UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

May 14, 2009

Mr. Timothy J. O'Connor
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company, Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT
NRC INTEGRATED INSPECTION REPORT 05000263/2009002

Dear Mr. O'Connor:

On March 31, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Monticello Nuclear Generating Plant. The enclosed report documents the inspection findings, which were discussed on April 1, 2009, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one NRC-identified and one self-revealed finding, both of very low safety significance, were identified. Each finding involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating these issues as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at Monticello Nuclear Generating Plant. The information you provide will be considered in accordance with Inspection Manual Chapter 0305. There were also two licensee-identified violations, which are listed in Section 4OA7 of this report.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Monticello Nuclear Generating Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Monticello Nuclear Generating Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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/***

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000263/2009002
w/Attachment: Supplemental Information

cc w/encl: D. Koehl, Chief Nuclear Officer
Manager, Nuclear Safety Assessment
P. Glass, Assistant General Counsel
Nuclear Asset Manager, Xcel Energy, Inc.
J. Stine, State Liaison Officer, Minnesota Department of Health
Commissioner, Minnesota Pollution Control Agency
R. Hiivala, Auditor/Treasurer,
Wright County Government Center
Commissioner, Minnesota Department of Commerce
Manager - Environmental Protection Division
Minnesota Attorney General's Office

a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Monticello Nuclear Generating Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Monticello Nuclear Generating Plant.

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Letter to T. O'Connor from K. Riemer dated May 14, 2009

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT
NRC INTEGRATED INSPECTION REPORT05000263/2009002

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REGION III

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|---|---|
| Docket No: | 50-263 |
| License No: | DPR-22 |
| Report No: | 05000263/2009002 |
| Licensee: | Northern States Power Company, Minnesota |
| Facility: | Monticello Nuclear Generating Plant |
| Location: | Monticello, MN |
| Dates: | January 1 through March 31, 2009 |
| Inspectors: | S. Thomas, Senior Resident Inspector<br>L. Haeg, Resident Inspector<br>K. Stoedter, Prairie Island Senior Resident Inspector<br>T. Bilik, Reactor Inspector<br>V. Meghani, Reactor Inspector |
| Approved by: | Kenneth Riemer, Chief<br>Branch 2<br>Division of Reactor Projects |

Enclosure

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## SUMMARY OF FINDINGS

IR 05000263/2009002; 01/01/2009 – 03/31/2009; Monticello Nuclear Generating Plant; Operability Evaluations; Surveillance Testing.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green findings were identified by the inspectors. Both findings were considered to be non-cited violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

### A. NRC-Identified and Self-Revealed Findings

**Cornerstone: Mitigating System**

- Green. The inspectors identified a finding of very low significance and NCV of 10 CFR 50, Appendix B, Criterion V, for the licensee approving the use of non-quality documents in the absence of approved procedures to modify the normal operation of intermediate range monitor (IRM) 11 and IRM 18. Implementation of this guidance would have resulted in a condition that would not have been in compliance with Technical Specifications. Once identified, the licensee took immediate corrective actions to correct the situation and entered the issue into their corrective action program. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having decision making components, and involving aspects associated with using conservative assumptions in decision making and adopting requirements to demonstrate that proposed actions are safe in order to proceed, rather than a requirement to demonstrate that it is unsafe in order to disapprove actions. [H.1(b)]

  The inspectors determined that the finding was more than minor because it impacted the Reactor Safety Mitigating System Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using IIMC 0609, Appendix A, Attachment 1, "Significance Determination of Reactor Inspection Findings for At-Power Situations," using the Phase 1 Worksheet for the Mitigating Systems Cornerstone. After answering 'No' to all five questions in the Mitigating Systems Cornerstone column of Table 4a, "Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones," the inspectors concluded that the finding was of very low safety significance. (Section 1R15)

- Green. A finding of very low safety significance and NCV of 10 CFR 50, Appendix B, Criterion V, was self-revealed when the performance of an inadequately prepared temporary procedure change resulted in the inadvertent repositioning of a reactor core isolation cooling (RCIC) containment isolation valve during the system restoration section of a RCIC surveillance test. The licensee took immediate corrective actions to identify the cause of the inadvertent valve actuation and to restore the valve to its proper position. The licensee entered this issue into their corrective action program. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having work practices components, and involving aspects associated with appropriately coordinating work activities by incorporating actions to ensure that supervisory and management oversight of work activities is sufficient to support nuclear safety. [H.4(c)]

The inspectors determined that the failure to adequately evaluate the impact of making temporary changes to existing plant procedures used to conduct Technical Specification surveillance testing was a performance deficiency warranting significance evaluation in accordance with IMC 0612, Appendix B, “Issue Disposition Screening.” The inspectors determined that the finding was more than minor because, if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern. The inspectors evaluated the finding using IMC 0609, Appendix A, Attachment 1, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” using the Phase 1 Worksheet for the Mitigating Systems Cornerstone. After answering ‘No’ to all five questions in the Mitigating Systems Cornerstone column of Table 4a, “Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones,” the inspectors concluded that the finding was of very low safety significance. (Section 1R22)

**B. <u>Licensee-Identified Violations</u>**

Violations of very low safety significance that were identified by the licensee have been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee’s corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

## REPORT DETAILS

### Summary of Plant Status

Monticello operated at full power for most of the assessment period except for brief downpower maneuvers to accomplish rod pattern adjustments and to conduct planned surveillance testing activities with the following exceptions:

- January 1 to January 7, 2009; reactor power was reduced to approximately 37 percent to establish the requisite plant conditions required to repair a leak located on the 12 reactor feedwater pump recirculation line.
- March 13, 2009; the licensee began reducing reactor power in preparation for reactor plant shutdown and commencement of Refueling Outage 24. At 00:06 on March 14, 2009, the main generator was taken off-line and turbine shutdown was commenced. The plant remained shutdown for the remainder of the inspection period.

## 1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition – Extreme Cold Conditions

a. Inspection Scope

Since extreme cold conditions were forecast in the vicinity of the plant for the week beginning January 11, 2009, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. On January 13, 2009, the inspectors walked down areas housing the 11 emergency diesel generator (EDG); 12 EDG; 13 diesel generator (DG); security DG; diesel fire pump; all safety-related service water pumps; station heating boiler; and temporary heating boiler because the selected system safety or risk-significant components could be affected or required as a result of the extreme cold conditions forecast for the plant. The inspectors observed insulation; heat trace circuits; space heater operation; and weatherized enclosures to ensure operability, functionality, or availability of the affected systems. In addition to monitoring the condition of the specific components discussed previously, the inspectors performed a general walkdown of areas in the reactor building; turbine building; and selected out-plant buildings. During this walkdown, the inspectors focused on monitoring the temperature of plant areas which had exterior walls; ventilation dampers which communicated directly with the outside; or fluid piping which entered/exited the areas. The inspectors reviewed selected procedures which addressed abnormal weather-related problems. Specific documents reviewed during this inspection are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Division I 125 V DC during a planned 12 EDG outage;
- 12 reactor building closed cooling water (RBCCW) system with 11 RBCCW out-of-service (OOS) for planned maintenance (PM);
- Division II residual heat removal service water (RHRSW) during planned Division I residual heat removal (RHR) maintenance; and
- Division II 125/250 V DC during planned high pressure coolant injection (HPCI) maintenance.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and; therefore, potentially increase risk. The inspectors reviewed applicable operating procedures; system diagrams; Updated Safety Analysis Report (USAR); Technical Specification (TS) requirements; outstanding work orders (WOs); condition reports; and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted four partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On January 13, 2009, the inspectors performed a complete system alignment inspection of the main feedwater system and auxiliaries to verify the functional capability of the system. This system was selected because misalignment, or equipment or operational issues could cause a plant transient or scram. The main feedwater system is also used in certain emergency operating procedures as a preferred water makeup source to the reactor. The inspectors reviewed Operating Experience Smart Sample: OpESS FY 2009-02, "Negative Trend and Recurring Events Involving Feedwater Systems" in preparation for this inspection. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

These activities constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability; accessibility; and the condition of firefighting equipment in the following risk-significant plant areas:

- Fire Zone 24 (diesel fire pump room);
- Fire Zone 8 (cable spreading room);
- Fire Zones 14A, 14B and 14C (upper 4kV bus area, isophase bus area and turbine building rail car area);
- Fire Zones 21A, 21B, 21C, and 21D (accessible areas of the radwaste building); and
- Fire Zones 16 and 17 (turbine building east/west corridor, elevations 911', 931' and turbine building north cable corridor 941').

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. The inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted five quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the USAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the CAP to verify the adequacy of the corrective actions. Documents reviewed are listed in the Attachment to this report. The inspectors performed a walkdown of the following plant area to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- torus room flood area.

This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of 11 and 12 RHR heat exchangers to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance; to identify any common cause issues that had the potential to increase risk; and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria; the correlation of scheduled testing and the frequency of testing; and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions; design conditions; and testing conditions. Documents reviewed are listed in the Attachment to this report.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07-05.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08G)

From March 23, 2009, through April 1, 2009, the inspectors conducted a review of the implementation of the licensee's inservice inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS), risk-significant piping and components, and containment systems.

The inspections described in Sections 1R08.1 and 1R08.2 below constitute one inspection sample as defined by IP 71111.08-05.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors observed the following nondestructive examinations (NDE) mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- ultrasonic examination (UT) of a core spray 8" pipe-to-penetration weld (weld W-3);
- UT of main steam C 18" pipe-to-elbow weld (weld W-25); and
- magnetic particle (MT) examination of RHR 'A', 24" shell-pad-nozzle weld (weld N-2).

During the prior outage non-destructive surface and volumetric examinations, the licensee did not identify any relevant/recordable indications. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed the following pressure boundary welds completed for risk-significant systems since the beginning of the last Refueling Outage (RFO) to verify that the welding and any associated non-destructive examinations were performed in accordance with the Construction Code and ASME Code, Section XI.

- weld repair/replacement of Class 1 main steam line drain outboard valve (valve MO-2374); and
- weld repair/replacement of Class 1 main steam line drain inboard valve (valve MO-2373).

The inspectors also reviewed the welding procedure specification and supporting weld procedure qualification records for the above, to determine if the welding procedures were qualified in accordance with the requirements of the Construction Code and the ASME Code Section IX.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI related issues entered into the licensee’s CAP and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI related issues;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience (OE) and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report. In addition, the inspectors verified that the licensee correctly assessed operating experience for applicability to the ISI group.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On February 9, 2009, the inspectors observed a crew of operators in the plant's simulator during a simulator scenario that included a station blackout and small reactor coolant leak event. During this scenario, the inspectors verified that operator performance was adequate; evaluators were identifying and documenting crew performance problems; and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

This inspection constituted one quarterly sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- 4.16 kV station auxiliary;
- 125 V DC system; and
- feedwater heater and moisture separator drain tank dump and drain valve level transmitters.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted three quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings of significance were identified.

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

### .1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- electrically backseating MO-2076 to reduce/minimize a significant packing leak;
- troubleshooting associated with relays 16A-K60A and 16A-K60C; and
- troubleshooting of MO-2020, Division I RHR drywell spray outboard valve handswitch following modification to allow throttling of valve.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work; discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor; and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems; when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Documents reviewed are listed in the Attachment to this report.

These maintenance risk assessments and emergent work control activities constituted three samples as defined in IP 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- OPR 01164976-01 (ability of MO-2076 to seal and maintain containment leakage within allowable limits);
- OPR 01113447-01 (pin hole leak above valve SW-30-2 in the service water line piping to the stator winding liquid cooler);
- OPR 01169229-01 (two of three phases on cable from B307 to MCC-133B);
- OPR 01170377-01 (AO-2-86B, ‘B’ outboard main steam isolation valve (MSIV));
- OPR 01173169-01 (bypass leakage for ‘B’ core spray system is greater than assumed); and
- OPR 01173014-01 (intermediate range monitor (IRM) 11 and IRM 18 could not be calibrated).

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and USAR to the licensee’s evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in-place would function as intended and were properly controlled. The inspectors determined; where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted six samples as defined in IP 71111.15-05

b. Findings

Introduction

The inspectors identified a finding of very low significance and non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, for the licensee approving the use of non-quality documents in the absence of approved procedures to modify the normal operation of

IRM 11 and IRM 18. Implementation of this guidance would have resulted in a condition that would not have been in compliance with TSs.

Description

On March 13, 2009, the licensee began to reduce reactor power in preparation for shutdown of the unit for a refueling outage. Prior to entering Mode 2, the licensee performed Procedure 0018, “IRM Heat Balance,” Revision 19. Function 1.a of TS 3.3.1.1 requires that the IRM flux scram setting shall occur before reaching 20 percent rated thermal power on the highest IRM range. After completing the surveillance, IRM 11 and IRM 18 could not be calibrated in accordance with the testing methodology described in the surveillance procedure and were determined to be inoperable. The control room operators took the appropriate actions, as required by TSs, for the two inoperable IRMs and continued with the reactor shutdown.

Subsequent to the plant shutdown, the inspectors became aware of an Operability Recommendation associated with IRM 11 and IRM 18 that had been accepted by Operations. The purpose of the Operability Recommendation was to justify the operability of IRM 11 and/or IRM 18, should an additional IRM become inoperable during the RFO. During the review of the Operability Recommendation, the inspectors identified the following deficiencies:

- Even though IRM 11 and IRM 18 had failed one of the surveillance procedures required by TSs to verify IRM operability, the Operability Recommendation allowed them to be declared operable-but-degraded without any additional testing;
- The Operability Recommendation placed conditions on the operability of IRM 11 and IRM 18; and
- The Operability Recommendation was used to provide operational guidance for IRM 11 and IRM 18, which differed from existing approved operating procedures.

The inspectors had several discussions with licensee management and staff regarding what constitutes operability when evaluating equipment required by TSs. Subsequent to the discussions with the inspectors, the licensee pulled the Operability Recommendation and associated Operations Memo and control panel information tags.

Since no additional IRM failures occurred during the time that this Operability Recommendation was in place, and when IRM 11 and IRM 18 were inappropriately declared operable, the inspectors determined that the licensee remained in compliance with the requirements of their TSs associated with IRMs.

Analysis

The inspectors determined that approving the use of non-quality documents in the absence of approved procedures to modify the normal operation of safety-related equipment constituted a performance deficiency warranting significance evaluation in accordance with IMC 0612, Appendix B, “Issue Disposition Screening.” The inspectors determined that the finding was more than minor because it impacted the Reactor Safety Mitigating System Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having decision making components, and involving aspects associated with using conservative assumptions in decision making and adopting requirements to demonstrate that proposed actions are safe in order to proceed, rather than a requirement to demonstrate that it is unsafe in order to disapprove actions. [H.1(b)]

The inspectors evaluated the finding using IMC 0609, Appendix A, Attachment 1, "Significance Determination of Reactor Inspection Findings for At-Power Situations," using the Phase 1 Worksheet for the Mitigating Systems Cornerstone. After answering 'No' to all five questions in the Mitigating Systems Cornerstone column of Table 4a, "Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones," the inspectors concluded that the finding was of very low safety significance (Green).

Enforcement

Title 10 CFR, Part 50, Appendix B, Criterion V states, in part, that "activities affecting quality shall be accomplished in accordance with prescribed instructions, procedures, or drawings." Contrary to this requirement, the licensee approved the use of non-quality documents in the absence of approved procedures to modify the normal operation of IRM 11 and IRM 18. Implementation of this guidance would have resulted in a condition that would not have been in compliance with TSs. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program (CAP 1173190), it is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000263/2009002-01)

## 1R18 Plant Modifications (71111.18)

### .1 Temporary Plant Modifications

#### a. Inspection Scope

The inspectors reviewed the following temporary modification:

- EC 12607 (pipe repair downstream of CV-3490).

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis; the USAR; and the TS; as applicable, to verify that the modification did not affect the operability or availability of the affected systems. The inspectors; as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations; engineering; and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in-place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one temporary modification sample as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The following engineering design package was reviewed and selected aspects were discussed with engineering personnel:

- EC 10856 (power range neutron monitoring system).

This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing plans, and relevant planned changes to procedures, design, and licensing documents. The inspectors observed ongoing work activities to verify that installation was consistent with the design control documents. Documents reviewed are listed in the Attachment to this report. The modification upgraded the existing power range monitoring equipment to a digital design. This modification was required for extended power uprate (EPU) and use of a different operating domain, where an oscillating power range monitor (OPRM) system is required. The modification required extensive control room and out-plant changes to accommodate the updated system. Post-modification and operational testing is scheduled for April 2009, and will be reviewed under a separate inspection sample. Portions of the OPRM protective system will not be in effect until the power uprate amendment is approved and additional plant modifications are completed during the 2011 RFO.

This inspection constituted one permanent plant modification sample as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance (PM) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- 12 EDG air start system following PM;
- 14 RHR pump motor following oil flushing and replacement;
- 12 control rod drive pump following February 23-25 PM work window; and
- HPCI steam line high area temperature instruments following calibration and replacement.

These activities were selected based upon the SSCs ability to impact plant risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TS; the USAR; 10 CFR Part 50 requirements; licensee procedures; and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with PM tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted four post-maintenance testing samples as defined in IP 71111.19-05.

b. <u>Findings</u>

No findings of significance were identified.

1R20 <u>Outage Activities</u> (71111.20)

.1 <u>Refueling Outage Activities</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the refueling outage which began on March 14, 2009, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed by the inspectors included daily operator logs, daily shutdown safety assessments, clearance orders, corrective action documents, work orders, and operating procedures. Additionally, the inspectors conducted daily plant walkdowns to assess the condition of equipment that was important to shutdown safety and the licensee's ability to control and protect that equipment.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out-of-service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.

- Controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.
- Licensee identification and resolution of problems related to RFO activities.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- 0003; drywell high pressure scram and Group 2, 3, & secondary containment isolation test and calibration procedure; Revision 26 (routine);
- 0533; containment sump flow measurement instrumentation; Revision 14 (RCS leakage detection);
- 0060; reactor core isolation cooling (RCIC) high steam flow and low steam pressure sensor test and calibration procedure; Revision 37, and 0062; RCIC steam line high area temperature test and calibration procedure; Revision 22 (routine);
- 0032; emergency core cooling system (ECCS) pump start permissive sensor; Revision 14 (routine);
- 0137-07A; reactor steam supply valves leak rate testing; Revision 23 (containment isolation valves); and
- 0255-08-IA-8; RCIC cold shutdown check valve test; Revision 23 (inservice test).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;

- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges, and the calibration frequency were in accordance with TSs; the USAR; procedures; and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, ASME Code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three routine surveillance testing samples; one inservice testing sample; one RCS leak detection inspection sample; and one containment isolation valve sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

Introduction

A Green NCV of 10 CFR 50, Appendix B, Criterion V, was self-revealed when inadequately prepared temporary procedure changes resulted in the inadvertent repositioning of a RCIC containment isolation valve during system restoration subsequent to RCIC system testing.

Description

Early in the inspection period, a significant packing leak was discovered on MO-2076 (RCIC outboard steam isolation). Back-seating the valve resulted in a significant reduction in packing leakage.

On February 9, 2009, the licensee performed Procedures 0060 (RCIC High Steam Flow and Low Steam Pressure Sensor Test and Calibration Procedure) and 0062 (RCIC Steam Line High Area Temperature Test and Calibration Procedure). These two procedures test and calibrate TS-required RCIC system-related instrumentation. During the course of the normal performance of these procedures, MO-2075 (RCIC inboard steam isolation) and MO-2076 receive auto-closing signals. To prevent the valves from repositioning upon receipt of these signals, they are normally closed or verified closed in Step 1 of both the 0060 and 0062 procedures. To eliminate the manipulation of MO-2076 and the potential for steam leakage into the steam chase via the valve's degraded packing, the licensee prepared a one-time use procedure change for Procedures 0060 and 0062 which left MO-2076 back-seated in the open position with drive power removed. The main purpose of the temporary changes to Procedures 0060 and 0062 was to keep the MO-2076 valve open and back-seated throughout the conduct of the testing.

Later the same day, the RCIC surveillance testing was completed. During RCIC system restoration, in accordance with the guidance provided by the one-time use procedure change, drive power was restored to MO-2076 by closing its power supply breaker. Immediately upon restoration of drive power to the valve, the valve unexpectedly closed. The licensee determined that all sources of power to MO-2076 were not evaluated when preparing the one-time use procedure change for Procedures 0060 and 0062. As a result, the impact of leaving control power for MO-2076 energized, in conjunction with performing Procedures 0060 and 0062, was not evaluated. The combination of MO-2076 being open, control power being energized, and a locked-in closed signal to the valve's internal logic resulted in MO-2076 closing as soon as drive power was restored. This unexpected repositioning of MO-2076 was exactly what the one-time use procedure change was intended to prevent.

Analysis

The inspectors determined that the failure to adequately evaluate the impact of making temporary changes to existing plant procedures used to conduct TS surveillance testing was a performance deficiency warranting significance evaluation in accordance with IMC 0612, Appendix B, "Issue Disposition Screening." The inspectors determined that the finding was more than minor because, if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having work practices components, and involving aspects associated with appropriately coordinating work activities by incorporating actions to ensure that supervisory and management oversight of work activities is sufficient to support nuclear safety. [H.4(c)]

The inspectors evaluated the finding using IMC 0609, Appendix A, Attachment 1, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 Worksheet for the Mitigating Systems Cornerstone. After answering 'No' to all five questions in the Mitigating Systems Cornerstone column of Table 4a, "Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones," the inspectors concluded that the finding was of very low safety significance (Green).

Enforcement

Title 10 CFR, Part 50, Appendix B, Criterion V states, in part, that “activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with prescribed instructions, procedures, or drawings.” Contrary to this requirement, the licensee failed to adequately evaluate and prepare appropriate temporary changes to RCIC surveillance procedures prior to implementation of the one-time use procedure change. This error resulted in the inadvertent repositioning of a RCIC containment isolation valve during system restoration, subsequent to RCIC system testing. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program (CAP 1168757), it is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000263/2009002-02)

## 4. OTHER ACTIVITIES

### 4OA1 Performance Indicator Verification (71151)

#### .1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours Performance Indicator (PI) for the period from the First Quarter 2008 through the Fourth Quarter 2008. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, was used. The inspectors reviewed the licensee’s operator narrative logs; issue reports; event reports; and NRC Inspection Reports for the period of First Quarter 2008 through the Fourth Quarter 2008 to validate the accuracy of the submittals. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned scrams per 7000 critical hours sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

#### .2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications PI for the period from the First Quarter 2008 through the Fourth Quarter 2008. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, was used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports and NRC Integrated Inspection Reports for the period of the First Quarter 2008 through the

Fourth Quarter 2008 to validate the accuracy of the submittals. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned scrams with complications sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours PI for the period from the First Quarter 2008 through the Fourth Quarter 2008. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's operator narrative logs; issue reports; maintenance rule records; event reports; and NRC Integrated Inspection Reports for the period of First Quarter 2008 through the Fourth Quarter 2008 to validate the accuracy of the submittals. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned transients per 7000 critical hours sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

**Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection**

.1 Routine Review of Items Entered into the Corrective Action Program

a. Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold; that adequate attention was being given to timely corrective actions; and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the attached List of Documents Reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead; by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and; as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-Up Inspection: Motor-Operated Valve Actuator Magnesium Rotor Degradation Inspection Program

a. Scope

Based on recent OE involving degradation of motor-operated valve (MOV) actuators with magnesium rotors, the inspectors reviewed the licensee's efforts to scope and inspect applicable MOV actuators at the facility. The inspectors also reviewed the licensee's disposition of relevant OE; past inspections of MOVs with magnesium rotors; and whether any significant deficiencies in the ongoing or planned inspection efforts existed.

On December 8, 2008, the NRC issued Information Notice (IN) 2008-20, "Failures of Motor Operated Valve Actuator Motors with Magnesium Alloy Rotors." This IN, and other related OE, documents several instances where MOV actuators with magnesium rotors were degraded due to various failure mechanisms (rotor general or galvanic corrosion from moisture intrusion where coatings were missing/worn, relative thermal expansion due to exposure to elevated temperatures, and others). In some cases, the valves could not perform their intended function(s). The licensee had reviewed this IN and other previous related OE, and had identified several valve actuators at the facility that contained magnesium rotors and were potentially susceptible to the degradation identified in the OE documents. The licensee performed an OE evaluation for IN 2008-20 and established inspection procedures and inspection priorities (via boroscope or other visual examination) for in-scope valves. For safety-related valve actuators, inspections were scheduled for the spring 2009 RFO, or at the first available opportunity once the plant was online. The inspection schedule for safety-related valves was based on insights from the OE and draft guidance from the BWR Owners Group (BWROG) Valve Technical Resolution Group for the inspection of MOV actuators with magnesium rotors. For non-safety-related MOV actuators, inspections were scheduled based on the BWROG document's inspection priority criteria.

The inspectors reviewed the licensee's scoping of valves, the inspection criteria, and the scheduling priorities considering the issues identified in relevant OE and the BWROG document. The inspectors verified that the licensee considered all relevant OE and BWROG insights to determine MOV actuator vulnerability, and that the scope of valves to be inspected was appropriate, considering various actuator ambient and operating characteristics. The inspectors determined that the licensee's overall effort and future plans were based on a sound evaluation of the issues; a comprehensive scoping of applicable MOV actuators; and a proactive approach to identifying any rotor degradation.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

.4 Selected Issue Follow-Up Inspection: Functionality of Alert and Notification System Sirens during Extreme Cold Weather

a. Scope

Early in January 2009, the inspectors questioned the licensee about the cold weather capabilities of the sirens used in their Alert and Notification System. This question was based on recent issues at other nuclear facilities associated with potentially adverse impacts of extreme cold weather on the functionality of the sirens. The inspectors evaluated the licensee's efforts to resolve the inspector identified vulnerability associated with extreme cold weather impact on the functionality of the site's Alert and Notification System sirens. The inspectors specifically focused on the initial design change package that replaced the Alert and Notification System sirens, the licensee's evaluation of the inspector identified vulnerability, and the development and implementation of appropriate corrective actions to address the issue.

The licensee's investigation revealed that Monticello updated and replaced their Public Alert and Notification System sirens in 2002. This design change specified that the operating limits for the sirens was -22 °F to +140 °F, provided that battery temperature is maintained at or above -0.4 °F. Battery heater blankets were installed in the siren cabinets to maintain the minimum required battery temperatures. The investigation also revealed that recent January 2009 low temperatures, as recorded by the site's meteorological tower temperature instruments, recorded temperatures below -22 °F. Temperatures below -22 °F are outside the operating temperatures for which the sirens are designed to be fully functional and could result in a major loss of offsite response capability since the functionality of the sirens outside of their design operating range is unknown. The licensee considered the condition to be reportable in accordance with 10 CFR 50.72(b)(3)(xiii).

The Nuclear Management Company Quality Assurance Topical Report (QATR) is the licensee's top level document that establishes the quality policy and assigns major functional responsibilities for plants operated by Northern States Power Company, Minnesota (formerly Nuclear Management Company). The document states that personnel supporting nuclear generation shall comply with the requirements of the Quality Assurance Program (QAP) described in the QATR. While the QAP primarily pertains to activities affecting the performance of safety-related structures, systems, and components, it also applies to certain equipment and activities that are not safety-related. The QATR addresses these non-safety activities and equipment as stated below.

- "It is NMC's policy to assure a high degree of availability and reliability of its nuclear plants while ensuring the health and safety of the public and its workers. To this end, selected elements of the Quality Insurance Program are also applied to certain equipment and activities that are not safety related, but support safe and reliable plant operations, or where other non-CFR NRC guidance establishes program requirements. These include, but may not be limited to emergency preparedness, security, radiation protection and fire protection."

Xcel Energy Nuclear Department Corporate Directive CD1.1, "NSPM Quality Assurance Program Structure," Attachment 4, outlines which sections of the QATR are applicable to emergency preparedness. These sections include Performance/Verification Methodology (B.1), Corrective Actions (B.13), and Document Control (B.14), as well as, Assessment Methodology (C.1), Self-Assessment (C.2), and Independent Assessments (C.3). QATR Appendix B outlines the types of procedures to which section B.14 apply and includes Emergency Plan Implementing Procedures. QATR Appendix B provides the following additional information regarding the Emergency Plan Operating Procedures:

- Emergency Plan Implementing Procedures: contain instructions for activating the Emergency Response Organization and facilities, protective action levels, organizing emergency response actions, establishing necessary communications with local, state, and federal agencies, and for periodically testing the procedures, communications and alarm systems to assure they function properly. Format and content of such procedures are such that requirements of each site's NRC approved Emergency Plan are met.

The Monticello Nuclear Generating Plant Emergency Plan (MNGP), Section 6.2.1.4, states, in part, that the decision to notify the general public will be made by State and/or Local authorities based on information provided by MNGP and that notification of the general public is accomplished through Local Authorities' use of the Public Alert and Notification System, Emergency Alert System, and/or a telephone auto dialing system.

The inspectors determined that the licensee failed to adequately evaluate and control the design of the Alert and Notification System sirens to ensure that the sirens remained functional during site-specific vulnerabilities such as extreme cold weather. After conferring with both region-based and headquarters-based emergency preparedness inspection staff, the inspectors concluded that, based on existing provisions in the licensee's procedures that implement their emergency preparedness plan and the licensee's planned corrective actions to specifically address the existing siren extreme cold weather vulnerability, the issue was of minor safety significance.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 Non-Routine Operator Activities: Establishing Plant Conditions to Facilitate the Repair of a Leaking Pipe Elbow Located on the 12 Reactor Feedwater Pump Minimum Flow Line

On December 31, 2008, the licensee identified a leak on a pipe elbow located on the 12 reactor feedwater pump minimum flow line downstream of CV-3490. On January 1, 2009, the licensee reduced reactor power to approximately 40 percent, removed the 12 reactor feedwater pump from service, and isolated the leaking pipe elbow from the feedwater system.

The inspectors observed portions of the power reduction, removal of the 12 reactor feedwater pump from service, and several licensee meetings and challenge calls associated with planning the subsequent repair of the leaking pipe elbow.

No findings of significance were identified during this inspection. The inspectors evaluated the licensee's repair of the leaking elbow and documented their inspection findings in Section 1R18 of this report.

This non-routine evolution observation constituted one sample as defined in IP 71153-05.

.2 Non-Routine Operator Activities: Taking the Main Generator Off-Line

On March 13, 2009, the licensee began reducing reactor power in preparation for reactor plant shutdown and commencement of RFO 24. At 00:06 on March 14, 2009, the main generator was taken off-line and turbine shutdown commenced.

From the control room, the inspectors observed the operators' actions to take the main generator off-line and shutdown the turbine. The primary focus of the inspectors during this inspection was to evaluate communications between the operators, command and control by senior licensed control room staff, and proper coordination of activities during a critical portion of the plant shutdown. No findings of significance were identified during this inspection.

This non-routine evolution observation constituted one sample as defined in IP 71153-05.

.3 Event Response: Invalid Actuation of Secondary Containment

On January 20, 2009, during the performance of Procedure 0003, "Drywell High Pressure Scram and Groups 2, 3, and Secondary Containment Isolation Test and Calibration," the standby gas system automatically initiated and secondary containment isolated. This resulted in increased steam chase temperatures and entry into several TS action statements. The licensee determined that the cause of the standby gas treatment system initiation and secondary containment isolation was the presence of a high resistance on two contacts of a relay that was being actuated as a result of the performance of the surveillance. The licensee determined that this issue was reportable per 10 CFR 50.73(a) as a Non-Reactor Protection System (RPS) Invalid Engineered Safety Feature (ESF) Actuation.

The inspectors evaluated the control room operators' response to the unplanned standby gas treatment system actuation and secondary containment isolation, the troubleshooting efforts to identify and repair the cause, and the restoration of standby gas treatment system and secondary containment components to their normal configuration. No findings of significance were identified during this inspection.

This event follow-up review constituted one sample as defined in IP 71153-05.

.4 Event Response: Suspicious Package Found in Mailroom

On January 30, 2009, the licensee discovered an untraceable package in the Station's mailroom. The mailroom is located outside of the Protected Area. Security personnel were dispatched to evaluate the package. The package tested positive for explosives and was subsequently moved to a more remote location within the owner controlled area. Members of local law enforcement, Federal Bureau of Investigation (FBI), and a bomb squad responded to the site to further evaluate the suspicious package. Subsequent evaluations determined that the package did not contain any explosive materials. The licensee determined that this issue was reportable per 10 CFR 50.72(b)(2)(xi).

The inspectors evaluated all aspects of the licensee's response to this event, including their interaction local law enforcement and other external agencies. No findings of significance were identified during this inspection.

This event follow-up review constituted one sample as defined in IP 71153-05.

.5 Event Response: Loss of Offsite Communications

At approximately 9:30 on the morning of January 22, 2009, the licensee became aware of a loss of the emergency notification system (ENS), health physics network (HPN), and emergency response data system (ERDS) lines, as well as a partial loss of incoming commercial phone lines. The licensee contacted the NRC Operations Center and established a backup means of communication. Backup communication methods with off-site State and local agencies remained available. The licensee determined that communications were lost due to a problem at the local telephone company office and was not due to any on-site equipment issues. Full communications capabilities were verified to be restored at 14:00 later the same day. The licensee determined that this event was reportable per 10 CFR 50.72(b)(3)(xiii).

The inspectors evaluated all aspects of the licensee's response to this event, including the verification that alternate means of communications existed throughout the event for the licensee to contact the NRC and State and local agencies. No findings of significance were identified during this inspection.

.6 (Closed) Licensee Event Report (LER) 05000263/200808-00: Technical Specification Required Shutdown Margin not met during All Conditions for Refueling Outage 23

Following a review of an industry refueling shutdown margin best practice, the licensee evaluated the best practice and tested the practice on core alterations made during the 2007 RFO 23 with SIMULATE-3, taking into account both a core exposure dependent bias and the use of a three-dimensional method. On November 13, 2008, it was determined that the TS shutdown margin was not met for all conditions during the RFO. The licensee determined that the root cause for not meeting the shutdown margin during RFO 23 was insufficient verification and validation, using known and industry accepted higher order methods, for determining TS shutdown margin compliance. The corrective action taken by the licensee to prevent recurrence was to require utilizing only a three-dimensional model when determining shutdown margin during future core alterations and operating cycles.

The inspectors evaluated the finding using IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process, Phase 1 Operational Checklists for Both PWRs and BWRs." The finding did not meet the criteria on Checklist 7, "BWR Refueling Operations with RCS Level > 23 feet," for requiring a Phase 2 or Phase 3 SDP and, as a result, screened to be of very low safety significance (Green). The licensee-identified finding involved a violation of TS 3.1.1a. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

This event follow-up review constituted one sample as defined in IP 71153-05.

4OA5 Other Activities

.1 Unit 1 Power Uprate Related Inspection Activities (71004)

During this inspection period, the inspectors observed several activities related to the power uprate amendment. As documented in Sections 71111.13 and 71111.18 above, the inspectors reviewed the power uprate related modification packages for emergent work and troubleshooting of MO-2020 (Division I RHR drywell spray outboard valve); and power range nuclear monitoring system installation, respectively.

.2 Licensee Strike Contingency Plans (92709)

During this inspection period, on two separate occasions the inspectors were notified by the licensee that the local union, which represents the security personnel, had posted notices which outlined the union's intention to have a strike vote. Based on this information, the inspectors began to perform the inspection activities described in IP 92709, "Licensee Strike Contingency Plans." Since a strike vote had not been taken, the inspectors focused on preparations the licensee was making to address security staffing at the site should an actual strike occur. The inspection activities included speaking with local union representatives and senior licensee management and reviewing the licensee's strike contingency plan.

On March 23, 2009, the inspectors were notified that the pending strike vote was cancelled and contract negotiations had resumed.

## 4OA6 Management Meetings

### .1 Exit Meeting Summary

On April 1, 2009, the inspectors presented the inspection results to Mr. O'Connor and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

### .2 Interim Exit Meetings

Interim exits were conducted for:

- On April 1, 2009, the inspectors presented the inservice inspection (71111.08G) results to the Site Vice-President, Mr. T. O'Connor, and other members of the licensee staff. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

### .3 End-of-Cycle Assessment Results Discussion

On April 1, 2009, directly following the quarterly integrated resident inspection exit meeting, the NRC met with Mr. T. O'Connor, Site Vice President, and members of the licensee staff to discuss their performance during the previous four quarters for the 2008 End-of-Cycle assessment, during which the plant was continually within the Licensee Response Column of the Action Matrix, in accordance with Section 06.05 of IMC 0305.

## 4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) were identified by the licensee and are 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.

- Technical Specification 3.1.1a, "Shutdown Margin," requires, in part, that the reactor shutdown margin be greater than or equal to 0.38 percent delta k/k at all times with the analytically determined highest worth rod fully withdrawn. Contrary to this requirement, the licensee's re-evaluation of shutdown margin for the most limiting point during the End-of-Cycle 23 core alterations was -0.86 delta k/k, assuming control rod 38-15 had been fully withdrawn. This issue was documented in the licensee's corrective action program as CAP 01159084. The finding did not meet the criteria of Checklist 7, "BWR Refueling Operations with RCS Level > 23 feet," for requiring a Phase 2 or Phase 3 SDP and, as a result, screened to be of very low safety significance (Green).

- Technical Specification 5.4.1 requires, in part, that written procedures shall be implemented covering the Fire Protection Program. Monticello's Administrative Work Instruction (AWI) 4 AWI-08.01.00, "Fire Protection Program Plan," identifies implementing documents for the MNGP fire protection program. Section 4.6 of 4 AWI-08.01.00 identifies Procedure 4 AWI-08.01.01, "Fire Prevention Practices" as the implementing document for the fire brigade, including program requirements for fire brigade minimum staffing and member qualification requirements. Specifically, Section 4.4.2 of 4 AWI-08.01.01 states, in part, that the Operations Manager SHALL ensure that at least five members of the fire brigade are on-site at all times. Furthermore, Section 4.4.3.C of 4 AWI-08.01.01 states, in part, that fire brigade members SHALL receive annual respiratory protection qualification. Contrary to these requirements, a non-licensed operator did not complete respirator refresher and self-contained breathing apparatus training, was unqualified as a fire brigade member, and stood duty on several dates in November 2008. During the dates where the individual stood duty, the licensee did not have at least five fire brigade members on shift. The issue was documented in the licensee's corrective action program (CAP 01160370). This finding is not subject to significance determination. However, it was determined to be of very low safety significance based on NRC management review.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee

T. O'Connor, Site Vice President
B. Sawatzke, Plant Manager
J. Grubb, Site Engineering Director
K. Jepson, Business Support Manager
S. Sharp, Operations Manager
W. Flaga, Acting Maintenance Manager
B. Cole, Radiation Protection/Chemistry Manager
G. Salamon, Regulatory Affairs Manager
B. Dixon, Engineering Analyst
J. Rootes, Engineering Analyst
R. Baumer, Compliance Engineer
R. Diopere, ISI Program Owner

Nuclear Regulatory Commission

K. Riemer, Chief, Reactor Projects Branch 2

## LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000263/2009002-01 | NCV | Use of Non-Quality Documents in the Absence of Approved Procedures to Modify the Normal Operation of IRM 11 and IRM 18 (Section 1R15) |
| 05000263/2009002-02 | NCV | Failure to Adequately Evaluate Temporary Changes to RCIC Surveillance Procedures Prior to Implementation (Section 1R22) |

Closed

| | | |
|---|---|---|
| 05000263/2009002-01 | NCV | Use of Non-Quality Documents in the Absence of Approved Procedures to Modify the Normal Operation of IRM 11 and IRM 18 (Section 1R15) |
| 05000263/2009002-02 | NCV | Failure to Adequately Evaluate Temporary Changes to RCIC Surveillance Procedures Prior to Implementation (Section 1R22) |
| 05000263/2008008-00 | LER | Technical Specification Required Shutdown Margin not Met during All Conditions for Refueling Outage 23 (Section 4OA3) |

## LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Section 1R01

B.08.07-05, Section G.9; Extreme Cold Weather Procedure; Revision 20

Section 1R04

2122; Plant Prestart Checklist Reactor Level Control System; Revision 3
2109; Plant Prestart Checklist Condensate and Feedwater System; Revision 14
2154-04; Condensate and Feedwater System Prestart Valve Checklist; Revision 34
CAP 01125950; A FW Regulating Valve Locked-Up while Reducing Reactor Power
CAP 01083900; FW-94-1 did not Meet Administrative Acceptance Criteria of 0137-08-01
CAP 01090307; Erratic FW Flow during Start Up Evolutions
CAP 01149748; Loss of Level Indication ‘C’ MSDT Dump Controller LC-1003
CAP 01152545; Low Level in 15A FW Heater Below 65 Percent Power
CAP 01157187; 13A FW Heater May have been Operated with Low Water Level
CAP 01157201; 12A FW Heater Operated with High Flow
CAP 01164823; 13B FW Heater Drain Approach Temperature Increased during Low Water Event
CAP 01164197; LT-1054 E-13B Drain Level Transmitter Failed
OEER 01147689; NRC IN 2008-13 Main Feedwater System Issues and Related 2007 Reactor Trip Data
2126-02; Plant Prestart Checklist – Batteries and DC Power Systems 125 VDC; Revision 16
2111; Plant Prestart Checklist RBCCW System; Revision 7
2154-24; Reactor Building Closed Cooling Water System Prestart Valve Checklist; Revision 16
Operations Manual C.4-B.02.05.A; Loss of RBCCW Flow; Revision 12
1386-05; RHRSW External Pipe Inspection Loop A and B; Revision 8
2154-23; RHR Service Water System Prestart Valve Checklist; Revision 29
M-112; P&ID, RHR Service Water and Emergency Service Water System; Revision 82
2126-02; Plant Prestart Checklist, Batteries and DC Power System 125VDC; Revision 18
2126-03; Plant Prestart Checklist, 250 VDC Batteries and DC Power System; Revision 16

Section 1R05

Fire Strategy A.3-24; Diesel Fire Pump Room; Revision 8
Fire Strategy A.3-08; Cable Spreading Room; Revision 11
0328; Cable Spreading Room Halon System; Revision 17
CAP 01165319; CSR Damper did not Close During Air Flow Test
Fire Strategy A.3-21D; Radwaste Building 947’ Elevation; Revision 4
Fire Strategy A.3-21C; Radwaste Shipping Buildings; Revision 6
Fire Strategy A.3-21B; Radwaste Trash Compactor; Revision 3
Fire Strategy A.3-21A; Radwaste Control Room; Revision 3
Fire Strategy A.3-14C; Railroad Car Area (Turbine Building); Revision 3
Fire Strategy A.3-14B; Isophase Bus Blower; Revision 4

Fire Strategy A.3-14A; Upper 4kV Bus Area (12, 14, and 16), Revision 13
Fire Strategy A.3-17; Turbine Building North Cable Corridor, 941'; Revision 4
Fire Strategy A.3-16; Corridor, Turbine Building East & West, Elevations 911' and 931'; Revision 10

<u>Section 1R06</u>

PRA-CALC-04-001; Flood Areas
PRA-CALC-04-003; Flood Sources Identification
PRA-CALC-04-004; Flood Initiating Event Frequencies
PRA-CALC-04-005; Equipment Vulnerabilities to Flooding
PRA-CALC-04-006; Flood Scenarios and Effects

<u>Section 1R07</u>

1136; RHR Heat Exchanger Efficiency Test; Revision 29

<u>Section 1R08</u>

CAP 01102080; Disc Nut was not Replaced during Reassembly of ESW-1-2
CAP 01082758; ISI-Head Vent Pipe Hanger has Bent Rod
CAP 01117340; Station Review of NRC Violation 2007-04-01
CAP 01088981; Advanced Trend:  Monitoring Fails to Pre-Identify Pipe Wall Failures
CAP 01109115; NRC Potential Green NCV on PEI-02.03.12 Inadequate
CAP 01088604; ESW-44, Pin Hole Leak on Backside of Pipe
CAP 01083461; Pipe Wall Thinning Upstream of ESW-31
CAP 01084933; IWE Indications-Drywell Head Flange Bolting
CAP 01145852; Slight Leakage Found on DPIS-14-43A Valve Manifold
CAP 01084505; ISI Requirements Step Performed out of Sequence
2009MT001; Magnetic Particle Examination of RHR Heat Exchanger a Weld N-2; dated March 27, 2009
2009UT010; UT Calibration/Examination of Core Spray a Pipe-to-Penetration Weld W-3; March 24, 2009
2009UT017; UT Calibration/Examination of Main Steam C Pipe-to-Elbow Weld W-25; March 26, 2009
PEI-02.01.01; Solvent Removable, Visible Dye Penetrant Examination; Revision 1
PEI-02.02.01; Dry Powder Magnetic Particle Examination; Revision 0
FP-PE-NDE-401; Ultrasonic Examination of Ferritic Pipe Welds – Supplement 3; Revision 2
FP-PE-NDE-425; Ultrasonic Thickness Examination – Localized Corrosion; Revision 1
FP-PE-NDE-530; Visual Examination, VT-3; Revision 3
PEI-02.03.05; Ultransonic Thickness Measurement; Revision 1
8146; Scaffold Control; Revision 29
4 AWI-04.05.10; Scaffolding Controls; Revision 7
WO 00323428; Replace/Repair Bent Rod @ Pipe Support V15-H3 on RPV Head Vent
WO 00151841; Replace MO-2373
WO 00151841; Replace MO-2374
2007VT023; Visual Examination of Component Supports and Snubbers (VT-3); dated March 18, 2007

Section 1R12

Monticello Maintenance Rule Program System Basis Document; 4.16kV Station Auxiliary
Maintenance Rule (a)(1) Action/Performance Improvement Plan; 4kV Bus 13, Bus 14, and 1R; 10/27/2008
CAP 01076866; Switch for 3N4 Hydraulic System Pressure Opened by Xcel
CAP 01082734; Lost Essential Bus 16 during Isolation Activities
CAP 01150362; Reactor Scram Number 121 Occurred on September 11, 2008
Monticello Maintenance Rule Program Basis Document: 125Vdc Battery System
DBD B.09.10; Design Basis Document for 125 Vdc System; Revision 4
Maintenance Rule (a)(1) Action Plan for D10 and D54 Battery Chargers; dated July 2008
MNGP Plant Procedure 0194; 11 and 12 125Vdc Battery Operability Check – Weekly Test; Revision 25
MNGP Plant Procedure 4066-PM; 125 Vdc Battery; D10 Battery Charger Preventive Maintenance; Revision 0
CAP 1137297; D10 Exhibits Erratic Voltage Output during Surveillance
CAP 1139415; 125 Vdc Battery Charger D10 Exceeds Maintenance Rule Unavailability
CAP 01168838; Replacement Transmitter Failed Bench Calibration
CAP 01173500; 13A FW Heater Dump Transmitter Failed After Replacement
CAP 01173668; Replacement Transmitter Failed Bench Testing for LT-1016
CAP 01174949; Adverse Trend in Feedwater Heater Level Transmitter Replacements
CAP 01174951; FW Heater Level Transmitter Does Not Work After Installation
CAP 01171964; Failure of Replaced LT-3124 to Calibrate
WO 376883; LT-1055, 13 B FW Heater Dump Level Transmitter Replacement

Section 1R13

WO 376594; MO-2076, Electrically Backseat
CAP 01165565; Thrust Estimate for MO-2076 Did not include All Uncertainty
WO 377219; Perform Check of Relay Contacts
EC 12361; EPU – Provide Operations the Ability to Throttle MO-2020 and MO-2021; Revision 0
WO 370902; EPU – EC-12361 – Remove and Replace Hand Switch 10A-S9A
CAP 01171702; Troubleshooting MO-2020 Identified Out-of-Spec Voltages
CAP 01171769; EC 12361 Pre-operational Testing Identifies Issue with MO-2020
CAP 01171673; EC-12361 MO-2020 (Drywell Spray) Circuit Configuration Concerns
CAP 01171823; EPU – Wiring Discrepancy Discovered in Field (EC 12361)
CAP 01171876; Installed Wiring Did not Match Controlled Drawing
CAP 01171900; EC-12361 MO-2020 (Drywell Spray) – ECN Error
CAP 01171953; Decision to Restore From EC 12361 In-Process Installation

Section 1R15

CAP 01164795; Water Leakage Identified on PCT Electrical Penetration X-104B
CAP 01164976; Unplanned TS Action Entry due to RCIC Inoperability
Operational Decision Making Issue Evaluation [associated with CAP 1164795]; dated January 11, 2009
1205; Systems Leakage Check Procedure – Reactor Core Isolation Cooling System; Revision 13
CAP 01113447; Pinhole Leak in Service Water Line at Stator Cooling Skid
CAP 01169229; Damaged Cables Noted above Load Centers LC103 and LC109
CAP 01167642; AO-2-86B Outboard MSIV has Dual Indication

CAP 01173014; IRM 11 and 18 could not Be Calibrated Properly
0018; IRM Heat Balance Calibration; Revision 19
CAP 1173169; Technical Specification for ‘B’ Core Spray Flow Possibly Non-Conservative
CAP 1174274; Emergency Operating Procedures Guidance for ‘B’ Core Spray
Northern States Power License Amendment Request; Revised Core Spray Pump Flow and other Editorial Changes; dated February 12, 1993
TS Amendment Number 93; Revised Core Spray Pump Flow; dated July 12, 1995

Section 1R18

EC 13607; Pipe Repair Downstream of CV 3490
CA-09-100; Qualification of the Repair to the Elbow Near Support FWH-90 on Line FW4-8”-DE with a 10”Sch. 80 Patch
EC 10856; Power Range Neutron Monitoring (PRNM) System
CAP 01173226; EPU-PRNM-Existing Drawing Does not Match Existing Condition
CAP 01174089; Rod Block Withdrawal Alarm Caused by PRNM

Section 1R19

4100-01-PM; EDG Air Start System Maintenance; Revision 14
4100-03-OCD; 12 Emergency Diesel Generator 1 Starting System; Revision 10
WO 372205; P-202D Flush Motor Lower Bearing Reservoir
4135-01; Lubrication and Oil Replacement; Revision 4
4135-02; Obtaining Oil Samples; Revision 5
4 AWI-04.05.06; Post-Maintenance and Return to Service Testing; Revision 20
WO 361225; PM 4200-2 (12 Control Rod Drive Pump)
WO 332805; ES-3-240, Check Out Power Supply Ripple As-Found
WO 331307; 12 CRD Pump Gear Unit Leaking Oil
WO 361869; CRD Drive Water Header Pressure Gauge is not Working
WO 356023; Leak on Union Fitting Near CRD-50-2
WO 366119; Perform 4948-PM 4KV/480V Motor Online Testing
0058; HPCI Steam Line High Area Temperature Test and Calibration Procedure; Revision 24
0056; HPCI HI Steam Flow and Low Steam Pressure Sensor Test and Calibration Procedure; Revision 44
0255-06-IA-1; HPCI Quarterly Pump and Valve Tests; Revision 80

Section 1R20

Operations Manual C.3; Shutdown Procedure; Revision 54
2300; Reactivity Adjustment; Revision 1 (Shutdown for Refueling Outage 24, March 13, 2009)
4 AWI-08.15.03; Risk Management for Outages; Revision 3
Monticello 2009 Refueling Outage Risk Management Guidelines
CAP 01172688; Vessel Temperature Monitoring Not Fully Functional
CAP 01172840; C.3 Shutdown Prerequisite Not Performed as Written
CAP 01172962; 12 Reactor Feedwater Pump Tripped During Reactor Cooldown and Depressurization
CAP 01172970; Reactor Water Cleanup Pump Tripped While Establishing Dump Flow to Hotwell
CAP 01174099; Danger-Tagged Component Operated
CAP 01174194; FW 98-1 Packing Leak During 0255-12-IIC-1
CAP 01174496; Possible Valve Leakage Causes Four Control Rods to Drift

Section 1R22

CAP 01168757; Unintended Closure of MO-2076 during Restoration of Power
CAP 01173431; AO-2-86D Has Excessive Seat Leakage
CAP 01174250; AO-2-80C Exceeded Appendix J Administrative Limit
CAP 01174283; Unable to Leak Rate Test MO-2076 due to Excessive Leakage
CAP 01174276; MO-2075 (RCIC Inboard Isolation Valve) Failed Administrative Limit
CAP 01174444; Local Leak Rate Test (LLRT) of HPCI Outboard Valve MO-2035 Exceeds Administrative Limit
CAP 01174869; Minor Discrepancies Discovered with LLRT Methodology
CAP 01174606; Small Packing Leak on HPCI Inboard Valve MO-2034
CAP 01175793; RCIC Exhaust Steam Line Vacuum Breaker Fails IST Exercise Test
CAP 01175829; RCIC-16 (Vacuum Pump Discharge Check Valve) Failed Leak Test

Section 4OA2

OEER 01066797; Station OE Screening Team Review of OE for Week of 12/08/06
GAR 01075911; Determine Means of Monitoring Degradation of MOV Motor Rotor
EWI-08.15.01; Motor Operated Valve Program; Revision 14
CAP 01167906; Develop Method to Monitor Magnesium Rotor Degradation
CAP 01159881; NRC IN-06-26 MOV Magnesium Rotor Degradation
CAP 01158444; OEE Potentially Adverse to Quality not Entered into CAP
MEI-09.03; MOV Preventive Maintenance and Periodic Verification Plan; Revision 8
WO 362605; Magnesium Rotor Inspection
OEER 01164115-02; IN-2008-20, Failure of MOV Actuator Motors with Magnesium Alloy Rotors

Section 4OA3

Operations Memo 09-01; Operational Procedural Guidance and Contingency Plans for Leak from 12 Feedwater Pump Recirculation Line; dated January 1, 2009
2300; Reactivity Adjustment (and applicable attachments which documented proposed reactivity maneuvering steps) for the January 1st Power Reductions to Approximately 40 Percent Power
CAP 01166058; Secondary Containment, Standby Gas System Initiated

## LIST OF ACRONYMS USED

| | |
|---|---|
| ASME | American Society of Mechanical Engineers |
| AWI | Administrative Work Instruction |
| BWR | Boiling Water Reactor |
| BWROG | BWR Owner's Group |
| CAP | Corrective Action Program |
| CFR | Code of Federal Regulations |
| CRD | Control Rod Drive |
| DG | Diesel Generator |
| DRP | Division of Reactor Projects |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| ENS | Emergency Notification System |
| EPU | Extended Power Uprate |
| ERDS | Emergency Response Data System |
| ESF | Engineered Safety Feature |
| °F | Degrees Fahrenheit |
| FBI | Federal Bureau of Investigation |
| FIN | Finding |
| HPCI | High Pressure Coolant Injection |
| HPN | Health Physics Network |
| IMC | Inspection Manual Chapter |
| IN | Information Notice |
| IP | Inspection Procedure |
| IR | Inspection Report |
| IRM | Intermediate Range Monitor |
| ISI | Inservice Inspection |
| kV | Kilovolt |
| LER | Licensee Event Report |
| MNGP | Monticello Nuclear Generating Plant Emergency Plan |
| MOV | Motor-Operated Valve |
| MSIV | Main Steam Isolation Valve |
| MT | Magnetic Particle Examination |
| NCV | Non-Cited Violation |
| NDE | Nondestructive Examination |
| NEI | Nuclear Energy Institute |
| NRC | U.S. Nuclear Regulatory Commission |
| OE | Operating Experience |
| OOS | Out-of-Service |
| OPRM | Oscillating Power Range Monitor |
| OSP | Outage Safety Plan |
| PI | Performance Indicator |
| PM | Planned or Post-Maintenance |
| QAP | Quality Assurance Program |
| QATR | Quality Assurance Topical Report |
| RBCCW | Reactor Building Closed Cooling Water |
| RCIC | Reactor Core Isolation Cooling |
| RCS | Reactor Coolant System |
| RFO | Refueling Outage |
| RHR | Residual Heat Removal |

| | |
|---|---|
| RHRSW | Residual Heat Removal Service Water |
| RPS | Reactor Protection System |
| SDP | Significance Determination Process |
| SSC | Systems, Structures, and Components |
| TS | Technical Specification |
| USAR | Updated Safety Analysis Report |
| UT | Ultrasonic Examination |
| Vdc | Volts Direct Current |