses were simulated being charged with water, and the nozzles were pattern (flow stream) tested prior to entering the fire area; the fire area was entered in a controlled manner; sufficient fire fighting equipment was brought to the scene by the fire brigade; the fire brigade leader's directions were thorough, clear, and effective; communications with plant operators and between fire brigade members were efficient and effective; the fire brigade checked for fire victims and for fire propagation into other plant areas; effective smoke removal operations were simulated; fire fighting pre-plan strategies were utilized; the drill scenario was followed and the drill objectives met.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

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 EDGs. The EDGs were selected based on being designated as risk significant under the Maintenance Rule.

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 reviewed; and current equipment performance status.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed and observed emergent work, preventive maintenance, or planning for risk significant maintenance activities. The inspectors observed maintenance or planning for the following activities or risk significant systems undergoing scheduled or emergent maintenance.

- No. 11 EDG Circulating Oil Pump Replacement
- Annunciator Panel C-06C Troubleshooting and 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, control of maintenance, and external impacts on risk. In-plant activities were reviewed to ensure that the risk assessment of maintenance or emergent work was complete and adequate, and that the assessment included an evaluation of external factors. Additionally, the inspectors verified that the licensee entered the appropriate risk category for the evolutions.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors reviewed personnel performance during an unplanned scram on October 23, 2001. The inspectors independently evaluated the initiating cause of the scram and operator actions in response to the event. This evaluation included a review of operator logs and plant computer data, and personnel interviews.

Due to the nature of the event, an issue involving an inadvertent partial reactor depressurization that occurred following the scram while the plant was in a hot shutdown condition was inspected as part of the Problem Identification and Resolution Inspection which took place from October 29, 2001, through November 9, 2001. Details are documented in NRC Inspection Report 50-263/01-16.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of the operability evaluation for the No. 12 residual heat removal (RHR) heat exchanger tube side relief valve, RV-3203, following a setpoint test failure, to determine the impact on Technical Specifications and to ensure that the evaluations that were documented were adequate. This operability evaluation was selected based upon the relationship of the safety-related component, specifically the No. 12 RHR heat exchanger, to shutdown risk.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (OWA) (71111.16)

.1 Review of Selected Operator Workarounds

a. Inspection Scope

The inspectors reviewed the following operator workarounds:

- No. 01-023, "Control Rod Drive (CRD) Pump Trip on HPCI Start"
- No. 01-136, "11 CRD Pump Discharge Check Valve (CRD-4-1) Will Not Seat"

The inspectors reviewed each workaround's potential to impact the operators' ability to respond to an initiating event or plant transient. Additionally, the inspectors evaluated the effect of each workaround on the operators' ability to implement abnormal and emergency operating procedures.

b. Findings

No findings of significance were identified.

.2 Semiannual Review of the Cumulative Effects of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the licensee's list of current OWAs for their cumulative effects on the reliability, availability, and, potential for misoperation of any system or component. Additionally, the inspectors reviewed the cumulative effects of OWAs for any increase in initiating event frequency or adverse impact on multiple mitigating systems. The inspectors evaluated the aggregate impact of the licensee's OWAs on the ability of operators to respond in a correct and timely manner to plant transients and accidents.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- No. 11 EDG Circulating Oil Pump Replacement
- Load Center (LC) 107 480 Vac Supply Breaker 52-701 10-Year Overhaul

The inspectors verified by witnessing the test or reviewing the test data that post-maintenance testing activities were adequate for the above maintenance activities. The inspectors' reviews 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, Technical Specification applicability, system restoration, and evaluation of test data. Also, the inspectors verified that maintenance and post-maintenance testing activities adequately ensured that the equipment met the licensing basis, Technical Specifications, and USAR design requirements.

b. Findings

A finding of very low significance (Green) relating to the post-maintenance testing of a 480 Vac LC supply breaker was identified by inspectors.

On October 1, 2001, the main supply breaker (52-701) for 480 Vac LC 107, load center breaker (LCB) 083, was returned to the plant following a 10-year overhaul at an offsite repair facility, Xcel Energy's Chestnut Service Center. As the breaker was being racked into its cubicle, the No. 13 diesel generator output breaker (52-710), which is also located on LC-107, tripped and locked out, rendering the diesel physically unavailable for several hours. LCB-083 was removed from the cubicle and examined. During an inspection, plant electricians noticed that the bell alarm contacts for the 52-107 breaker were closed and would not reset. The closed bell alarm contacts caused the lockout of the No. 13 diesel generator breaker. LCB-083 was returned to the Chestnut Service Center on October 2, 2001, for rework.

Technicians at the service center found an arc chute retaining bolt lodged in the bell alarm reset mechanism. The condition was corrected and the breaker bench tested for overcurrent tripping and proper operation of all auxiliary contacts. The breaker was returned to the plant on October 3, 2001, and again returned to its cubicle. However, the breaker failed to close and trip when cycling was attempted in the "test" position. While the breaker was still in the cubicle, it was discovered that a mechanical trip mechanism was not operating properly. The breaker was again removed from the cubicle and examined. The cause of problem was determined to be a guide roller that would not roll freely. The breaker was again reworked on October 4, 2001, and returned to service following successful operation in the "test" position.

Upon examination of this issue, inspectors determined it to be more than minor in that, in causing the No. 13 diesel generator to become locked out, the attempted return of LCB-083 to its cubicle with the bell alarm contacts for the breaker closed had an actual and credible impact on plant safety. Additionally, inspectors determined that the issue affected the mitigating systems cornerstone of reactor safety in that it had an actual impact on the function of a mitigating system, specifically the No. 13 diesel generator.

The inspectors employed the SDP to determine the potential risk significance of the finding. Although in causing the lockout of the No. 13 diesel generator it affected the function of non-Technical Specification risk-significant equipment, because this loss of function was for such a short duration the finding was determined to be of very low significance and within the licensee's response band (Green). The licensee entered this issue into their corrective action program as Condition Report (CR) 20015786 (Finding (FIN) 50-263/01-09-01(DRP)).

1R20 Outage Activities (71111.20)

.1 Unscheduled Maintenance Outage

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled maintenance outage that began on October 23, 2001, following an inadvertent reactor scram (Section 4OA3), and ended on October 25, 2001. The inspectors reviewed activities to ensure that the licensee appropriately considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the post-scram reactor shutdown activities, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, identification and resolution of problems associated with the outage, and reactor startup and heat up activities.

b. Findings

No findings of significance were identified.

.2 Scheduled Refueling Outage

a. Inspection Scope

The inspectors evaluated outage activities for a refueling outage that began on November 3, 2001, and continued through the end of the inspection period. The inspectors reviewed activities to ensure that the licensee appropriately considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, and identification and resolution of problems associated with the outage.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors selected the following surveillance test activities for review. Activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a system, structure, or component could impose on the Unit if the condition were left unresolved.

- Standby Liquid Control (SLC) Quarterly Pump and Valve Test

- No. 14 Emergency Service Water (ESW) Pump and Valve 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, procedure use and adherence, control of temporary modifications or jumpers required for test performance, documentation of test data, Technical Specification applicability, impact of testing relative to performance indicator reporting, and evaluation of test data.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary modifications:

- No. 01-08, Control Wiring for Automatic Isolation of V-O-39 and V-O-40
- No. 01-35, Replacement of Failed Excitation Transformer for #11 Cooling Tower Pump
- No. 01-74, Temporary Circuit Protection During Replacement of 125 Vdc Fuses

The inspectors reviewed the safety screening, design documents, USAR, and applicable Technical Specifications to determine that the temporary modification was consistent with modification documents, drawings, and procedures. The inspectors also reviewed the post-installation test results to confirm that tests were satisfactory and the actual impact of the temporary modification on the permanent system and interfacing systems were adequately verified.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors verified the accuracy and completeness of the “Safety System Functional Failure” performance indicator data submitted by the licensee from July 2000 through September 2001. The inspectors reviewed data reported to the NRC since the last verification. The review was accomplished, in part, through evaluation of the Technical Specification requirements, plant records, and procedural reviews.

b. Findings

No findings of significance were identified.

.2 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors verified the accuracy and completeness of the “Reactor Coolant System Activity” performance indicator data submitted by the licensee from September 2000 through September 2001. The inspectors reviewed data reported to the NRC since the last verification. The review was accomplished, in part, through evaluation of the Technical Specification requirements, chemistry records, procedural reviews, and reactor coolant sample data.

b. Findings

No findings of significance were identified.

4OA3 Event Follow Up (71153)

a. Inspection Scope

On October 23, 2001, inspectors responded to an inadvertent reactor scram initiated by a Group 1, main steam isolation valve, closure signal which occurred when a radiation protection technician bumped a sensitive instrument rack in the reactor building. The inspectors monitored critical plant parameters and observed and evaluated the status of mitigating systems, fission product barriers, and licensee actions to respond to the scram. Inspectors verified that the licensee made timely and appropriate notifications as required by 10 CFR 50.72. Additionally, inspectors communicated the details of the event to NRC Region III management to aid in the determination of its risk significance.

b. Findings

A finding of very low significance (Green) associated with the lock up of both feedwater regulating valves (FRVs) during a recent scram was identified by inspectors.

On October 23, 2001, immediately following an inadvertent reactor scram from a Group 1 isolation, both FRVs locked up in the fully open position. This failure contributed to a subsequent high water level condition in the reactor vessel which caused reactor feed pumps to trip and complicated the operating crew's response to the scram.

Inspectors reviewed selected data from the scram and the licensee's analysis of the event. At the point of the scram resulting from the loss of steam flow, reactor level dropped rapidly from a nominal 35 inches. The digital feedwater control system (DFCS) sent a full open demand to each FRV. Instrumentation plots of total feedwater flow at this point indicated that the FRVs had, in fact, fully opened. Reactor level trend plots indicated that level ceased decreasing at approximately minus-seven inches, and began to rapidly rise. As level rose, the DFCS began reducing the valve demand as expected. However, no concurrent change in total feedwater flow was observed until level reached 48 inches and the feedwater pumps tripped on high level. This indicates that, while the DFCS was calling for the valves to close, they never responded.

Following troubleshooting and investigation, the licensee was able to recreate the FRV lock up event. The DFCS incorporates a design feature that "freezes" FRV position "as-is" if the DFCS demand signal and actual valve position diverge by more than 27 percent within a 100 msec feedback window. The feature is intended to protect the plant from a feedwater transient, which could be caused as a result of various DFCS failures. During the recreation of the event, the licensee determined that under severe feedwater transient response conditions, such as that experienced during a Group 1 isolation, the 100 msec feedback window may be too short and that actual valve response can lag the DFCS demand signal by up to 200 msec. The licensee updated the DFCS feedback window to 300 msec to correct the condition via their setpoint change process.

Inspectors determined the finding to be more than minor in that the lock up of both FRVs in the fully open position had an actual and credible impact on plant safety. While not beyond the operating crew's ability to cope with, the condition contributed to the high reactor vessel water level condition and associated trip of both reactor feedwater pumps which complicated the scram response. Additionally, inspectors determined that the issue affected the mitigating systems cornerstone of reactor safety in that it had an actual impact on the reliability of a mitigating system, specifically the main feedwater and condensate system.

The inspectors employed the SDP to determine the potential risk significance of the finding. During a Phase 1 SDP review, the inspectors determined that because the operating crew had the ability to perform manual actions to recover from the condition, the function of the mitigating system was not lost. Additionally, the lock up of both FRVs is bounded by exiting transient analyses for the plant. As a result, the finding was determined to be of very low significance and within the licensee's response band

(Green). The licensee entered this issue into their corrective action program as CR 20016421 (FIN 50-263/01-09-02(DRP)).

4OA6 Meeting

Exit Meeting

The inspectors presented the inspection results to Mr. Forbes and other members of licensee management on November 16, 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.

KEY POINTS OF CONTACT

Licensee

G. Bregg, Manager, Quality Services
D. Fadel, Director of Engineering
J. Forbes, Site Vice-President
J. Grubb, General Superintendent, Engineering
K. Jepson, General Superintendent, Chemistry and Radiation Services
B. Linde, Superintendent, Security
D. Neve, Acting Licensing Project Manager
J. Purkis, Plant Manager
B. Sawatzke, General Superintendent, Maintenance
C. Schibonski, General Superintendent, Safety Assessment
E. Sopkin, General Superintendent, Operations

NRC

B. Burgess, Chief, Reactor Projects Branch 2

ITEMS OPENED, CLOSED, AND DISCUSSED

Open

50-263/01-09-01	FIN	Inadequate Post-Maintenance Testing Following Circuit Breaker Overhaul (Section 1R19)
50-263/01-09-02	FIN	Scram Response Complicated By Lock Up of Both Feedwater Regulating Valves (Section 4OA3)

Closed

50-263/01-09-01	FIN	Inadequate Post-Maintenance Testing Following Circuit Breaker Overhaul (Section 1R19)
50-263/01-09-02	FIN	Scram Response Complicated By Lock Up of Both Feedwater Regulating Valves (Section 4OA3)

Discussed

None.

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
AWI	Administrative Work Instruction
CFR	Code of Federal Requirements
CR	Condition Report
CRD	Control Rod Drive
DC	Direct Current
DFCS	Digital Feedwater Control System
DGN	Diesel Generators
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EPR	Electronic Pressure Regulator
ESW	Emergency Service Water
EWI	Engineering Work Instruction
FRV	Feedwater Regulating Valve
HPCI	High Pressure Core Injection
IEEE	Institute of Electrical & Electronic Engineers
IPEEE	Individual Plant Examination of External Events
IV	Independent Verification
LC	Load Center
LCB	Load Center Breaker
LER	Licensee Event Report
msec	millisecond
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NUMARC	Nuclear Management and Resources Council
OWA	Operator Workaround
OWI	Operations Work Instruction
PM	Planned or Preventative Maintenance
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SCR	Setpoint Change Request
SDP	Significance Determination Process
SLC	Standby Liquid Control
SRI	Safety Review Item
SWI	Scheduling Work Instruction
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current
WO	Work Order
WM	Welding Manual
WRGM	Wide Range Gas Monitor

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

	Operations Manual:	
B.09.08	- Emergency Diesel Generators	
B.03.02	- HPCI System	
B.08.01.02	- EDG ESW System	
B.02.03	- RCIC System	
B.09.06	- 4160 Vac System	
	Drawings:	
M-123	- HPCI System (Steam Side)	Revision AF
M-123-1	- HPCI Hydraulic Control and Lubrication	Revision B
M-124	- HPCI System (Water Side)	Revision Y
M-125	- RCIC System (Steam Side)	Revision AK
M-126	- RCIC System (Water Side)	Revision Y
M-110-1	- Service Water System	Revision BL
M-112	- RHR and Emergency Service Water Systems	Revision BF
M-811-1	- Service Water and Make-Up Intake Structure	Revision CD
M-133-1	- Diesel Oil System	Revision AD
NF-36298-1	- Electrical Load Flow	Revision M
NF-36298-2	- DC Electrical Load Distribution	Revision A

1R05 Fire Protection

NX-16991	Monticello Updated Fire Hazards Analysis	
	Monticello Fire Strategies:	
A.3-15-A	- No. 12 EDG Room	Revision 4
A.3-15-B	- No. 11 EDG Room and Day Tank Rooms	Revision 5
A.3-03-A	- Recirculation Motor Generator Room	Revision 3*
A.3-04-D	- Standby Gas Treatment System Room	Revision 3*
A.3-37	- Station Main and Auxiliary Transformers	Revision 2*
	Procedures and Administrative Work Instructions (AWIs):	
4AWI-08.01.01	- Fire Prevention Practices	Revision 17
4AWI-08.01.02	- Combustion Source Use Permit	Revision 6
2176	- Fire Drill Procedure	Revision 11
A.3-004	- Fire Fighting Procedures and Strategies	Revision 4
A.2-101	- Classification of Emergencies	Revision 27
QUAD-5-80-009	Quadrex Corporation Report, Specifications for Installation of Electrical and Mechanical Penetration Seals at the Monticello Nuclear Generating Plant	Revision 7

NSPLMI-95001	Monticello Individual Plant Examination of External Events (IPEEE), November 1995	Revision 1
Drill Guide No. 20	Fire Brigade Training - No. 13 Diesel Generator Enclosure Fire	

1R12 Maintenance Rule Implementation

	NUMARC [Nuclear Management and Resources Council]:	
93-01	- Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	Revision 2
93-01, Section 11	- Assessment of Risk Resulting from the Performance of Maintenance Activities	February 22, 2000
	Regulatory Guides:	
1.160	- Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	Revision 2
1.182	- Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants	May 2000
EWI-05.02.01	Monticello Maintenance Rule Program Document	Revision 5
	Monticello Maintenance Rule Periodic Assessment Report	1st Quarter - 2001
	Operations Manual:	
Section B.9.08	- Diesel Generators	
	Maintenance Rule Program System Basis Document:	
Section B.9.8	- Diesel Generators	Revision 1
	USAR:	
Section 8.4	- Plant Standby Diesel Generator Systems	Revision 18
IEEE 323	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations	1974
IEEE 387	IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations	1977
SRI 90-031	Clarification of EDG Starting Air Requirements	
	Maintenance Rule Summary Report for System DGN	

Regulatory Guide 1.9	Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electrical Power Systems at Nuclear Power Plants	Revision 3

1R13 Maintenance Risk Assessments and Emergent Work Control

	Procedures:	
4AWI-04.01.01	- General Plant Operating Activities	Revision 28
4AWI-08.15.01	- Risk Management Fore Outage and On-line Activities	Revision 0
SWI-14.01	- Risk Management of On-line Maintenance	Revision 0
WO 0109202	Repair/Replace No. 11 EDG Circulating Oil Pump	
WO 0109334	Annunciator Cabinet C-06C Not Working Correctly	
CR 20016131	Loss of Annunciators on C-06C Results in Increased Compensatory Actions and Abnormal Procedure Entry	
	Operations Manual:	
B.09.08	- Emergency Diesel Generators	
B.05.13	- Annunciators	
	Drawings:	
M-133-1	- Diesel Oil System	Revision AD

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

	Operations Manual:	
C.3	- Shutdown Procedure	Revision 27
C.4-A	- Reactor Scram	Revision 19
C.4-B.5.7.A	- Loss of Reactor Water Level Control	Revision 5
C.4-B.6.5.A	- Reactor Feedwater Pump Trip	Revision 5
C.1	- Startup Procedure	Revision 32
	Scram 112 Summary Report	
WO 0109431	Investigate FRV Lockup	
0000-B	Operations Daily Log - Part B	Revision 82

1R15 Operability Evaluations

3108	Pump / Valve / Instrument Record of Corrective Action for RV-3203	Revision 7
CR 20016569	RV-3203 Test Values Outside Acceptance Bands for 0255-05-IB-02	
WO 0107782	Remove and Reinstall RV-3203 for Testing	
ASME OM-1	Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices	1981 Edition

1R16 Operator Workarounds

	Monticello Operational Challenges List	10/11/01
	Operations Integrated Improvement Plan (2001-2003)	10/6/01
	Operations Department Quarterly Effectiveness Reports: - 1st Quarter 2001 - 2nd Quarter 2001	 6/2/01 8/29/01
OWI-01.07	Operations Department Self Assessment	Revision 14
2220	Operational Challenge Resolution - Operator Workarounds	Revision 2
OWA 01-023	CRD Pump Trips on HPCI Initiation	
OWA 01-136	11 CRD Pump Discharge Check Valve (CRD-4-1) Will Not Seat	
CR 20010874	CRD Pump Trips on HPCI Initiation	
CR 20015689	11 CRD Pump Discharge Check Valve (CRD-4-1) Will Not Seat	

1R19 Post-Maintenance Testing

WO 0109204	Rework Bell Alarm on LCB-083, LC-107 Main Breaker
WO 0109202	Repair/Replace No. 11 EDG Circulating Oil Pump
CR 20015819	Rework of LCB-083 Following Performance of 4851-11-PM

CR 20015937	Revise 4851-XX-PMs, Verify Mechanisms for Racking, Reset, Trip for Ease of Movement and IV of Reassembly Following Major PM	
4851-11-PM	General Electric AKR-6D-75 480 Vac Breaker Mechanical and Electrical Maintenance	Revision 4

1R20 Outage Activities

	Operations Manual:	
C.3	- Shutdown Procedure	Revision 27
	Drawings:	
M-135	- Fuel Pool Cooling and Clean-up System	Revision AB
9001	Reactor Well and Dryer - Separator Storage Pool Filling Procedure	Revision 13
WO 0005177	Disassemble / Inspect / Reassemble No. 11 RHR Heat Exchanger	
8151	Heavy Load Movement Procedure	Revision 6
8136-01	Secondary Containment Penetration Work Control Index	Revision 1

1R22 Surveillance Testing

0255-02-III	Standby Liquid Control System Pump Inservice Test	Revision 34
0255-02-IA-1	Standby Liquid Control System Valve Inservice Test	Revision 34
0255-11-III-4	14 Emergency Service Water Pump Flow Test	Revision 27
0255-11-IA-6	14 Emergency Service Water Valve Operability Test	Revision 27
4AWI-09.04.01	Inservice Testing Program Implementation	Revision 8
EWI-09.04.01	Inservice Testing Program	Revision 6
EWI-08.04.02	Relief Valve Testing and Repair	Revision 1
WM 460	Testing and Repair of ASME and National Board Stamped Pressure Relief Valves (Xcel Energy Welding Manual)	9/9/98

	Operations Manual:	
B.03.05	- Standby Liquid Control	
B.08.01.04	- Emergency Service Water	
	Drawings:	
M-127	- Standby Liquid Control	Revision V
M-110-1	- Service Water System	Revision BL
M-112	- RHR and Emergency Service Water Systems	Revision BF
M-811-1	- Service Water and Make-Up Intake Structure	Revision CD
NUREG 1482	Guidelines For Inservice Testing at Nuclear Power Plants	April 1995
	Technical Specifications and Bases:	
Section 3.4/4.4	- Standby Liquid Control System	
Section 3.17/4.17	- Control Room Habitability	
WO 0109404	No Flow Through ESW-41 When Opened	
CR 20016404	Step in Test 0255-11-III-4 Does Not Address the Possibility of the Drain Path Being Plugged, and No Vent Path	
ASME OMa-1988	Operation and Maintenance of Nuclear Power Plants	

1R23 Temporary Plant Modifications

Jumper Bypass No. 01-08	Provide Temporary Air to EPR-8000K-27A and to EPR-8000K-27B. Change Control Wiring for Automatic Isolation of V-O-39 and V-O-40	2/21/01
Jumper Bypass No. 01-35	Replacement of Failed Excitation Transformer for #11 Cooling Tower Pump	7/13/01
Jumper Bypass No. 01-74	Provide Temporary Circuit Protection During Replacement of 125 Vdc Fuses	10/23/01
B.04.02	Operations Manual: Secondary Containment/Standby Gas Treatment System	
4AWI-04.04.03	Bypass Control	Revision 16
NEI 96-07	Guidelines for 10 CFR 50.59 Evaluations	Revision 1
	Drawings:	
NE-36640-2	- 125V DC Distribution Electrical Scheme	Revision J
NE-36640-3	- 125, 250 & 24 Volt DC Systems	Revision AA
NE-36375-20	- Reactor Building Main Exhaust Fans V-EF-24A & B and Spent Fuel Pool Exhaust Fan V-EF-28	Revision N

4OA1 Performance Indicator Verification

	Monticello Performance Indicator Data Summary Report - 3rd Quarter 2001	October 11, 2001
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	Revision 1
3530-06	Performance Indicator Radiation Safety Worksheets: - 4th Quarter 2000 - 1st Quarter 2001 - 2nd Quarter 2001 - 3rd Quarter 2001	Revision 1
TS 3.6.C/4.6.C	Reactor Coolant Chemistry and Basis	
LER 2001-03	Inadequate Procedures Result in Failure to Recognize Entry into 36-Hour Limiting Condition for Operation Required When Standby Gas Treatment System Doors Opened for Access	Revision 0
LER 2001-05	Ten Minute Torus Cooling Design Assumption Not Achievable	Revision 0
LER 2001-06	Alternate Shutdown System Design Deficiencies Result in Vulnerability to Single Hot Shorts During Postulated Control Room or Cable Spreading Room Fire	Revision 0
CR 20016842	Potential Failure to Report 3 LERs as Safety System Function Failures in the 2nd Quarter NRC Performance Indicator Report	

4OA3 Event Follow Up

96Q175	Digital Feedwater Control System Modification	
RP2448-3-001	Digital Feedwater Control System Requirements Specification	Revision G
RP2448-3-002	Digital Feedwater Control System Purchase Specification	Revision 1
SCR 01-044	Setpoint Change Request for DFCS Time Delay Constant	Revision 0
3278	10 CFR 50.59 Screening for SCR 01-044	Revision 3*

	Operations Manual:	
C.3	- Shutdown Procedure	Revision 27
C.4-A	- Reactor Scram	Revision 19
C.4-B.5.7.A	- Loss of Reactor Water Level Control	Revision 5
C.4-B.6.5.A	- Reactor Feedwater Pump Trip	Revision 5
C.1	- Startup Procedure	Revision 32
	Scram 112 Summary Report	
WO 0109431	Investigate FRV Lockup	
0000-B	Operations Daily Log - Part B	Revision 82