October 24, 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 50-263-01-08(DRP)

Dear Mr. Forbes:

On September 30, 2001, the NRC completed an inspection at your Monticello Nuclear Generating Plant. The results of this inspection were discussed on September 28, 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, 10 CFR 50.59 evaluations, 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 one issue of very low safety significance (Green) involving a violation of NRC requirements. If you deny this Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 801 Warrenville Road, Lisle, Illinois 60532-4351, the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspectors' Office at the Monticello Nuclear Generating Plant

Since September 11, 2001, Monticello Nuclear Generating Plant has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

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/NRC/ADAMS/index.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 50-263-01-08(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
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D. Gruber, Auditor/Treasurer
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263/01-08(DRP)
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	August 15 through September 30, 2001
Inspectors:	S. Burton, Senior Resident Inspector D. Kimble, Resident Inspector S. Ray, Senior Resident Inspector - Prairie Island Station S. Thomas, Resident Inspector - Prairie Island Station D. Chyu, Regional Inspector C. Phillips, Regional Inspector D. Pelton, Regional Inspector
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000263-01-08(DRP), on 08/15-09/30/2001; Nuclear Management Company, LLC; Monticello Nuclear Generating Plant; Other Activities; Flood Protection.

The inspection was conducted by resident inspectors and regional inspectors. The report covers a 6½-week period of resident inspection. The inspection identified a Green finding associated with a Non-Cited Violation. The significance of the finding is indicated by its 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

Cornerstone: Mitigating Systems, Initiating Events

Green. The inspectors identified a procedure deficiency for mitigation of flooding events. Licensee procedures related to flooding failed to establish a maximum allowable level of water accumulation between an erected berm and emergency diesel generator buildings. Excessive accumulation of water would cause hydraulic lift of the floor causing failure of the associated diesel generator.

The lack of procedural controls to adequately respond to flooding of the emergency diesel generator constituted a Non-Cited Violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instruction, Procedures and Drawings.” These findings were of very low safety significance because of the low probability of occurrence of a flooding event that would rise to the elevation required to cause hydraulic lift of the emergency diesel generator floor. (Section 4OA3.2)

B. Licensee-Identified Violations

One violation of very low significance identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. This violation is listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

The unit began the inspection period operating at or near full power. On September 9-10, 2001, power was reduced to approximately 90 percent for a routine rod pattern adjustment. Power was reduced again on September 15-16, 2001, to approximately 77 percent for additional planned rod pattern adjustments. On September 27, 2001, an unplanned power reduction was performed to approximately 55 percent to allow the No. 12 reactor feed pump to be removed from service for emergent repairs to the motor cooler. Repairs were completed on September 28, 2001, and the unit returned to approximately 100 percent power on September 29, 2001. The unit remained at approximately full power through the end of the inspection period.

1. REACTOR SAFETY

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

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of Evaluations and Screenings for Changes, Tests, or Experiments

a. Inspection Scope

The inspector reviewed eight safety evaluations performed pursuant to Federal Regulations 10 CFR 50.59. The safety evaluations were related to temporary and permanent plant modifications, set-point changes, procedure changes, potential conditions adverse to quality, and changes to the licensee's updated safety analysis report (USAR). The inspectors confirmed that the safety evaluations were thorough and that prior NRC approval was obtained when appropriate. The inspectors also reviewed 11 safety evaluation screenings, where the licensee had determined that a 10 CFR 50.59 safety evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 safety evaluation was performed, the inspectors reviewed the changes to verify that they did not meet the threshold requiring a 10 CFR 50.59 safety evaluation. These safety evaluations and screenings were chosen based on risk significance of samples from the different cornerstones.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the licensee's condition reports concerning 10 CFR 50.59 safety evaluations and screenings to verify that the licensee had an appropriate threshold for identifying issues. The inspectors evaluated the effectiveness of the corrective actions for the identified issues.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed a partial walkdown of various Division 1 emergency core cooling systems (ECCS) to verify operability and proper equipment lineup while the No. 12 residual heat removal (RHR) pump was disabled for planned maintenance. The various components were selected due their increased risk significance associated with the No. 12 RHR pump being out-of-service for maintenance. 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)

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 2-C, Reactor Building (West HCU [Hydraulic Control Unit] Area) - Elevation 935'
- Fire Zone 2-B, East HCU Area
- Fire Zone 2-A, Tip Drive Area
- Fire Zone 1-A, 12 RHR & Core Spray Pump Room
- Fire Zone 1-F, Torus Area - Elevation 896' and 923'
- Fire Zone 1-G, CRD [Control Rod Drive] Pump Room - Elevation 921'
- Fire Zone 1-B, 11 RHR & Core Spray Pump Room
- Fire Zone 3-29, Security Diesel Building
- Fire Zone 3-34, East Electrical Equipment Room and No. 13 Diesel

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.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed a training crew during an evaluated simulator scenario and reviewed licensed operator performance in mitigating the consequences of events. The scenario included high vibration on 11 Condensate pump and resultant condensate pump trip, a failure within a 4160 Vac circuit breaker causing a loss of related electrical switchgear, a loss of offsite power resulting in a station blackout, and a break in a reactor pressure vessel (RPV) level instrument reference line causing a loss of RPV level indication and increasing drywell pressure. The transient resulted in the entry into the RPV flooding emergency operating procedure. 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.

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 selected systems. The following systems were selected based on being designated as risk significant under the Maintenance Rule, or being in the increased monitoring (Maintenance Rule category a(1)) group:

- Annunciators
- Alternate Nitrogen System
- 480 Vac System
- 4160 Vac System
- Primary Containment System
- 24 Vdc System

- 125 Vdc System

- 250 Vdc 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 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.

- Weekly Scheduling and Planning Meetings

- Outage Planning and Emergent Work Review

- Repair Oil Leaks on 10TR Transformer

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 the operations department and emergency preparedness department personnel response and associated 8-hour notification that was made for a loss of off-site response capability when the weekly evacuation siren signal test indicated that sirens were not receiving an actuation signal and were inoperable. Inspectors observed the emergency communicator make notifications and reviewed applicable documentation and proposed corrective actions for the event.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of the following operability evaluations to determine the impact onTechnical Specifications (TS), the significance of the evaluations, and to ensure that adequate justifications were documented.

- Reactor Recirculation Pump Interface with Process Computer Caused Scoop Tube Lock and Resulted In Degraded Ability to Control Reactor Power

- 4160 Vac Circuit Breakers Potentially Degraded Due to Maintenance Practices at Chestnut Service Center That Were Identified During Prairie Island Breaker Fire Review

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

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

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.

- Residual heat Removal Pump 14 Breaker Maintenance

- Reactor Manual Control System Select Relay Card and Pushbutton Replacement

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, TS 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, TS, and USAR design requirements.

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.

- Core Spray Header Differential Pressure Test and Calibration
- Residual Heat Removal Pump and Valve Test
- Core Spray 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, 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.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

Cornerstones: Mitigating Systems, Barrier Integrity, and Initiating Events

4OA3 Event Follow-up (71153)

.1 (Closed) LER 50-263/2001-06; URI 50-263/01-04-02: Alternate shutdown system design deficiencies result in vulnerability to hot shorts during postulated control room or cable spreading room fire. This issue was determined to be of very low safety significance (Green).

During an engineering review, the licensee identified inadequate electrical separation/isolation between the control room/cable spreading room and the alternate shutdown panel . There were two hot short vulnerabilities in the alternate shutdown system for a postulated fire in either the control room or cable spreading room in the event of loss of offsite power. The first vulnerability affected the alternate shutdown system when powering Bus 16 from the 12 emergency diesel generator (EDG) and could lead to the inability to operate the 12 RHR, 12 core spray and 12 RHR service water (RHRSW) pumps from the alternate shutdown panel. The second vulnerability could potentially cause reclosing of certain breakers, therefore overloading the EDG or 1AR transformers. These vulnerabilities had the potential to impair the licensee's ability to achieve and maintain safe shutdown conditions (maintain reactor vessel water levels and initiate suppression pool cooling). These vulnerabilities existed since the alternate shutdown panel was installed in 1986.

Since the issue involved degradation of the defense-in-depth elements (protection of components important to safety), the inspectors evaluated the issue using NRC Manual Chapter (MC) 0609, "Appendix F, Fire Protection Significance Determination Process." Using Phase 1 of the significance determination process (SDP), the issue screened out because it did not affect a fixed fire suppression system, a fire barrier forming the area boundary interface with recovery area, detection, or fire brigade effectiveness. However, the inspectors and regional senior reactor analysts (SRAs) determined that the issue should be further evaluated using Phase 2 of the SDP for the potential additive effect of core damage sequences involving external initiating events that could increase the total change in core damage frequency to greater than 1E-6 per year (MC 0609, Appendix A).

<u>Control Room</u>: The inspectors toured the control room and made the following observations which were used in developing a realistic fire scenario. The circuits of concern are located in Division 2 cabinet of control panel C-08. Panel C-08 contained two separate electrical cabinets, Divisions 1 and 2 cabinets. Therefore, there were two metal walls separating the redundant divisions. Although the cabinets are open at the back, the amount of combustibles within Division 2 cabinet was considered too small to sustain a fire that could propagate to Division 1 cabinet. In addition, the control room is continually manned by operators who would be able to initiate immediate manual fire fighting activities. Therefore, the realistic fire scenario is a small, localized electrical cabinet fire within the Division 2 cabinet which could not spread to the Division 1 electrical cabinet.

The Division 2 electrical cabinet was measured to be about 2.5 percent of the entire control room area. The ignition frequency for this fire scenario was determined to be 2.45E-4 per year (total 9.8E-3 per year multiplied by 2.5 percent). There were no fire barriers installed between redundant trains of safe shutdown equipment and no automatic suppression systems within the control room. The manual fire fighting effectiveness was assumed to be in the normal operating state. Since the fire within the Division 2 cabinet would not spread to the Division 1 cabinet, offsite power would still be available to power other equipment to achieve and maintain safe shutdown condition.

Based on the above information, the inspectors determined the issue, through the use of Phase 2 of the SDP, to be of very low safety significance (Green).

Cable Spreading Room: The cable spreading room had four rows of low voltage electrical cabinets (120 Vac or 125 Vdc) with cable trays traversing the top of the cabinets. The circuits of concern are located at one end of the room and at the end of the fourth cabinet with no other ignition source in the vicinity. For the realistic fire scenario, the inspectors assumed that the fire would start either from the fourth row of electrical cabinets or from transient combustibles located near the circuits of concern. The fire was assumed to damage that portion (25 percent) of the room and not further propagate to other parts of the room due to extinguishment of the fire due to automatic actuation of the halon system.

The fire ignition frequency was calculated to be 25 percent of the total frequency of 4.4E-3 per year (taken from Monticello Individual Plant Evaluation for External Events [IPEEE] data). There were no fire barriers installed between the redundant trains of safe shutdown equipment. The halon system, which is an automatic fire suppression system and the manual fire fighting effectiveness were assumed to be in the normal operating state. Because automatic fire suppression and manual fire fighting effectiveness both depended on a common water delivery and supply system, an adjustment was made to the fire mitigation frequency.

A fire in that portion of the room would potentially affect the availability of offsite power such that the hot short vulnerabilities described above may occur. However, the circuits associated with high pressure core injection (HPCI) and reactor core isolation cooling (control room panel C-03) are located on the opposite side of the cable spreading room and would not be affected. Therefore, operation of these systems could mitigate the loss of the 12 core spray pump for maintaining reactor levels. However, since the systems would repeatedly start and stop, the inspectors assumed a failure probability of 0.1 for the combined HPCI and reactor core isolation cooling (RCIC) systems. Once the operators recognized the load shed problem with Bus 16, the load shed signals could be easily bypassed at the breakers for RHR and RHRSW pumps, which then could be loaded onto the bus. These two pumps are not needed immediately but would eventually be required to support safe shutdown functions. Therefore, there would be adequate time for the operators to diagnose the load shed problem. The actions to recognize and restore RHR and RHRSW pumps were assigned a failure probability of 0.1. Based on the above information, the inspectors determined that the issue, through the use of Phase 2 of the SDP, to be of very low safety significance (Green).

This issue is dispositioned in Section 4OA7 of this report. The LER and URI are closed.

.2 (Closed) URI 50-263/01-04-01: Flood Protection Deficiencies. As discussed in Inspection Report (IR) 50-263/2001-004, Section 1R06, "Flood Protection," the inspectors identified procedural deficiencies for external flood mitigation plans. The most significant concerns were: the lack of procedural guidance for protecting the EDG from hydraulic lifting due to high water level, and drawings that did not contain sufficient detail to ensure that steel erected to mitigate flooding at the intake structure was properly installed.

Because these deficiencies affected both essential service water (ESW) and EDGs, the inspectors concluded that this issue had a credible impact on safety. Additionally, the issue potentially impacted the operability of these systems during an external flooding event and the SDP was entered for initiating events. Upon conducting the Phase 1 SDP, the inspectors concluded that the finding was potentially risk significant in that it involved an external flooding scenario whereby the function of both trains of ESW and/or both EDGs could be affected. The inspectors consulted the plant's IPEEE and referred the issue to the regional senior reactor analyst (SRA) to perform a Phase 3 SDP in accordance with NRC MC 0609, Appendix A.

The SRA performed a Phase 3 analysis and identified that the likelihood of core damage from failing to seal the intake structure, or protect the EDG's during an external flood, was low, and was not significant in terms of overall risk to the public. The SRAs' determination was based on: the very low probability of the type of flooding event in question; the probability that the licensee would be unsuccessful in erecting a berm to protect plant equipment; the probability that the licensee would be unsuccessful in installing temporary barriers around vital equipment; the probability that the licensee would be unsuccessful in arranging alternative cooling options within 12 days from the onset of flooding; and the availability of long-term core cooling from alternate sources.

The licensee's review of the items identified in IR 50-263-01-04, Section 1R06, indicated that drawing corrections and procedural modifications were necessary to correct deficiencies that impacted their ability to respond to an external flooding condition. The above assumptions resulted in an average annual frequency of 10^{-7} per year. Therefore, the issue was determined to be of very low risk significance (Green).

To protect emergency diesel generators from damage due to hydraulic lift created under the floor of the generating units, procedures required that the diesel generator room floors be sandbagged or a berm erected around the building task (Inspection Report 2001-04, Section 1R06, URI 50-263/01-04-01). It was determined that an accumulation of approximately ten inches of water between the berm and diesel generator buildings would be sufficient to cause sufficient lift for equipment failure. No procedural guidance was provided to limit the accumulation of water between an erected berm and the diesel generator building. The failure to establish procedural guidance to prevent the allowable accumulation of water allowed between the diesel generator building and an erected berm which would exceed the design hydraulic lift on the diesel generator room floor is considered a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" (NCV 50-263/01-08-01(DRP)).

4OA6 Meeting

Exit Meeting

The inspectors presented the inspection results to Mr. Fadel and other members of licensee management on September 28, 2001. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was properly controlled.

4OA7 Licensee-Identified Violation

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV. If you deny this NCV, you should provide a response with the basis of your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspectors at the Monticello facility.

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-263/01-08-02	10 CFR Part 50, Appendix R, Section III.L.7 required, in part, that the safe shutdown equipment and systems for each fire area shall be known to be isolated from associated non-safety circuits in the fire area so that hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment. Contrary to the above, the licensee failed to provide electrical isolations for the 12 EDG which could lead to inability to operate safe shutdown equipment. In addition, the licensee failed to provide electrical isolations for breakers not associated with safe shutdown equipment such that hot shorts could re-close these breakers and overload the electrical bus. This issue was entered into the licensee's corrective action program as Condition Report (CR) 20011046. This is being treated as a Non-Cited Violation.

KEY POINTS OF CONTACT

Licensee

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

NRC

B. Burgess, Chief, Reactor Projects Branch 2

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-263/01-08-01	NCV	Failure to establish procedural guidance to prevent the allowable accumulation of water allowed between the diesel generator building and an erected berm (Section 4OA3.2)
50-263/01-08-02	NCV	Hot shorts, open circuits, or shorts to ground in the associated circuits may prevent operation of safe shutdown equipment (Section 4OA7)

Closed

50-263/01-08-01	NCV	Failure to establish procedural guidance to prevent the allowable accumulation of water allowed between the diesel generator building and an erected berm (Section 4OA3.2)
50-263/01-08-02	NCV	Hot shorts, open circuits, or shorts to ground in the associated circuits may prevent operation of safe shutdown equipment (Section 4OA7)
50-263/01-04-01	URI	Flood protection potential deficiencies (Section 4OA3.2)
50-263/01-04-02	URI	Alternate shutdown system modifications to resolve hot short issues (Section 4OA3.1)

50-263/2001-06	LER	Alternate shutdown system design deficiencies result in vulnerability to hot shorts during postulated control room or cable spreading room fire (Section 4OA3.1)

Discussed

None

LIST OF ACRONYMS USED

AC	Alternating Current
ASME	American Society of Mechanical Engineers
AWI	Administrative Work Instruction
CFR	Code of Federal Requirements
CR	Condition Report
CRD	Control Rod Drive
DBD	Design Basis Document
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling Systems
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESW	Essential Service Water
HCU	Hydraulic Control Unit
HPCI	High Pressure Core Injection
HVAC	Heating, Ventilation and Air Conditioning
IPEEE	Individual Plant Examination - External Events
IR	Inspection Report
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MC	Manual Chapter
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NUMARC	Nuclear Management and Resources Council
PRV	Pressurizer Relief Valve
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPV	Reactor Pressure Vessel
SBGT	Standby Gas Treatment
SCR	10 CFR 50.59 Screening
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SRI	10 CFR 50.59 Safety Review Item
SRV	Safety Relief Valve
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current

LIST OF DOCUMENTS REVIEWED

1R02 Evaluations of Changes, Tests, or Experiments

Number	Title	Revision/Date
4AWI-05.06.01	Safety Review Item	8
4AWI-05.06.02	10 CFR 50.59 Applicability and Screening	4
4AWI-05.06.03	10 CFR 50.59 Evaluations	2
CR 20010614	Initiation of Torus Cooling for Small Break LOCA Is Not Consistent with Design Basis Event Assumptions	0
CR 20015068	10 CFR 50.59 Screening $01-244 Incorrectly Concluded That No 10 CFR 50.59 Evaluation Was Needed	0
CR 20015069	Evaluation for Instrument Inaccuracy When Establishing Maximum River and Torus Temperature Limits	0
SCR 01-0003	Revision to Operations Manual B.09.06-05	0
SCR 01-0017	Volume F Memo for Core Spray Venting	0
SCR 01-0061	Replacement of Pressure Gauge for RHR Minimum Flow Valve	0
SCR 01-0093	Additional Guidance to Tripping the Field Breaker on the Main Generator	0
SCR 01-0103	Corrosion Allowance for MSIVs	0
SCR 01-0106	Drywell Floor Drain Sump Operation	0
SCR 01-0128	Verification of Dc Voltage on the Local Test Panel	0
SCR 01-0141	SRV Bellow Leak Detection System	0
SCR 01-0244	High River Water Temperature	0
SCR 01-0197	Declared All Core Spray and RHR Pumps Inoperable When Drywell Temperature Exceeded 135°F	0
SCR 01-0173	Starting Both RHRSW Pumps with Normal Offsite Power Available	0
SCR 01-0222	Replacing Instrument Isolation Valves for PT-4067A, B, C, and D	0

SRI 99017	Time After the Initiating Event Before Taking Credit for Operator Actions	0
SRI 99019	Preventing Restart of Secondary Containment HVAC When SBGT Is Available	0
SRI 00021	PRV Blowdown Initiated at Difference Torus Water Temperatures	0
DC 01Q080	Suppression Chamber to Drywell Vacuum Breakers Counter Balance Weight Repositioning	0
DC 00Q250	EOP ECCS Drywell Cooling Trip Bypass Switch	0
DC 99Q190	HPCI and RCIC Exhaust Vacuum Breaker Valve Replacement	0
DC 00R100	Eliminate Manual Valving Prior to Start of RHRSW Pump	0
DC 01Q075	Fuel Zone Level Instrumentation Reference Leg Modification	0
Form 3278	NMC Standard 10 CFR 50.59 Screening Form	0
Form 3280	Regulatory Process Applicability Determination	0
Quality Assurance Evaluations 2001-05-124 50.59	NMC Oversight Assessment Observation Report on 10 CFR 50.59 Evaluations	8/28/01
Volume F Memo 1986	High River Water Temperature	0
GE-NE-T2300731-2	Final Report LOCA Containment Analyses For Use in Evaluation of NPSH for the RHR and Core Spray Pumps	6/16/97
DRF T23-00789-00	Monticello Nuclear Power Station - Response to NMC Question Regarding Maximum Time to Vessel Reflood and Initiation of Containment Cooling	3/25/01

1R04 Equipment Alignment

4AWI-08.15.01	Risk Management For Outage and On-Line Activities	Revision 0

	Drawings:	
M-123	- High Pressure Coolant Injection (Steam Side)	Revision AF
M-124	- High Pressure Coolant Injection (Water Side)	Revision Y
M-112	- RHR & Emergency Service Water	Revision BF
M-121, Sheet 2	- RHR (Division 1)	Revision BK
M-811, Sheet 1	- Service Water & Makeup Intake Structure	Revision CD
	Operations Manual:	
B.3.1	- Core Spray System	
B.3.2	- High Pressure Coolant Injection System	
B.3.4	- Residual Heat Removal System	

1R05 Fire Protection

NX-16991	Technical Manual, Monticello Updated Fire Hazards Analysis	
	Monticello Fire Strategies:	
A.3-02-C	- Reactor Building (West HCU Area) - El. 935'	Revision 4
A.3-02-B	- East HCU Area	Revision 5
A.3-02-A	- Tip Drive Area	Revision 2*
A.3-01-A	- 12 RHR & Core Spray Pump Room	Revision 2*
A.3-01-F	- Torus Area - Elevation 896' and 923'	Revision 4
A.3-01-G	- CRD Pump Room - Elevation 921'	Revision 2*
A.3-01-B	- 11 RHR & Core Spray Pump Room	Revision 2*
A.3-29	- Security Diesel Building	Revision 3*
A.3-34	- East Electrical Equipment Room and # 13 Diesel	Revision 5
	Procedures and Administrative Work Instructions (AWIs):	
4AWI-08.01.01	- Fire Prevention Practices	Revision 16
4AWI-08.01.02	- Combustion Source Use Permit	Revision 6
0271	- Fire Hose Station and Yard Hydrant Hose House Equipment Inspection	Revision 16
0275-2	- Fire Barrier Wall, Damper, and Floor Inspection	Revision 9
0274	- Fire Hose Hydrostatic Test Interior Hose Stations	Revision 22
0275-1	- Fire Barrier Penetration Seal Visual Inspection	
0275-3	- Fire Door Inspections	
QUAD-5-80-009	Quadrex Corporation Report, Specifications for Installation of Electrical and Mechanical Penetration Seals at the Monticello Nuclear Generating Plant	Revision 7

	Operations Manual:	
B.8.5	- Fire Protection	
B.8.12.2	- Security Buildings and Receiving Warehouse Fire Protection	

1R11 Licensed Operator Requalification Program

RQ-SS-24E	Station Blackout with Loss of All Level Indication	Revision 0
C.5-2006	RPV Flooding	Revision 8
C.5-2002	Emergency RPV Depressurization	Revision 4
C.5-1100	RPV Control	Revision 7
C.5-1200	Primary Containment Control	Revision 10
C.4-A	Reactor Scram	Revision 18

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
05.02.01	Monticello Maintenance Rule Program Document	Revision 5
	Monticello Maintenance Rule Periodic Assessment Report	1st Quarter - 2001

	Operations Manual:	
B.5.13	- Annunciators	
B.8.4.3	- Alternate Nitrogen System	
B.9.6	- 4.16 kV Station Auxiliaries	
B.9.7	- 480 Vac Station Auxiliaries	
B.6.4	- Circulating Water System	
B.4.1	- Primary Containment System	
B.9.9	- 250 Vdc Battery System	
B.9.10	- 125 Vdc Battery System	
B.9.11	- 24 Vdc Battery System	
	Maintenance Rule Program System Basis Document:	
B.5.13	- Annunciators	Revision 2
B.8.4.3	- Alternate Nitrogen System	Revision 1
B.9.6	- 4.16 kV Station Auxiliaries	Revision 3
B.9.7	- 480 Vac Station Auxiliaries	Revision 2
B.6.4	- Circulating Water System	Revision 0
B.4.1	- Primary Containment System	Revision 2
B.9.9	- 250 Vdc Battery System	Revision 1
B.9.10	- 125 Vdc Battery System	Revision 1
B.9.11	- 24 Vdc Battery System	Revision 2
	System Performance Data Sheet for ANN [Annunciators]	
CR 19982884	Annunciator Card C-05-B-4 Failed and Cannot be Removed From Panel	
CR 20002536	Annunciator Card C-06-B-1 Failed Causing Small Flame On The Annunciator Card. Promptly Extinguished By Operator	
CR 20000606	Train ‘B’ Alternate N2 Check Valves Leakage Greater Than Acceptance Limit	
CR 20000628	Train A System Alternate N2 Leakage Greater Than Acceptance Limit	
CR 20000489	152-405 Overcurrent Alarm Relay Exceeded As-Found Calibration Criteria	
CR 20000601	1R Transformer Capability With One of 20 Cooling Fans Out of Service	
CR 20011590	X8 Transformer Developed an Oil Leak With an Initial Estimate of 10 Gallons Spilled into the Cement Dike Area	

CR 20003907	During Bus Transfer from 2R to 1R Panel Meters Indication Did Not Respond to Paralleled Transformers as Expected
CR 20011918	Cable Separation Concern for LC 103 Normal and Standby Control Power
CR 20004712	Failure of Breaker LCB-014 in Cubicle 52-204 to Close on First Attempt to Start No. 13 Service Water Pump
CR 20004304	Breaker 2138 Found in Tripped Condition (1R Transformer Auxiliary Power)
CR 20002876	Breaker Cable Strands Found in B2215 During Performance of Preventative Maintenance
CR 20001651	No. 12 Service Water Pump Breaker 52-405 Failed to Close When Pump Was Given a Start Signal After Breaker Was Racked In
CR 20011536	Procedure 0127 Doesn't Verify the Open Position Switches are Adjusted as Described in the Basis for Technical Specification 3.7
CR 20011427	Acceptance Criteria in Procedure 0127 Does not Ensure that Torus to Drywell Vacuum Breakers Will Meet Analysis Assumptions
CR 20011173	Torus to Drywell Vacuum Breaker AO-2382K Failed to Open Via Test Switch During Containment Purging
CR 20011113	Torus Found Pressurized During 8021 Procedure Initiated by Work Orders 0003805 and 0003819
CR 19991532	Inoperable Torus to Drywell Vacuum Breaker Considered to be a Maintenance Rule Functional Failure
CR 20000133	Drywell to Torus Vacuum Breaker Exceeded 354 Inch-Pounds During Performance of Procedure 0127
CR 20011316	Unplanned LCO Due to Low Specific Gravity on 24 Volt Battery No. 15 Resulting in Battery Being Declared Inoperable
CR 20004003	24 Volt Battery No. 14, Cell 18, and 24 Volt Battery No. 15, Cell 8, Show Signs of Post Positive Buckling

CR 20000721	Noticed Changes in IRM/SRM/PRM Caused by No. 14 Battery Voltage Dropping to 16 Volts Due to Charger D23 Shutdown	
CR 20014380	Some Battery Cell Temperatures Found Out of Acceptance Band (high) During Performance of 0196 Quarterly Battery Check	
CR 20010500	D40 Swing Battery Charger for 125 Vdc System Was Removed From Service Due to Erratic Charging Current	
CR 20003368	Routine Battery Inspections Identified Potential Internal Corrosion of the (+) Terminals for Six Cells in No. 11/12 Battery	
CR 20012501	Entered Unplanned LCOs for Primary Containment Integrity and RCIC Due to Failure of Control Power to RCIC MCC-311	
CR 20000199	No. 16 250 Vdc Battery Capacity Test 0197-2 Problems	
CR 20003198	250 Vdc Division 1 Battery Ground Indication. Indication Out of Specification High.	
CR 20013506	Failure of HPCI Undervoltage Alarm Results in Unplanned LCO Entry and NRC 50.72 8-Hour Notification	
MNGP Table 1	GAP Closure Activities Requiring Additional Resource	August 2001
	USAR:	Revision 18
10.3.4	- Plant Air and Nitrogen Systems	
5.2	- Primary Containment System	
8.5	- Direct Current Power Supply Systems	
	Drawings:	
M-131, Sheet 10	- Alternate Nitrogen Supply System	Revision M
NF-36298-2	- Direct Current Electric Load Distribution	Revision A
NF-36298-1	- Electric Load Flow	Revision M
WO 0000676	Seat Leakage is excessive per 0255-17-ID-1 (AI-713)	
WO 0000677	Seat Leakage is excessive per 0255-17-ID-1 (AI-714)	
WO 0000813	Battery Charger D-53 High Voltage Shutdown Problem	

WO 0105916	Battery Charger D-40 Not Functioning Properly	
	Monticello Maintenance Rule Program Boundary Definition Guidance Document	Revision 1
CA-94-017	Calculation of Alternate Nitrogen System Operability Leakage Criteria	

1R13 Maintenance Risk Assessments and Emergent Work Control

	Procedures:	
4AWI-04.01.01	- "General Plant Operating Activities"	Revision 28
SWI-14.01	- "Risk Management of On-line Maintenance"	Revision 0
WO 0108960	Repair Leaks On 10TR Transformer	
XCEL 091101-1	Monticello TR#10 Outage Operating Guide	September 11, 2001

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

CR 20015132	Wright County Sirens Found Inoperable When Performing Public Alert Notification System Weekly Cancel Signal Test 1359	

1R15 Operability Evaluations

CR 20013563	Accidental Trip of Computer Power Distribution System Results in Recirc. Scoop Tube Locks and Loss of 3D Monicore	
CR 20015134	PI Breaker 12-4 Root Cause Report	

1R19 Post-Maintenance Testing

WO 0108118	Swap Breakers Serving 152-603, 14 RHR Pump 4kV Supply	
3069	PMT for WO 0108118	
3069	PMT for WO 0005221	
WO 0005221	Reactor Manual Control Select Pushbutton Relay Card Replacement	
NX-7866-74-11	Reactor Manual Control System	Revision D
EWD 729E823 Sh-7 of 10	Reactor Manual Control System	Revision 4

WO 005243	Reactor Manual Cont5orl Rod Select Electrical Push Button Wipe	

1R22 Surveillance Testing

0098	Core Spray Header Differential Pressure Test and Calibration	Revision 11
0255-04-IA-1	RHR Pump and valve Test	Revision 52
	Technical Specifications:	
TS 3.5.B	- RHR Intertie Return Line Isolation Valves	Amendment 79
TS 4.15.B	- Inservice Testing	Amendment 104
TS 4.5.A.2	- Core and Containment Spray / Cooling Systems	Amendment 122
0255-03-IA-1	Core Spray System Tests	Revision 31
CA 90-007	Core Spray Surveillance Test Acceptance Criteria	May 29, 1990
6.2.2	FSAR - Core Spray System	
NH-36248	Core Spray System	Revision AH

4OA3 Event Follow-up

	Individual Plant Examination of External Events (IPEEE)	
Procedure A.6, Section B	Operations Manual, "External Flooding"	Revision 11
DBD T.5	Design Basis Document "External Flooding"	
	Updated Safety Analysis Report (USAR):	Revision 18
Section 12.2.1.7.1.		
Section 2.4.1	External Flooding	
Section 1.3.1.4	Surface Water	
	Hydrology	
FOI 91-0126	Emergency Procedure (Flooding) Concerns-A.6, Section B	
FOI 91-0125	Unlocated External Flood Study Documents	
FOI 91-0073	Predicted Delivery Time for Sandbags	