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

August 8, 2011

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
INTEGRATED INSPECTION REPORT 05000263/2011003

Dear Mr. O'Connor:

On June 30, 2011, 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 June 28, 2011, 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, three self-revealed findings and one NRC-identified finding of very low safety significance were identified. Two of the findings involved violations 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 the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

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 cross-cutting aspect assigned to 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 System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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 05000263/2011003
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	05000263/2011003
Licensee:	Northern States Power Company, Minnesota
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello, MN
Dates:	April 1 to June 30, 2011
Inspectors:	S. Thomas, Senior Resident Inspector P. Voss, Resident Inspector J. Bozga, Reactor Inspector M. Mitchell, Health Physicist A. Scarbeary, Reactor Engineer G. O'Dwyer, Reactor Engineer
Approved by:	K. Riemer, Chief Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000263/2011003; 04/01/2011 - 06/30/2011; Monticello Nuclear Generating Plant. Maintenance Risk Assessment and Emergent Work Control; Outage Activities; Radiological Hazard Assessment and Exposure Controls; Follow-Up of Events and Notices of Enforcement Discretion.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were self-revealed and one Green finding was NRC-identified. Two of the findings were considered violations 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: Initiating Events

- Green. A finding of very low safety significance and non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when an unexpected recirculation pump runback occurred during the performance of Reactor Dynamics Testing. The event was the result of the licensee failing to adequately assess the operational impact of a recent revision to Procedure C.1, "Startup Procedure," which resulted in operating the plant in a manner that challenged feedwater pump protective features. The licensee entered this issue into their corrective action program (CAP 01288070) and initiated corrective actions to address the issue. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency was associated with the cross-cutting area of Human Performance, having decision-making components, and involving aspects associated with the licensee conducting effectiveness reviews of safety-significant decisions to verify the validity of the underlying assumptions, identify possible unintended consequences, and determine how to improve future decisions. [H.1(b)]

  The finding was more than minor because it impacted the procedure quality attribute of the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors applied IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," to this finding. The inspectors utilized Column 1 of the Table 4a worksheet to screen the finding. The inspectors answered 'No' to the questions associated with loss-of-coolant accident (LOCA) initiators, transient initiators, and external events initiator, and screened the finding to be of very low safety significance. (Section 4OA3)

- Green. A finding of very low safety significance was self-revealed when, on two separate occasions, CV-3490 (12 reactor feedwater pump recirculation to the condenser) failed closed while the 12 reactor feedwater pump was being placed in service. The cause of each failure was directly related to poor maintenance practices while performing work on CV-3490's valve positioner. Additionally, each failure resulted

in an automatic trip of the 12 reactor feedwater pump. The licensee entered this issue into their corrective action program, corrected the mechanical issues, and performed an extent-of-condition review. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having work practice components, and involving aspects associated with ensuring that supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported. [H.4(c)]

The finding was more than minor because it impacted the configuration control attribute of the Initiating Events Cornerstone objective of limiting those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors applied IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” to this finding. The inspectors utilized Column 1 of the Table 4a worksheet to screen the finding. The inspectors answered ‘No’ to the question “does the finding contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available” and, therefore, the finding was screened to be of very low safety significance. (Section 1R13)

Cornerstone: Barrier Integrity

- Severity Level IV. A Severity Level IV non-cited violation (NCV) of 10 CFR 50.71(e), “Periodic Update of the Final Safety Analysis Report” and an accompanying Green finding were identified by the inspectors for the licensee’s failure to update the Updated Safety Analysis Report (USAR) with the cask maximum lift height restrictions imposed by Nuclear Regulatory Commission (NRC) staff. As a result, the licensee had not adequately evaluated whether the plant licensing basis necessitated retention of cask lift height limitations when transitioning from the use of the 25 ton NFS-4 or 25 ton NAC-1 spent fuel shipping cask and 70 ton IF-300 spent fuel shipping cask to the heavier 105 ton NUHOMS cask. The licensee entered this issue into its corrective action system.

  The inspectors determined that the failure to update the USAR with the cask lift height restrictions for the 25 ton and 70 ton spent fuel cask was contrary to 10 CFR 50.71(e) and was a performance deficiency warranting a significance evaluation. Violations of 10 CFR 50.71 (e) are dispositioned using the traditional enforcement process instead of the SDP because they are considered to be violations that potentially impede or impact the regulatory process. However, if possible, the underlying finding is evaluated under the SDP to determine the significance of the violation. The finding was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," because, if left uncorrected, the performance deficiency could have led to a more significant safety concern. Specifically, the inspectors could not readily conclude that the absence of lift height limitations would not require additional calculational analyses and/or require a license amendment. The inspectors determined that the finding was of very low safety significance following a qualitative significance determination review. Specifically, the inspectors determined that only seismic events exceeding the level of an Operational Basis Earthquake (OBE) of 0.03g could impact core damage frequency (CDF). The licensee supplied information that the median annual probability of exceeding the peak ground acceleration for the OBE at Monticello was approximately 7.0E-4/yr. In addition, the predicted shipping cask lifts was 19.2/yr with an average lift duration of 30 minutes. Thus, the frequency of exceeding the OBE while lifting a shipping cask was estimated to be 7.7E-7/year. This value is a bounding

frequency estimate for delta-CDF in that it does not imply with certainty that there will be a cask drop during an earthquake nor does it imply with certainty of core damage during an earthquake given a cask drop. The Senior Reactor Analyst (SRA) concluded that the risk due to simultaneous occurrence of an OBE or greater seismic event during use of the reactor building crane for shipping cask lifts was best characterized as very low (Green). The inspectors determined that this finding did not reflect current performance because it was a legacy issue with the failure to properly update the USAR occurring almost 30 years ago and, therefore, there was no cross-cutting aspect associated with this finding. (Section 1R20)

Cornerstone: Occupational/Public Radiation Safety

- Green. A finding of very low safety significance (Green) was self-revealed due to the licensee having unplanned and unintended occupational collective radiation dose because of deficiencies in the licensee's as-low-as-is-reasonably-achievable (ALARA) planning and work control program. Specifically, the licensee failed to properly incorporate ALARA strategies or insights while planning and executing a maintenance activity on the 'C' inboard main steam isolation valve. This issue resulted in the expansion of collective exposure for this work from 4.044 person-rem to 9.654 person-rem. The licensee entered this issue into their corrective action program as CAP 1281395.

  The finding was more than minor because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone. Additionally, this issue affected the cornerstone objective of ensuring the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Also, the finding was similar to Example 6.i in Appendix E of IMC 0612, in that it resulted in a collective exposure of greater than 5 person-rem and exceeded the outage goal by greater than 50 percent. The inspectors determined that this finding was of very low safety significance because Monticello Nuclear Generating Plant's (MNGP's) current three-year rolling average collective dose is 136.266 person-rem, less than the 240 person-rem per unit standard. This finding had a cross-cutting aspect in the area of Human Performance, related to the cross-cutting component of work control, in that the outage plan did not adequately incorporate actions to address the impact of work on different job activities. [H.3(b)] (Section 2RS2)

B. Licensee-Identified Violations

No violations were identified.

REPORT DETAILS

Summary of Plant Status

The refueling outage continued during the first 51 days of this inspection period. The licensee began to startup the reactor on May 22, 2011, synched the main generator to the electric grid on May 25, 2011, and reached 100 percent power on May 31, 2011. The licensee operated the plant at approximately 100 percent power until a leaking safety relief valve required a plant shutdown on June 24, 2011. Subsequent to the repair of the safety relief valve, the licensee began to startup the reactor on June 27, 2011, synched to the grid on June 28, 2011, and had not yet achieved 100 percent power by the end of this inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems during adverse weather were appropriate. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors' review included:

- The coordination between the TSO and the plant during off-normal or emergency events;
- The explanations for the events;
- The estimates of when the offsite power system would be returned to a normal state; and
- The notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- The actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;
- The compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;

- A re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- The communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

Documents reviewed are listed in the Attachment to this report. The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures.

This inspection constituted one readiness of offsite and alternate AC power systems sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings 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:

- emergency filtration train (EFT)/control room ventilation (CRV) train ‘B’ during work on EFT Train ‘A’;
- 11 and 12 standby liquid control systems (SBLCs);
- standby gas treatment (SBGT) trains ‘A’ and ‘B’; and
- ‘B’ residual heat removal service water (RHRSW) system restoration following forced outage.

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 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 were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On June 13-14, 2011, the inspectors performed a complete system alignment inspection of the reactor core isolation cooling system (RCIC) to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; 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 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 12E (steam jet air ejector room);
- Fire Zone 37 (transformers—during initial main transformer energization);
- Fire Zone 14C (railroad car area—turbine building during condensate demin pipe support hot work);
- Fire Zone 14A (upper 4kV bus area); and
- Fire Zone 1F (torus area—elevation 896').

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 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.
Using the documents listed in the Attachment to this report, 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 were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On June 20, 2011, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify 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:

- licensed 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. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings 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:

- nuclear instrumentation system—start up range monitors.

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 one quarterly maintenance effectiveness sample as defined in IP 71111.12-05.

b. Findings

No findings 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:

- SBGT emergent work on system due to valves sticking closed;
- relay 10A-K36B contacts not open as expected during emergency core cooling system (ECCS) surveillance test;
- CV-3490 valve positioned failures; and
- stator cooling water flow adjustment.

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 four samples as defined in IP 71111.13-05.

b. Findings

Introduction

A finding of very low safety significance was self-revealed when, on two separate occasions, CV-3490 (12 reactor feedwater pump (RFP) recirc to the condenser) failed closed while 12 RFP was being placed in service. The cause of each failure was directly related to poor maintenance practices while performing work on the CV-3490's valve positioner. Additionally, each failure resulted in an automatic trip of 12 RFP.

Description

During the refueling outage (RFO), which completed May 25, 2011, a licensee contractor performed maintenance on Valve CV-3490. This maintenance required disassembly of the valve and removal of the actuator and valve positioner. Subsequent to the completion of the maintenance activity, the positioner was installed, adjusted, and the valve was diagnostically tested.

On May 23, 2011, at approximately 23:53, with reactor power at approximately 12 percent, CV-3490 failed closed, resulting in a low flow trip of 12 RFP.

On May 24, at approximately 00:35, 11 RFP was promptly placed in service with minimal impact on reactor water level. A subsequent licensee investigation revealed that the cause of CV-3490 failing closed was that the bellows spring in the valve's positioner had become uncoupled due to the bellows jam nut not being secured in place. The licensee initiated a WO to reconnect the bellows to the proper configuration and calibrate CV-3490.

On May 24, 2011, at approximately 21:16, with the reactor at approximately 12 percent power, the control room operators attempted to place 12 RFP in service to facilitate testing of CV-3490. Approximately two minutes after the pump was started, CV-3490 failed closed, resulting in a low flow trip of 12 RFP. A subsequent licensee investigation revealed that the positioned linkage had become uncoupled due to a locking nut on the linkage not being tightened. The licensee initiated a WO to reconnect the positioned linkage to the proper configuration and calibrate CV-3490.

The licensee entered issue into their CAP, corrected the mechanical issues, and performed an extent-of-condition review.

Analysis

The inspectors determined that deficient maintenance practices associated with the work on the CV-3490 positioner; specifically, not ensuring locking nuts were in place to prevent the uncoupling of the positioner linkage and the bellows spring, was a performance deficiency because it was the result of the failure to meet a requirement or a standard; the cause was reasonably within the licensee's ability to foresee and correct; and should have been prevented. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having work practice components, and involving aspects associated with ensuring that supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported. [H.4(c)]

The inspectors screened the performance deficiency per IMC 0612, "Power Reactor Inspection Reports," Appendix B, and determined that the issue was more than minor because it impacted the configuration control attribute of the Initiating Events Cornerstone objective of limiting those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors applied IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," to this finding. The inspectors utilized Column 1 of the Table 4a worksheet to screen the finding. The inspectors answered 'No' to the question "does the finding contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available" and, therefore, the finding was screened to be of very low safety significance (Green).

Enforcement

The inspectors concluded that no violations of NRC requirements occurred. This is considered a finding of very low safety significance (**FIN 05000263/2011003-01, Poor Maintenance Practices Result in CV-3490 Failing Shut**). The licensee entered this issue into their corrective action program (CAPs 01287453; 01288446), corrected the mechanical issues, and performed an extent-of-condition review to look for similar maintenance issue.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- 11 standby liquid control pump (SBLC);
- 16 battery;
- 'E' safety relief valve (SRV) increased tailpipe temperatures; and
- forced outage-related operational decision making instructions (ODMIs).

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 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 four samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the following permanent plant modification:

- ECP 16861 (tritium mitigation modification).

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening 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 system(s). The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing

documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

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

b. Findings

No findings 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:

- SBGT system valve work post-maintenance testing (PMT) prior to fuel moves;
- hydrostatic testing of the RCS following outage work;
- 14A and 14B feedwater heater shell side initial service leak test;
- 15A and 15B feedwater heater shell side initial service leak test;
- SBLC refueling test; and
- main transformer back feed testing.

These activities were selected based upon the SSCs ability to impact 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 TSs, 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 six PM testing samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors continued their review of outage related activities for the RFO that began during the last inspection period on March 5, 2011. The licensee began startup of the reactor on May 22, 2011, and completed the RFO when they synched to the electric grid on May 25, 2011.

During this inspection period, the inspectors observed the following RFO related activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the outage safety plan (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;
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block ECCS suction strainers, and reactor physics testing;
- Licensee identification and resolution of problems related to RFO activities.

This inspection constituted one RFO sample as defined in IP 71111.20-05.

b. Findings

No findings were identified.

.2 Unresolved Item (URI) 05000263/2011002-02: Calculation of Work Hours during Fatigue Rule Implementation

a. Inspection Scope

The inspectors reviewed additional documentation and observed portions of site shift turnover in order to determine whether the station was appropriately recording fatigue rule work hours. The inspectors also reviewed actual minimum break times, self-declarations of fatigue, and for-cause fatigue assessments for shift personnel in order to determine potential impacts of incorrect work hour calculations. The inspectors requested Regional review in order to appropriately disposition the resulting data collected during inspection activities.

b. Observations

In March 2011, while reviewing the site's control of work hours during the extended outage, the inspectors observed a CAP document citing work hour violations by one onsite department that was attributed to human error during the process of inputting employee time into the site's work hours program. As a corrective action for the errors, the site created an action to "add a 30 minute fixed secondary turnover period if period meets applicable regulatory and industry requirements and guidelines." The completion notes for this action, dated October 15, 2010, stated:

> "NEI 06-11, Managing Personnel Fatigue at Nuclear Power Reactor Sites, allows licensees to exclude either the oncoming or off going shift turnover, but not both, from the calculation of break time between successive work periods.
> For Monticello covered workers [in the affected department], 15 minutes of oncoming shift turnover activity is discounted from the calculation of break times. All time past 0600 or 1800 for off going shifts is considered secondary turnover time and entered specific for each worker to the time work activity ends.
> The corrective action proposal is to eliminate the need to enter the specific off going turnover time by assuming a 30 minute off going turnover time for each [member of this department].
>
> Because some of this off going secondary turn over time period may include actual shift work due to variability in oncoming crew arrival times, the practice of assigning a fixed secondary turnover period would not consistently cover duties meeting regulatory and NEI 06-11 definitions of qualifying turnover activity.
> An acceptable alternative means eliminating the need to update secondary turnover periods on an individual basis is to build the off going turnover into the [worker] schedules and set an acceptable period for oncoming [worker] turnover time. An acceptable alternative would be to establish a 45 minute fixed primary turnover with a 0630 and 1830 12-hour shift start time with any time beyond 12-hours entered as work time. ITAR01254318 has been initiated to track completion of this acceptable alternative."

The inspectors noted that this action, which extended the workers' oncoming turnover time by 30 minutes, was created with the purpose of decreasing the number of times workers' hours would have to be inputted into the work hours system to decrease the opportunities for human error, rather than having the purpose of correctly capturing the amount of time the workers were actually spending on oncoming shift turnover.

For example, it may have been reasonable to extend the turnover time by 30 minutes because the site had identified that oncoming turnover was taking much longer than they were giving credit for. Instead, the focus of this action was to create a fixed buffer on the front end of the schedule to allow for off going turnover time to be built into the shift. Note that prior to the implementation of this corrective action, 15 minutes of oncoming shift turnover was excluded from work hour calculations, the start of shift briefs began at 5:45 a/pm, and the shifts began at 6:00a/pm daily (12 hours shifts). Secondary turnover time was input on an individual basis, and added onto the back end of the shift. Following corrective action implementation, shift briefs began at 5:45 a/pm, and the scheduled shift time began at 6:30a/pm and ended 12 hours later.

The inspectors discussed this corrective action with the affected department manager and other representatives. These individuals commented that the shift scheduling and turnover briefs remained the same before and after this corrective action was put in place. They noted that the only thing that changed was the way it was being counted. They also noted that when they were implementing the action, they had taken a closer look at the amount of time spent on primary shift turnover, and that they had determined that this took about 30 minutes for individuals reporting to the furthest work locations. They noted that shift turnover included such activities as the daily shift status brief (by the oncoming shift supervisor), walking to and from their initial work locations, additional activities specifically classified as turnover activities by NRC guidance, and position specific turnover (which occurred once they arrived at their work location, with the individual from the previous shift).

To more closely examine this issue, the inspectors observed turnover activities within the department. On April 13, 15, and 19, the inspector observed the morning oncoming shift turnover activities. On April 13, the inspector observed turnover for a worker that was reporting to the work location closest to the shift briefing room. On April 15, and 19, the inspector observed the turnover activities for individuals reporting to the two work locations furthest from the shift briefing room. At least two of these observations took place when a crew was returning to shift work following a few days off. The shift briefs began promptly at 5:45am daily, and on the days of these observations, the individuals being observed began their shift working duties at approximately 6:04am, 6:07am, and 6:08am. When questioned, two of the observed workers said that in general, they normally have completed turnover activities, and take shift at 6:00am or shortly thereafter. The inspectors also observed the sign in log for two of the observed work locations, and this log was consistent with the information the workers gave, with the majority of sign in times listed as 6:00a/pm. At the time of the observations, the licensee's work hour counting process began counting the shift hours for individuals from this department at 6:30am (or 6:30pm for the night shift). All time between 5:45a/pm and 6:30a/pm was completely excluded from the work hour and break time calculations, despite the fact that the workers appeared to be taking shift at around 6:05a/pm.

The inspectors determined that setting up a work hour calculating system that fails to accurately account for the period of time (up to 30 minutes) where the workers are performing shift activities following oncoming shift turnover was not consistent with the Fatigue Rule requirements. The inspectors noted that accounting for all hours with perfect precision may be nearly impossible; however, they determined that as a result of the implementation of this corrective action, the amount of time that was being inappropriately attributed to oncoming shift turnover was excessive. The inspectors noted that the reasoning cited by the licensee for why a fixed 30 minute secondary

turnover period could not be established should also have been valid for the implementation of an inappropriately lengthy primary turnover period. The inspectors determined that counting periods of time as shift turnover, when that time instead meets the definition of work activities (not turnover), did not meet the requirements of the rule, the licensee's procedures, and applicable guidance.

In addition, the inspectors noted that if for any reason individuals were held over, or needed to perform additional work activities, which extended their time past the 6:30a/pm end of shift time, then, the time at the beginning of shift which was inappropriately excluded had the potential to cause additional violations. Specifically, individuals could have challenged their required 10 hour minimum break period without recognizing the challenge. In this case, the inspectors did not identify any instances where the exclusion of this time on shift for individuals in this department caused additional violations of the minimum break requirements during the period between November 2010 and May 31, 2011. However, the inspectors observed that there was at least one individual that had been scheduled for an extended shift, which would have challenged the individual's required minimum 10 hour break period if oncoming shift time had been properly accounted for, except that further investigation revealed that the individual had left 40 minutes prior to the end of the recorded shift. The inspectors observed that failing to accurately account for oncoming shift time would leave the licensee vulnerable to additional minimum 10 hour break challenges.

Title 10 CFR 26.203(b)(2) requires, in part, that "…licensees shall develop, implement, and maintain procedures that—describe the process for implementing the controls required under § 26.205 for the individuals who are performing the duties listed in § 26.4(a)." The licensee's work hour control procedure, FP-S-CWH-01 "Calculating Work Hours," states that "the calculated work hours SHALL include all time performing duties for work, including all within-shift break times and rest periods during which there are no reasonable opportunities or accommodations appropriate for restorative sleep." This procedure also states, in the section entitled "Included in Work Hour Calculations," that "one period of shift turnover time, at the end of the shift, SHALL be included."

Contrary to this requirement, the licensee failed to include all time performing duties for work, by excluding activities that did not meet the definition of shift turnover from the calculation of work hours for workers in one department. Specifically, on April 13, 15, and 19, following conclusion of the morning oncoming shift turnover activities, the licensee counted time spent performing required shift duties by these individuals as shift turnover. On these days, individuals took shift at approximately 6:04am, 6:07am, and 6:08am, despite the fact that their scheduled shifts began at 6:30am, and ended at 6:30pm. These time periods were excluded from the calculations of work hours and breaks.

The inspectors determined that the failure to adhere to their work hour control procedures was a performance deficiency warranting further evaluation, because it was the result of the failure to meet a requirement; the cause was reasonably within the licensee's ability to foresee and correct, and should have been prevented. The inspectors concluded that the finding was not more than minor because it did not impact the cornerstone objectives.

Although the inspectors determined that the safety significance of the finding was minor, they concluded that it was a violation of regulatory requirements. The licensee entered this issue into their corrective action program and implemented corrective actions to more accurately reflect the time spent performing shift turnovers and shift duties.

Unresolved Item (URI) 05000263/2011002-02 is closed.

.3 <u>(Closed) Unresolved Item (URI) 05000263/2010002-01: Reactor Building Crane Design and Licensing Basis Issues</u>

a. <u>Inspection Scope</u>

During the 2010 Refueling and Other Outage Activities Inspection, in accordance with NRC's Operating Experience Smart Sample (OpESS) FY2007–03, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20," Revision 2, the inspectors identified an unresolved item involving the reactor building crane design and licensing basis. Specifically, the inspectors were concerned that the licensee had not properly evaluated an increase in capacity of the crane from 85 tons delineated in the licensing basis to 105 tons to accommodate a larger spent fuel shipping cask. Specifically:

- The applicable Safety Evaluation explicitly approved the crane for a maximum 85 tons lifted load without the need to analyze for a concurrent seismic event based on low probability. It was not clear whether that basis remained valid with the increase in capacity.
- The licensee's calculations with respect to the crane and reactor building support structure for the increased capacity used friction in a linear elastic analysis to reduce seismic load effects. The calculations had not established rail minimum yield strength of the trolley and bridge and credited friction resistance to resist sliding of the rail with respect to the rail restraint mechanism. It was not clear whether this analytical approach was appropriate.

Subsequently, the inspectors reviewed additional information provided by the licensee regarding design and licensing basis and consulted with technical staff from the office of Nuclear Reactor Regulation (NRR) to ascertain licensing basis and technical adequacy. Based on this review, the inspectors sufficiently resolved these concerns and consider URI 05000263/2010002-01 closed. Specifically, the inspectors did not identify any actions inconsistent with the licensing basis or technically inadequate with respect to these concerns. However, the inspectors indentified one additional issue described below.

b. <u>Findings</u>

<u>Introduction</u>

A Severity Level IV non-cited violation (NCV) of 10 CFR 50.71(e), "Periodic Update of the Final Safety Analysis Report," and an accompanying Green finding were identified by the inspectors for the licensee's failure to update the Updated Safety Analysis Report (USAR) with the cask maximum lift height restrictions imposed by NRC staff.

Description

The inspectors were concerned that the licensee had not adequately evaluated whether the plant licensing basis necessitated retention of cask lift height restrictions when transitioning from the use of the 25 ton NFS-4 or 25 ton NAC-1 cask and 70 ton spent fuel shipping cask IF-300 to the heavier 105 ton NUHOMS cask. These lift height restrictions were established in the following series of licensing documents.

- In the Atomic Energy Commission (AEC) Letter to NSP, "RE: License No. DPR-22," dated February 4, 1974, AEC staff identified in their review of the Monticello Final Safety Analysis Report (FSAR) that a postulated spent fuel shipping cask drop accident had not been addressed and evaluated. The AEC requested an analysis of possible damage that could occur to plant structures, systems and components important to safety as a result of a spent fuel shipping cask drop. The licensee performed a cask drop accident analysis and determined that the Monticello plant structures cannot withstand the impact of a dropped spent fuel shipping cask in all cases as described in NSP letter to NRC, "Response to Request for Additional Information Northern States Power Monticello Nuclear Generating Plant (MNGP) Docket No. 50-263," dated February 17, 1975. This analysis was based on the drop of a 70 ton IF-300 cask. As a result, the licensee's intention changed to compete a safety evaluation for the spent fuel cask handling using a lighter, two-element shipping cask instead of the IF-300 cask as stated in NSP letter to NRC, "Status Report on Plans for Off-Site Shipment of Spent Fuel," dated May 30, 1975.

- In NSP Letter to NRC, "Off Site Shipment of Spent Fuel," dated January 22, 1976, the licensee completed a safety evaluation which contained an engineering analysis of the Monticello reactor building structures for the effects of a 25 ton NFS-4 spent fuel cask drop. The analysis determined a maximum safe lift height of six inches above the operating floor at EL. 1027'-8" and six inches above EL. 1027'-8" for a cask drop over the spent fuel pool storage pool slab at EL. 988'-11" with no loss in structural integrity for the operating floor and pool slab. However, the cask drop analysis over the equipment hatch at EL. 935'-0" indicated a loss of structural integrity of the concrete slab.

- The licensee provided additional information to NRC staff in NSP letter to NRC, "Off-Site Shipment of Spent Fuel," dated October 27, 1976, to address the cask drop analysis in the equipment hatch area. This letter stated "Our evaluation of the cask drop in the equipment hatch also shows that the effects on the health and safety of the public would be negligible. Under worst case conditions, however, it is conceivable that the torus and RHR [residual heat removal] systems could be damaged sufficiently to prevent their use in the ensuing plant shutdown and cool-down. We have determined that a complete shutdown and cool-down can be completed without the torus and RHR systems."

- The NRC provided a safety evaluation in NRC letter to NSP, "Safety Evaluation by the Office of Nuclear Reactor Regulation [NRR] Supporting Use of 25 ton Spent Fuel Shipping Casks NFS-4 and NAC-1 Northern States Power Company MNGP Docket No. 50-263," dated January 25, 1977, which stated "based on our review of the analyses and descriptive information provided by NSP, as discussed above, we have

concluded that the provisions for preventing postulated spent fuel shipping cask accidents at the Monticello Nuclear Generating Plant are acceptable and that the results of such postulated accidents have been shown to be acceptable. We, therefore, conclude that NSP's interim use of the NFS-4 and NAC-1 25 ton, two element spent fuel shipping casks at the Monticello Nuclear Generating Plant is acceptable."

- Also, during the 1976 timeframe the licensee submitted a reactor building crane safety evaluation as documented in NSP letter to NRC, "Design Report for Redundant Reactor Building Crane," dated November 22, 1976, to seek review and acceptance of the reactor building crane as single failure proof in accordance with Regulatory Guide (RG) 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants," dated February 1976. This crane would be used to transport either the 70 ton IF-300 cask or the 25 ton NFS-4 or NAC-1 cask. The licensee took several exceptions to the regulatory position requirements of RG 1.104, one of which was RG position C.1.c. Regulatory Guide position C.1.c stated "the crane should be classified as Seismic Category I and should be capable of retaining the maximum design load during a safe shutdown earthquake." The licensee provided their basis for the exception on page 3-8 of the November 22, 1976, letter, as well as in NSP letter to NRC, "Response to Request for Additional Information," dated February 28, 1977.

- NSP letter to NRC, "Design Report for Redundant Reactor Building Crane," dated November 22, 1976, page 3-8 stated "The requirement to reclassify the crane as Seismic Category I is not considered practical for a licensed operating plant. To reclassify cranes previously licensed as Seismic Class II would require extensive modifications to the bridge track-way and new bridge. It is considered sufficient to analyze the new trolley in a manner consistent with the design codes and loading conditions applicable at the time of the original installation." Northern States Power letter to NRC, "Response to Request for Additional Information," dated February 28, 1977, stated "The method used in the evaluation are in accordance with the original design criteria for Monticello, which is keeping with the statements on page 3-8 of our November 22, 1976, submittal. The design codes and loading conditions applicable at the time of the original installation did not include the lifted load in the seismic analysis because of the extremely low probability of both events occurring simultaneously."

- NRC letter to NSP, "Safety Evaluation by the Office of Nuclear Reactor Regulation [NRR] Supporting Approval of Crane Modification and Use of 70 Ton Spent Fuel Shipping Cask IF-300," dated May 19, 1977; NRC Safety Evaluation established the reactor building crane as single failure proof in accordance with NRC guidance with specific exceptions taken by the licensee. The Safety Evaluation Report stated "We find that NSP's proposed modifications to the reactor building crane have incorporated all the provisions of the draft Regulatory Guide 1.104 that are practical for the Monticello design." The reactor building crane was reviewed and accepted with a specific exception that the crane would not be design and licensed to hold a lifted load concurrent with a seismic event. However, NRC staff provided the following conditional requirements in the May 19, 1977 letter, which stated "The carry height of the IF-300 70 ton cask shall be administratively limited to a maximum of the

minimum height necessary to gain floor clearance during cask swing plus two inches and the carry height of the NFS-4 and NAC-1 casks approved for use in our January 25, 1977 letter, shall be limited to a maximum of six inches."

- The licensee agreed to the conditional requirements in NSP letter to NRC, "Reactor Building Redundant Crane," dated June 24, 1977.

The inspectors noted that the licensee had incorporated these maximum lift height restrictions in the plant procedures for the 70 ton cask to limit plant damage as required by NRC staff in case of a cask drop accident. However, maximum lift height restrictions had not been later carried forward to the procedures for the heavier 105 ton NUHOMS cask. These restrictions established in the licensing basis had not been incorporated in the USAR. Therefore, removal of the lift height restrictions from the plant procedures when transitioning to the heavier cask was not identified as a change that needed to be evaluated in accordance with 10 CFR 50.59. Specifically, the licensee transported ten 105 ton NUHOMS casks from September 8, 2008, through December 12, 2008. The licensee performed 50.59 screenings SCR 06-0059, dated February 9, 2007, and SCR 08-0105, dated May 7, 2008, to support the cask movement but did not evaluate whether or not a maximum cask lift height restriction for transport of the 105 ton NUHOMS cask inside the Monticello reactor building facility was required.

The licensee was not able to provide sufficient technical rationale why restrictions specifically added by the NRC to the licensing basis to establish an adequate factor of safety would not logically apply to even heavier casks.

Although not specifically focused on the maximum lift height restrictions, the inspectors noted that the licensee had developed Calculation Nos. CA-05-101, "Evaluation of Reactor Steel Superstructure for 105 Ton Reactor Building Crane," Revision 3A; CA-05-103, "Reactor Building Superstructure Seismic Response Analysis with 105 Ton Crane," Revision 0A; and CA-05-107, "Structural Seismic Qualification Reactor Building Crane Upgrade for ISFSI," Revision 0B. These calculations were inferred to show the crane can withstand loading conditions beyond its design basis requirements. While not required by the licensee's licensing and design basis, the licensee partially employed seismic analysis methods delineated in American Society of Mechanical Engineers (ASME) NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," 2004. The ASME NOG-1-2004 was endorsed by the NRC per Regulatory Issue Summary 2005-25, Supplement 1, "Clarification of NRC Guidelines for Control of Heavy Loads," as an acceptable method for satisfying the guidelines of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." However, the inspectors noted that the licensee used the assumption that sliding would occur at the crane rail/wheel interface, thus limiting the applied loads to the frictional forces. This assumption was inconsistent with the boundary condition requirements stipulated in Section 4153.6 of ASME NOG-1-2004. Hence, these calculations have no bearing on the licensing basis maximum lift height restrictions.

<u>Analysis</u>

The inspectors determined that the failure to update the USAR with the cask lift height restrictions for the 25 ton and 70 ton spent fuel cask was contrary to 10 CFR 50.71(e) and was a performance deficiency warranting a significance evaluation.

Violations of 10 CFR 50.71 (e) are dispositioned using the traditional enforcement process instead of the SDP because they are considered to be violations that potentially impede or impact the regulatory process. However, if possible, the underlying finding is evaluated under the Significance Determination Process (SDP) to determine the significance of the violation. The finding was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," because, if left uncorrected, the performance deficiency could have led to a more significant safety concern. Specifically, the inspectors could not readily conclude that the absence of lift height limitations would not require additional calculational analyses and/or require a license amendment. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Attachment 4, "Initial Screening and Characterization of Findings." In accordance with Table 3b, "SDP Phase 1 Screening Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones," the inspectors determined that all three Cornerstones could be affected but the cornerstone best reflecting the dominant risk was the Barrier Integrity Cornerstone for Spent Fuel Pool Issues. In accordance with Table 4b, "Seismic, Flooding, or Severe Weather Screening Criteria," the finding screened as potentially risk significant due to external initiating event core damage sequences. Therefore, the Region III Senior Reactor Analyst (SRA) performed an SDP Phase 3 risk-assessment of this performance deficiency.

The inspectors determined that only seismic events exceeding the level of an Operational Basis Earthquake (OBE) of 0.03g could impact core damage frequency (CDF). The licensee supplied information that the median annual probability of exceedance of peak ground acceleration for the OBE at Monticello was approximately 7.0E-4/yr. In addition, the predicted shipping cask lifts was 19.2/yr with an average lift duration of 30 minutes. Thus the frequency of exceeding the OBE while lifting a shipping cask was estimated to be 7.7E-7/year. This value is a bounding frequency estimate for delta- CDF in that it does not imply certainty that there will be a cask drop during an earthquake nor does it imply certainty of core damage during an earthquake given a cask drop. The SRA concluded that the risk due to simultaneous occurrence of an OBE or greater seismic event during use of the reactor building crane for shipping cask lifts was best characterized as very low (Green).

In accordance with Section 6.1.d.3 of Section D.5 of Supplement I to the NRC Enforcement Policy, this violation is categorized as Severity Level IV because the resulting changes were evaluated by the SDP as having very low safety significance (Green).

The inspectors determined that this finding did not reflect current performance because it was a legacy issue with the failure to properly update the USAR occurring almost 30 years ago; therefore, there was no cross-cutting aspect associated with this finding.

<u>Enforcement</u>

Title 10 CFR 50.71(e) required that licensees shall periodically update the FSAR, originally submitted as part of the application for the operating license, to assure that the information included in the report contains the latest information developed. This submittal shall include the effects of all the changes necessary to reflect information and analysis submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submittal of the original FSAR, or as

appropriate, the last update to the FSAR under this section. Contrary to the above, from July 22, 1982, the licensee failed to update the USAR with the cask lift height restrictions for the 25 ton and 70 ton casks. Specifically, the licensee failure to update the USAR impacts the licensee's ability to adequately evaluate plant changes under the 10 CFR 50.59 processes and could lead to the licensee's erroneously making unacceptable changes to the facility. In this case, a 105 ton spent fuel cask drop from any lift height inside the Monticello reactor building facility had not been evaluated to establish whether or not a maximum cask lift height restriction was required in accordance with the licensing basis.

In accordance with the Enforcement Policy, the violation was classified as Severity Level IV because the underlying technical issue was of very low safety significance. Because this violation was of a very low safety significance, was not repetitive or willful, and was entered into the licensee's CAP as AR 001290843; "Spent Fuel Cask Height Restriction is not in USAR," dated June 16, 2011, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000263/2011003-02, Failure to Update USAR for Cask Lift Height Restriction)**.

This finding is evaluated separately from the traditional enforcement violation; therefore, the finding is being assigned a separate tracking number **(FIN 05000263/2011003-03, Failure to Update USAR for Cask Lift Height Restriction)**.

.4 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on June 24, 2011, and continued through the June 28, 2011. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, personnel fatigue management, startup and heatup activities, and identification and resolution of problems associated with the outage. The reason for the forced outage was to address an emergent condition associated with increasing tailpipe temperature downstream of 'E' safety relief valve. During the outage, the licensee replaced the leaking safety relief valve. Subsequent to plant startup, the licensee successfully tested the relief valve.

This inspection constituted one other outage sample as defined in IP 71111.20-05.

b. Findings

No findings 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:

- low pressure ECCS automatic initiation and loss of auxiliary power test (routine);
- secondary containment capability test (routine);
- scram discharge volume hi level scram test (routine);
- reactor coolant drywell leak rate check (reactor coolant system (RCS) leakage); and
- control rod drive (CRD) scram insertion time test (inservice testing (IST)).

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 was 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 IST 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 RCS leakage detection sample, and one IST sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted a partial sample as defined in IP 71124.01-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of radiation protection program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there have been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors evaluated whether the licensee assessed the potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed the last two radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys were appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation:

- repair/machine main steam isolation valve (MSIV)parts;
- reactor cavity decontamination and dryer/separator pit decontamination; and
- investigate, repair inboard MSIV.

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if hazards were properly identified, including the following:

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials (This evaluation may include licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel.);
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected various containers holding non-exempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, "Labeling Containers," or met the requirements of 10 CFR 20.1905(g), "Exemptions To Labeling Requirements."

The inspectors reviewed the following radiation work permits used to access high radiation areas and evaluated the specified work control instructions or control barriers.

- RWP 0676, reactor disassembly/reassembly and associated activities, Revision 8;
- RWP 0679, reactor cavity decontamination and dryer/separator pit decontamination, Revision 9; and
- RWP 0708, general drywell in-service inspections, Revision 3.

For these radiation work permits, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each radiation work permit were clearly identified. The inspectors evaluated whether electronic personal dosimeter alarm set-points were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker's electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the CAP and dose evaluations were conducted as appropriate.

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee's means to inform workers of changes that could significantly impact their occupational dose.

b. <u>Findings</u>

No findings were identified.

.4 <u>Radiological Hazards Control and Work Coverage</u> (02.05)

a. <u>Inspection Scope</u>

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, radiation work permits, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of electronic personal dosimeters in high noise areas as high radiation area monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors reviewed the following radiation work permits for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- RWP 0676, reactor disassembly/reassembly and associated activities, Revision 8;
- RWP 0679, reactor cavity decontamination and dryer/separator pit decontamination, Revision 9
- RWP 1272, inboard MSIV associated work, Revision 0.

For these radiation work permits, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, entry into tanks, cubicles, and reactor cavities).
The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air ventilation system operation.

b. Findings

No findings were identified.

.5 Risk Significant High-Radiation-Area and Very-High-Radiation-Area Controls (02.06)

a. Inspection Scope

The inspectors discussed with the radiation protection manager the controls and procedures for high-risk-high-radiation areas and very high radiation areas.
The inspectors discussed methods employed by the licensee to provide stricter control of very-high-radiation-area access as specified in 10 CFR 20.1602, "Control of Access to Very-High-Radiation-Areas," and Regulatory Guide 8.38, "Control of Access to High and Very-High-Radiation-Areas of Nuclear Plants." The inspectors assessed whether any changes to licensee procedures substantially reduce the effectiveness and level of worker protection.

The inspectors discussed the controls in place for special areas that have the potential to become very high radiation areas during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication beforehand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

The inspectors evaluated licensee controls for very high radiation areas and areas with the potential to become very high radiation areas to ensure that an individual was not able to gain unauthorized access to the very high radiation area.

b. Findings

No findings were identified.

.6 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors assessed whether workers were aware of the radiological conditions in their workplace and the radiation work permit controls/limits

in place, and whether their performance reflected the level of radiological hazards present.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions planned or taken.

b. Findings

No findings were identified.

.7 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technicians with respect to all radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the radiation work permit controls/limits, and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings were identified.

.8 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings were identified.

2RS2 Occupational As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71124.02)

This inspection constituted a partial sample as defined in IP 71124.02-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges. The inspectors reviewed the plant's three-year rolling average collective exposure.

The inspectors reviewed the site-specific trends in collective exposures (using NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," and plant historical data) and source term (average contact dose rate with reactor coolant piping) measurements (using Electric Power Research Institute (EPRI) TR-108737, "BWR Iron Control Monitoring Interim Report," issued December 1998, and/or plant historical data, when available).

The inspectors reviewed site-specific procedures associated with maintaining occupational exposures ALARA, which included a review of processes used to estimate and track exposures from specific work activities.

b. Findings

No findings were identified.

.2 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors selected the following work activities of the highest exposure significance.

- repair inboard MSIV; and
- replacement of condensate demineralizers.

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined whether the licensee reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors assessed whether the licensee's planning identified appropriate dose mitigation features; considered alternate mitigation features; and defined reasonable dose goals. The inspectors evaluated whether the licensee's ALARA assessment had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment (e.g., ice vests). The inspectors determined whether the licensee's work planning considered the use of remote technologies (e.g., teledosimetry, remote visual monitoring, and robotics) as a means to reduce dose and the use of dose reduction insights from industry operating experience

and plant-specific lessons learned. The inspectors assessed the integration of ALARA requirements into work procedure and radiation work permit documents.

The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in the licensee's ALARA planning for these work activities. The inspectors compared the person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements, and evaluated the accuracy of these time estimates. The inspectors assessed the reasons (e.g., failure to adequately plan the activity, failure to provide sufficient work controls) for any inconsistencies between intended and actual work activity doses.

b. Findings

Failure to Maintain Radiation Exposure ALARA During Inboard Main Steam Isolation Valve Repair

Introduction

A finding of very low safety significance (Green) was self-revealed due to the licensee having unplanned, unintended occupational collective dose because of deficiencies in the licensee's ALARA planning and work control program. Specifically, the licensee failed to properly incorporate ALARA strategies or insights while planning and executing a repair of the 'C' inboard (C-MSIV). This issue resulted in the expansion of collective exposure for this work from 4.044 person-rem to 9.654 person-rem.

Description

Prior to the 2011 RFO, the licensee determined that the C-MSIV valve seat was in need of repair. This work activity was anticipated and planned as part of the licensee's pre-outage preparation. A work order (WO 319048) was assigned and a Radiation Work Permit (RWP 1272) was developed based on the ALARA plan. During the initial valve seat repair, the licensee identified base metal intrusion into the valve's seat material. This issue added scope to the repair. This scope increase involved grinding out the stellite seat material down to the base metal and replacing the valve seat. The licensee expanded the initial dose estimate of 3.186 person-rem to 4.044 person-rem based on an estimate of the hours for the increase in work scope.

Shortly after the licensee approved the dose increase to 4.044 person-rem, the licensee discovered that the base metal would have to be removed in addition to the valve seat. Subsequently, the work required two additional dose additions of 1.931 and 4.565 person-rem respectively. These dose additions were necessary because the work estimates submitted by the crew assumed a level of worker efficiency and equipment operability that could not be met. To help complete this work activity, additional welding personnel were rushed to the site. These workers were put to work under "escorted status." The decision to change the status of these maintenance workers from an "escorted" to a non-escorted status was not made when the licensee recognized that the work would take longer than originally estimated (from ~2 days to ~7 days) and the escorts accumulated the additional project dose. This decision alone resulted in 1.3 person-rem of additional dose, because the escorts needed to be in the work area to maintain visual contact with the escorted worker. The inspectors reviewed several "Work in Progress" reviews and ALARA committee meeting notes and noted a lack of

detail in radiological assessments related to dose reducing strategies. The licensee entered this issue into their CAP as AR 1281395.

Analysis

The failure to appropriately plan and coordinate outage activities, together with the failure to properly incorporate ALARA strategies or insights while planning and executing the repair of the C-MSIV seat was a performance deficiency that was within the licensee's ability to control and prevent. The finding was more than minor because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone. Moreover, the finding affected the cornerstone objective of ensuring the adequate protection of worker's health and safety from exposure to radiation from radioactive materials during routine civilian nuclear reactor operations. Additionally, the finding was similar to example 6.i in Appendix E of IMC 0612, in that, it resulted in collective exposure of greater than 5 person-rem and exceeded the outage goal by greater than 50 percent. The inspectors determined that the finding was not discovered through a licensee program or process that stopped the work and reassessed the maintenance activity, to incorporate dose reducing strategies and methodologies based on the review of records provided by the licensee. The finding is not subject to traditional enforcement because it did not affect the regulatory process or result in actual safety consequences. The inspectors evaluated the significance of this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The finding was determined to be of very low safety significance (Green) because MNGP's current three-year rolling average collective dose is 135.266 person-rem, less than 240 person-rem per unit. This finding had a cross-cutting aspect in the area of Human Performance related to the cross-cutting component of work control, in that the outage plan did not adequately incorporate actions to address the impact of work on different job activities. (H.3(b))

Enforcement

No violation of regulatory requirements occurred. The ALARA rule (10 CFR 20.1101) Statements of Consideration indicate that compliance with the ALARA requirement will be judged on whether the licensee has incorporated measures to track and, if necessary, to reduce exposures, and not whether exposures and doses represent an absolute minimum or whether the licensee has used all possible methods to reduce exposures. The overall exposure performance of a nuclear power plant is used to determine its compliance with the ALARA rule. The licensee has entered this issue into their CAP as AR 1281395. Since MNGP's three-year rolling average is 135.266 person-rem, below a three-year rolling average of 240 person-rem per unit and has an established ALARA program to reduce exposure consistent with the 10 CFR 20.1101 Statements of Consideration, no violation of 10 CFR Part 20.1101 (b) is considered. Because this finding does not involve a violation and has very low safety significance it is identified as **FIN 05000263/2011003-04, Failure to Maintain Radiation Exposure ALARA During Inboard Main Steam Isolation Valve Repair**.

.3 Radiation Worker Performance (02.05)

a. Inspection Scope

The inspectors observed radiation worker and radiation protection technician performance during work activities being performed in radiation areas, airborne radioactivity areas, or high radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice (e.g., workers are familiar with the work activity scope and tools to be used, workers used ALARA low-dose waiting areas) and whether there were any procedure compliance issues (e.g., workers are not complying with work activity controls). The inspectors observed radiation worker performance to assess whether the training and skill level was sufficient with respect to the radiological hazards and the work involved.

b. Findings

No findings were identified.

.4 Problem Identification and Resolution (02.06)

a. Inspection Scope

The inspectors evaluated whether problems associated with ALARA planning and controls are being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures PI for the period from the 1st Quarter 2010 through the 1st Quarter 2011. 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 6, dated October 2009, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance, were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance WOs, issue reports, event reports and NRC Integrated Inspection Reports for the period of the 1st Quarter 2010 through the 1st Quarter 2011 to validate the accuracy of the submittals.
The inspectors also reviewed the licensee's issue report database to determine if any

problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one safety system functional failures sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage PI for the period from the 2nd Quarter 2010 through the 2nd Quarter 2011. 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 6, dated October 2009, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated Inspection Reports for the period of the 2nd Quarter 2010 through the 2nd Quarter 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RCS leakage sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Unresolved Item (URI) 05000263/2009005-03: MSPI Basis Document does not Reflect Current Plant Configuration for RHRSW

During an inspection conducted in the fourth quarter 2009, the inspectors identified that the licensee's Mitigating Systems Performance Indicator (MSPI) basis document description of the RHRSW system did not match the actual in-plant configuration. An engineering modification installed during the spring 2009 RFO changed the method of how cooling water was delivered to the RHRSW motor coolers. The inspectors noted that the modification impacted RHRSW monitored components and PRA event sequences assessed in the MSPI basis document which were used to evaluate unavailability and unreliability data. At the time the issue was identified by the inspectors, the licensee evaluated the modification's impact on their existing PRA analysis and determined the impact to be minimal.

The licensee entered this issue into their corrective action process and evaluated the existing guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, associated with the required periodicity of MSPI basis document changes subsequent to plant modifications which impact monitored components. Since the existing revision of 99-02 did not provide explicit guidance on

this topic, the issue was resolved via the Frequently Asked Question (FAQ) process. As a result of this process, FAQ 477 was developed and will become effective October 1, 2011. FAQ 477 clarifies NEI 99-02, Section 2.2, to provide details on when MSPI basis changes are required and the level of detail required to be supplied via comments in the consolidated data entry (CDE) when changes to either the basis document or CDE are made. Specifically, the FAQ requires, in part, that modifications to the plant design that result in a change to segment or train boundaries, monitored components, or affect monitored functions or success criteria, shall be reflected in the MSPI basis document the quarter following the completed implementation.

The licensee is aware of the new guidance and has corrective actions in place to ensure that the appropriate MSPI basis document changes are made in accordance with the timelines specified in FAQ 477.

Unresolved Item (URI) 05000263/2009005-03 is closed.

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. Inspection 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: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; 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 Attachment to this report.

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 were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection 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 were identified.

.3 Selected Issue Follow-Up Inspection: Outstanding Operable but Degraded/Non-Conforming Issues Still Existing at Time of Plant Restart

a. Inspection Scope

As an extension of their plant restart readiness inspection activities, the inspectors reviewed outstanding CAP items that were required to be evaluated and addressed prior to mode changes and plant startup. These items included the status of high energy line break (HELB) barriers; operator workarounds (OWAs) and burdens; mode change restraints; and conditions of plant equipment or regulatory noncompliance where the licensee determined that an operable but degraded (OBD) or nonconforming (OBN) condition existed. The primary focus of the inspectors during this inspection was the evaluation of the eight OBN issues that existed at the time of plant restart. Specifically, the inspectors reviewed: 1) the evaluation associated with each issue which supported the OBN classification; 2) the licensee's proposed corrective actions associated with each issue; and 3) the timeliness of each corrective action to ensure they were commensurate with the safety significance of the noncompliance. No issues of significance were identified by the inspectors during this inspection.

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

b. Findings

No findings were identified.

.4 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors'

review nominally considered the six month period of 1st and 2nd quarters of 2011, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted a single semi-annual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

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

.1 Unexpected Recirculation Pump Runback

a. Inspection Scope

The inspectors reviewed operator and plant response to an unexpected recirculation pump runback, which occurred during planned testing associated with reactor plant power ascension, on May 27, 2011.

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

b. Findings

Introduction

A finding of very low safety significance and NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when an unexpected recirculation pump runback occurred during the performance of reactor dynamics testing. The event was the result of the licensee failing to adequately assess the operational impact of a recent revision to Procedure C.1, "Startup Procedure," which resulted in operating the plant in a manner that challenged feedwater pump protective features.

Description

Prior to startup from their recent RFO, the licensee decided to implement a new operational strategy to maximize the plant's hydrogen water chemistry (HWC) availability. This strategy was documented in Procedure C.1, "Startup Procedure." The basic tenants of the strategy applicable to this discussion, included: placing HWC in service at approximately 35 percent reactor power; raising reactor power to approximately 55 percent; perform high pressure coolant injection (HPCI), RCIC and reactor dynamic testing; securing HWC; starting the second reactor feedwater pump;

placing HWC back in service; and continuing with power ascension. Historically, the second RFP was placed in service at approximately 50 percent reactor power and HWC was not placed in service until approximately 70 percent reactor power.

As a protective feature for the RFPs, an automatic runback function associated with the recirculation pumps is enabled when the total steam flow signal reaches 55.4 percent. With the runback permissive enabled, coincident with a tripped condensate pump or RFP, the recirculation pumps runback to approximately 50 percent speed. On May 27, 2011, the control room operators commenced Procedure 1076, "Reactor Dymanics Test Procedure." At the time the test was initiated, reactor power was at approximately 55 percent and only one RFP was in service. Subsequent to the performance of Step 5.c, which directs the operators to "decrease the MPR [mechanical pressure regulator] setpoint about 5 psi by turning MPR setpoint handwheel about ¼ turn as rapidly as possible," recirculation flow automatically reduced approximately 2 Mlb/hr with a corresponding decrease in reactor power of approximately 30 MWth.

Further evaluation of plant data revealed that since the steam flow was initially at approximately 55 percent flow, decreasing the MPR setpoint by about 5 psi, and the resulting opening of the turbine control valves, was sufficient to increase total steam flow to a level which enabled the runback permissive. Since only one RFP was operating at the time the runback permissive was enabled, the runback logic was satisfied and the recirculation pumps reduced speed to approximately 50 percent. The recently revised C.1 procedure did not recognize that the new power ascension and testing sequence placed the plant in a condition that would challenge the RFP/recirculation runback feature, nor were the control room operators aware that the performance of the reactor dynamics testing, given the existing plant conditions at the time, would satisfy all of the conditions necessary for the runback to occur.

The licensee entered this issue into their corrective action program (CAP 01288070) and initiated appropriate corrective actions to address the issue.

<u>Analysis</u>

The inspectors determined that the licensee's failure to adequately assess the operational impact of a revision to their C.1 plant startup procedure was a performance deficiency because it was the result of the failure to meet a requirement or a standard, the cause was reasonably within the licensee's ability to foresee and correct, and should have been prevented. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency was associated with the cross-cutting area of Human Performance, having decision-making components, and involving aspects associated with the licensee conducting effectiveness reviews of safety-significant decisions to verify the validity of the underlying assumptions, identify possible unintended consequences, and determine how to improve future decisions. [H.1(b)]

The inspectors screened the performance deficiency per IMC 0612, "Power Reactor Inspection Reports," Appendix B, and determined that the issue was more than minor because it impacted the procedure quality attribute of the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors applied IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of

Findings," to this finding. The inspectors utilized Column 1 of the Table 4a worksheet to screen the finding. The inspectors answered no to the questions associated with LOCA initiators, transient initiators, and external events initiator, and screened finding to be of very low safety significance (Green).

Enforcement

Title 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented procedures, of a type appropriate to the circumstances, and shall be accomplished in accordance with these procedures. Contrary to this requirement, the licensee failed to adequately assess the operational impact of the revision to their C.1 startup procedure, in regards to challenging RFP protective features, prior to the first use of the revised procedure. This failure resulted in an unexpected recirculation pump runback during the performance of reactor dynamic testing that was being performed as part of normal reactor plant startup testing. Because the violation was of very low safety significance and was entered into the licensee's corrective action program (CAPs 01288070), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000263/2011003-05, Inadequate C.1 Startup Procedure Review)**.

.2 Turbine Roll, Generator Synch, and Main Transformer Loading Following Major Outage Work and Modification Activities

a. Inspection Scope

On May 25, 2011, the inspectors observed the licensee's performance of a turbine roll followed by a generator synch and loading of the main transformer. During the extended outage, the licensee had performed extensive work scopes on both the generator and the main transformer. Specifically, the licensee had completed a generator rewind, which involved removal and reinstallation of the generator rotor. In addition, the licensee had replaced in its entirety the previous main transformer, which had been in place for the life of the plant. Prior to this evolution, the new main transformer had never been loaded. During the evolution, the inspectors observed operator actions and communications, turbine generator testing, and equipment response. As part of this inspection, the inspectors also attended the infrequently performed test evolution brief, and monitored the operators' performance of selected testing activities. Documents reviewed in this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.3 Loss of Reactor Water Clean Up Dump Flow Path while Being Used to Control Reactor Pressure Vessel Level

a. Inspection Scope

The inspectors observed the licensee's response to a loss of reactor water clean up (RWCU) dump flow path on May 5, 2011. While coordinating ECCS testing and various other outage work activities, the operators responded to indications that the RWCU dump flow had isolated. At the time, the dump flow path was being used to remove excess water from the RCS in order to maintain reactor water level. Following dump flow path isolation, reactor water level began to steadily rise due to the CRD pump continuing to supply flow, and in response, operators removed the CRD pump from service. The primary focus of the inspectors during the operators' response to the unexpected isolation was evaluation of the operating crew's ability to assess plant indications and take actions to place the plant in a safe configuration. Ultimately, the licensee attributed the cause of the isolation to a spurious RWCU temperature instrument spike, and following a review of operability requirements, the instrument was bypassed for the remainder of the outage. Documents reviewed in this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.4 'E' Safety Relief Valve Elevated Tailpipe Temperatures

a. Inspection Scope

The inspectors observed the licensee's response to elevated tailpipe temperatures associated with 'E' SRV, occurring between May 25, 2011 and June 24, 2011, which ultimately resulted in the decision to shutdown the plant and repair the leaking SRV. As part of this inspection, the inspectors evaluated the licensee's efforts to monitor, trend, and evaluate the source of the leakage. Additionally, the inspectors reviewed operational decision making documentation associated with continued plant operation with the leaking SRV, evaluated the risk associated with continued plant operation with increasing tailpipe temperatures, attended licensee meetings in which potential courses of action were discussed, continued to monitor the SRV during the plant shutdown.

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

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Closed) Temporary Instruction 2515/179, "Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)"

a. Inspection Scope

The inspectors confirmed that the licensee has reported the initial inventories of sealed sources pursuant to 10 CFR 20.2207 and verified that the National Source Tracking System database correctly reflects the Category 1 and 2 sealed sources in custody of the licensee. Inspectors interviewed personnel and performed the following:

- Reviewed the licensee's source inventory;
- Verified the presence of any Category 1 or 2 sources;
- Reviewed procedures for and evaluated the effectiveness of storage and handling of sources;
- Reviewed documents involving transactions of sources; and
- Reviewed adequacy of licensee maintenance, posting, and labeling of nationally tracked sources.

b. Findings

No findings were identified.

.2 (Closed) NRC Temporary Instruction (TI) 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event"

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi Nuclear Plant fuel damage event. This included: (1) an assessment of the licensee's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order, Section B.5.b, issued February 25, 2002, as committed to in severe accident management guidelines (SAMGs), and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63 and station design bases; (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

Inspection Report 05000263/2011009 (ML111320400) documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted detailed follow-up on selected issues.

.3 <u>(Closed) NRC Temporary Instruction (TI) 2515/184, “Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)”</u>

On May 20, 2011, the inspectors completed a review of the licensee’s SAMGs, implemented as a voluntary industry initiative in the 1990’s, to determine: (1) whether the SAMGs were available and updated; (2) whether the licensee had procedures and processes in place to control and update its SAMGs; (3) the nature and extent of the licensee’s training of personnel on the use of SAMGs; and (4) licensee personnel’s familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for MNGP were provided as an Enclosure to a Memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated June 1, 2011, (ML111520396).

.4 <u>(Closed) Corrective Action Document for an Adverse Trend in LLRT Results (Unresolved Item (URI) 05000263/2010007-01)</u>

During a Problem Identification and Resolution (PI&R) inspection documented in Inspection Report 05000263/2010007, the inspectors questioned the adequacy of an Apparent Cause Evaluation (ACE) associated with CAP 1202466 for an adverse trend in double disc gate valve local leak rate testing (LLRT) performance. The inspectors questioned if the safety function of the HPCI, RCIC and RWCU steam supply valves to close after detection of a HELB should have been considered. The licensee responded that the ACE did not need to consider the effect of the valves’ increased leakage on the HELB analyses because any leakage would not impact the alternate shutdown path. The inspectors reviewed the assumptions and acceptance criteria of the HELB calculations for HPCI, RCIC and RWCU line breaks and identified potential inconsistencies between the calculations’ assumptions with TSs and UFSAR allowed values for valve closure times, incorporation of delay actuations, and isolation initiation signals. The licensee entered the NRC concerns with these potential inconsistencies into the CAP by initiating CAP 01252363 on October 1, 2010. The licensee stated that the calculations were appropriate and provided the inspectors with some original licensing documents for the HELB analyses; however, additional questions remained at the end of the PI&R inspection period so the URI was issued. Subsequent to the PI&R inspection, the inspectors reviewed the licensee responses, the HELB analyses, the original and current license bases. The inspectors determined that the steam line break design analyses were appropriate for the current licensing basis and the calculations had sufficient conservatisms. The inspectors also determined that the leakages through the double disc gate valve valves were insignificant compared to the steam assumed to be released during steam line break design scenarios. This URI is closed with no further action required.

4OA6 Management Meetings

.1 Exit Meeting Summary

On June 28, 2011, the inspectors presented the inspection results to Mr. T. 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:

- Radiological Hazard Assessment and Exposure Controls, Occupational As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls during Refueling Outage Activities, and Temporary Instruction 2515/179, "Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System" with Mr. T. O'Connor, Site Vice President, on May 13, 2011;
- The inspectors presented results of the followup inspection review pertaining to URI 05000263/2010002-01, Reactor Building Crane Design and Licensing Basis Issues, to the Site Vice President, Timothy O'Connor, and other members of the licensee's staff via telephone on June 22, 2011; and
- The inspectors presented the results of the followup inspection review pertaining to URI 05000263/2010007-01, Corrective Action Document for an Adverse Trend in LLRT Results, to Ms. S. Oswald on July 7, 2011.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. O'Connor, Site Vice President
J. Grubb, Plant Manager
W. Paulhardt, Operations Manager
N. Haskell, Site Engineering Director
K. Jepson, Assistant Plant Manager
S. Radebaugh, Maintenance Manager
M. Holmes, Chemistry Manager
A. Zelie, Radiation Protection Manager
P. Kissinger, Regulatory Affairs Manager
P. Anderson, Nuclear Licensing/Regulatory Affairs Corporate Director
S. Quiggle, Mechanical Design Engineering Supervisor
P. Young, Program Engineering Supervisor
R. Loeffler, Senior Licensing Engineer
D. Horgen, Performance Assessment
L. Taufen, Performance Assessment

Nuclear Regulatory Commission

K. Riemer, Chief, Reactor Projects Branch 2
A. M. Stone, Chief, Engineering Branch 2

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000263/2011003-01	FIN	Poor Maintenance Practices Result in CV-3490 Failing Shut (Section 1R13)
05000263/2011003-02	NCV	Failure to Update USAR for Cask Lift Height Restrictions (Section 1R20)
05000263/2011003-03	FIN	Failure to Update USAR for Cask Lift Height Restrictions (Section 1R20)
05000263/2011003-04	FIN	Failure to Maintain Radiation Exposure ALARA During Inboard Main Steam Isolation Valve Repair (Section 2RS2)
05000263/2011003-05	NCV	Inadequate C.1 Startup Procedure Review (Section 4OA3)

Closed

2515/183	TI	Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event (Section 4OA5)
2515/184	TI	Availability and Readiness Inspection of Severe Accident Management Guidelines (Section 4OA5)
2515/179	TI	Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)
05000263/2011002-02	URI	Calculation of Work Hours during Fatigue Rule Implementation (Section 1R20)
05000263/2010002-01	URI	Reactor Building Crane Design and Licensing Basis Issues (Section 1R20)
05000263/2009005-03	URI	MSPI Basis Document does not Reflect Current Plant Configuration for RHRSW (Section 4OA1)
05000263/2010007-01	URI	Corrective Action Document for an Adverse Trend in LLRT Results (Section 4OA5)

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or 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.

<u>Section 1R01</u>

List of Open Work Orders—345 kV Substation; June 15, 2011
List of Open Work Orders—115 kV Substation; June 15, 2011
List of Open Work Orders—230 kV Substation; June 15, 2011
CAP 1291057; Pallet of Quarry Line Jumpers in Substation not Tied Down
CAP 1291159; Coolant Pump Flow Indications found Unplugged 10 XFMR
CAP 1290586; Unexpected Panel Alarm Received
Operations Manual B.09.03-05 345kV System—System Operation; Revision 30
Operations Manual B.09.05-05 115kV System—System Operation; Revision 14
Operations Manual B.09.03-01 345kV System—Function and General Description of System; Revision 12
Operations Manual B.09.05-01 115kV System—Function and General Description of System; Revision 7
C.4-B.09.02B; Abnormal Procedures—Loss of Normal Offsite Power; Revision 12
4 AWI-08.15.01; Risk Management for Outage and On-line Activities; Revision 6
3497; Housekeeping Inspection; Revision 5
Monticello Generating Station Transmission Guide
Monticello Maintenance Rule Program System Basis Document—345 kV Substation; Revision 2
Monticello Maintenance Rule Program System Basis Document—115 kV Substation; Revision 1
Monticello Maintenance Rule Program System Basis Document—4kV System; Revision 3

<u>Section 1R04</u>

0465-01; Emergency Filtration Treatment System; Revision 37
Operations Manual B.08.13-02; Control Room Heating and Ventilation and Emergency Filtration Train—Description of Equipment; Revision 9
Operations Manual B.08.13-06; Control Room Heating and Ventilation and Emergency Filtration Train—Figures; Revision 7
Operations Manual B.08.13-05; Control Room Heating and Ventilation and Emergency Filtration Train—System Operation; Revision 19
Operations Manual B.08.13-01; Control Room Heating and Ventilation and Emergency Filtration Train—Function and General Description of System; Revision 10
NH-170037; Main Control Room CRV / EFT System; Revision 80
2154-13; RCIC System Prestart Valve Checklist; Revision 25
2121; Plant Prestart Checklist RCIC System; Revision 14
NH-36251; RCIC (Steam Side); Revision 78
NH-36252; RCIC (Water Side); Revision 77
Operations Manual B.02.03-05; Reactor Core Isolation Cooling – System Operation; Revision 23

Calculation 10-118; RCIC MOV Functional Analysis; Revision 000
CAP 1218251; Inability to Adjust Limit Switches for HO-8
WO 407296; Install Small Alignment Bracket on HO-8/RCIC Indication Arms; 11/30/2010
EC 16392; Install Small Alignment Bracket on HO-8/RCIC Indication Arms
2113; Plant Prestart Checklist Standby Liquid Control; Revision 3
2154-07; Standby Liquid Control System Prestart Valve Checklist; Revision 11
Operations Manual B.03.05-01; Standby Liquid Control System – Function and General Description of System; Revision 5
Operations Manual B.03.05-05; Standby Liquid Control System – System Operation; Revision 15
NH-36253; Standby Liquid Control; Revision 77
2154-06; Standby Gas Treatment System Prestart Valve Checklist; Revision 11
2112; Plant Prestart Checklist Standby Gas Treatment System; Revision 13
2158-02; System Prestart Checklist – Secondary Containment Integrity; Revision 9
NH-36881; Standby Gas Treatment Flow Diagram; Revision 76
Operations Manual B.04.02-01; Secondary Containment/Standby Gas Treatment – Function and General Description of System; Revision 9
Operations Manual B.04.02-05; Secondary Containment/Standby Gas Treatment – System Operation; Revision 26
CAP 01283023; AO-2944: Air Fitting Leak/Cable Cover/Annubar Mounting
CAP 01284890; AR 01283023 Closed with Component Listed as “Inoperable”
WO 429808-04; MECH-AO-2944, Repair Leak on Air Supply Tubing
2154-23; RHR Service Water Prestart Valve Checklist; Revision 30
NH-36664; RHR Service Water & Emergency Service Water Systems; Revision 85
Operations Manual B.08.01.03-01; RHR Service Water System – Function and General Description of System; Revision 10
Operations Manual B.08.01.03-05; RHR Service Water System – System Operation; Revision 43

Section 1R05

Pre-Fire Strategy A.3-37; Transformers; Revision 10
Pre-Fire Strategy A.3-12-E; Steam Jet Air Ejector Room; Revision 1
NX-16991-46; Fire Penetration Seal Locations – Reactor Building, Turbine Building, and Intake Structure; Revision 78
3067; Combustion Source Use Permit; Revision 12
4 AWI-08.01.02; Combustion Source Use Permit (CSUP); Revision 11
4 AWI-08.01.01; Fire Prevention Practices; Revision 38
Pre-Fire Strategy A.3-01-F; Torus Area—Elevation 896’ and 923’; Revision 7
Pre-Fire Strategy A.3-14-A; Upper 4KV Bus Area (12, 14, & 16); Revision 13
Pre-Fire Strategy A.3-14-C; Railroad Car Area (Turbine Building); Revision 4
CAP 1282348; FME Barrier Fire on Phase C Iso-Phase Bus Duct
CAP 1291615; Housekeeping items identified

Section 1R11

SEG # RQ-SS-110; Simulator Exercise Guide—Licensed Operator Requalification for June 20, 2011 Exam; Revision 1
FP-T-SAT-75; Job Performance Measure and Simulator Exercise Guide Development; Revision 2
FP-PA-ARP-01; CAP Action Request Process; Revision 29

Section 1R12

Operations Manual B.05.01.01-05; Startup Range Monitors—Function and General Description of System; Revision XX
Operations Manual B.05.01.01-01; Startup Range Monitors—System Operation; Revision XX
CAP 1281170; SRM-24 Failed to Meet SR during SRM Functional Test
CAP 1281182; SRM/IRM Equipment Issues Affecting RMRB Indicator
CAP 1280039; SRM-23 Upscale High High Results in RPS Trip
CAP 1281109; Half-Scram Received on RPS 'B'
CAP 1281875; Inadequate controls in Place to Control Undervessel Activity
Discussion Regarding the Current Condition of SRM/IRM Systems in Anticipation of Refueling Activities; 4/13/2011
2nd Quarter 2011 System Health Report—NIS Startup Range Monitoring System
Monticello Maintenance Rule Program System Basis Document—Startup Range Monitors; Revision 1
Monticello Maintenance Rule Program—Boundary Definition Guidance Document; Revision 1
9007; Procedure for Moving Fuel into, Out of, and Within the Core; Revision 40
Operations Manual D.2-05; Reactor and Core Components Handling Equipment; Revision 23

Section 1R13

WO 395042; OPS – AO-2944, PMT/RTS Instructions; 9/10/2010
CAP 1279897; SBGT Suction Valve Position Indication Issues
EWI-08.14.01; AOV Program Document; Revision 10
WO 425886; AO-2944, Failed to Open on 'B' SGT Train Start; 4/13/2011
CAP 1280341; AO-2944 Failed to Open during System Start Up
CAP 1280340; AO-2979 Failed to Open
NUMARC 91-06; Guidelines for Industry Actions to Assess Shutdown Management; December 1991
FP-OP-ROM-02; Shutdown Safety Management Program; Revision 0
NX-9288-7-3; MNGP Electrical Schematic—Standby Gas Treatment—Units A & B; Revision L
NH-36881; Standby Gas Treatment Flow Diagram; Revision 76
Operations Manual B.04.02-05 Secondary Containment/Standby Gas Treatment-System Operation; Revision 25
Operations Manual B.04.02-01; Secondary Containment/Standby Gas Treatment—Function and General Description of System; Revision 9
WO 430152; Repair/Replace Relay 10A-K36B
CAP 01284261; Relay 10A-K36B Contacts not Open as Expected during ECCS Test
Operations Manual B.06.02.04-05; Generator-Stator Cooling—System Operation; Revision 16
B.06.02.04-05.G.2 Adjustment for Normal Operation; Revision 16
Operations Manual B.06.02.04-02; Generator-Stator Cooling—Function and General Description of System; Revision 16
WO 00392632; Adjust DP on Pressure Control Valve PCV-7418; 10/29/2009

Section 1R15

16 Battery Capacity Rested Results; April 13, 2011
CAP 01280463; 16 Battery Test Results WO 00394338-03
CAP 01285834; Elevated Vibration Levels on 11 SBLC
CAP 01286064; Abnormal Oil Trending in 11 SBLC Gearbox

CAP 01286722; Water Coming Out of P-203A No. 11 SBLC Pump Stuffing Box
EC 18234; P-203A Oil and Vibration Results Disposition
Operational Decision-Making Issue: SRV Decision Making Guide ODMI; 5/21/2011
ODMI 1289887 Attachment 1; OSCAR Evaluation of Elevated SRV ‘E’ Tailpipe Temperature; Revision 0; 6/9/2011
ODMI 1289887 Attachment 1; OSCAR Evaluation of Elevated SRV ‘E’ Tailpipe Temperature; Revision 1
Operations Manual C.4-B.03.03.B; Abnormal Procedures – Relief Valve Leaking; Revision 10 and 11
0112; Safety Relief Valves Operability Check; Revision 34
GE SIL-196; Summary of Recommendations for Target Rock Main Steam Safety/Relief Valves; September 30, 1976
CA-05-030; Determination of SRV Self Actuation Setpoint to be Used in Reactor Vessel Overpressure Analysis; Revision 13
USAR Section 4.4.2; Reactor Pressure Relief System; Revision 22
CAP 01290042; ‘E’ SRV Discharge Line Temp Approaching Baseline +30F
CAP 01289887; RV-2-71E Elevated Tailpipe Temps at Steady State Conditions
CAP 01289214; Unexpected Alarm ANN-3-A-9, Relief Valve Leaking
CAP 01291640; SRV ‘E’ Tailpipe Temperature Exceeded 165 F
0112; Safety Relief Valves Operability Check; Revision 34
CAP 01291952; Possible Wipe Indication on Low Pressure ‘A’ Bearing 3 during Shutdown
Operational Decision Making Evaluation for LP ‘A’ Turbine Bearing 3; June 26, 2011
Operational Decision Making Evaluation for 2-184-24A (11 Reactor Recirc Pump Power Indication); June 23, 2011
CAP 01291417; #11 Recirc Pump Power Indicator 2-184-24A Failed
CAP 01292138; Div. 2 Tailpipe Pressure Switch Did Not Actuate for ‘E’ SRV
CAP 01292205; E SRV Low-Low Set Tailpipe DP Transmitter Calibration Issue

<u>Section 1R18</u>

SCR-10-0421; EC 16861, Turbine Building CRW Modification for Tritium Mitigation 50.59 Screening; Revision 0
SCR-10-0421; EC 16861, Turbine Building CRW Modification for Tritium Mitigation 50.59 Screening; Revision 1
EC 16861; Turbine Building CRW Modification for Tritium Mitigation; Revision 3
NH-36033; P&ID Main Steam; Revision 80
NH-36034; P&ID Turbine & Extraction Steam Sheet 1 of 2; Revision 77
NH-36035; P&ID Turbine & Extraction Steam Sheet 2 of 2; Revision 77
NH-36035-2; P&ID Steam Jet Air Ejectors; Revision 79
NH-366036; P&ID Condensate & Feedwater; Revision 79
NH-36037; P&ID Condensate & Feedwater; Revision 80
2011 Refuel Outage Tritium Project Work Order Schedule; April 13, 2011
CAP 1278449; Follow up Items for Tritium Mitigation Project
CAP 1274699; Turbine Bldg CRW Mod for Tritium Mitigation 2011 Milestone
CAP 1265198; Poor Scope Control on Tritium Mitigation Project
CAP 1264409; Tritium Mitigation Alignment Meeting Actions
CAP 1280202; 2154-18 Rev 16. EC16861
CAP 1276758; Groundwater Leakage into S-44 While Cleaning the Sump
FG-E-SE-03; 50.59 Resource Manual; Revision 2
Operations Manual B.07.01-05; Liquid Radwaste—System Operation
Operations Manual B.07.01-02; Liquid Radwaste—Description of Equipment

EC 16861 Design Input Checklist Part B
EC 16861 Design Input Checklist Part A
FP-E-MOD-02; Engineering Change Control; Revision 9

Section 1R19

CAP 1279897; SBGT Suction Valve Position Indication Issues
EWI-08.14.01; AOV Program Document; Revision 10
4700-MAO; MAO-Component Test Matrix Air Operated Valve; Revision 3
WO 390721; Mech – AO-2945, Adjust Actuator Closed Travel Stop; 5/3/2010
NH-36881; Standby Gas Treatment Flow Diagram; Revision 76
Operations Manual B.04.02-05 Secondary Containment/Standby Gas Treatment-System Operation; Revision 25
WO 387168; CV-2942—Rebuild Valve and Actuator; 12/10/2009
MNGP AOV Data Sheet; CV-2942—SBGT B Loop Flow Control; 6/8/2010
WO 379413; 94-5B/D – Replace "B" Cutler-Hammer Relays; 6/7/2009
WO 387149; AO-2944 PMT; 1/13/2010
4 AWI-04.05.06; Post-Maintenance Testing; Revision 18
Operations Manual B.04.02-01; Secondary Containment/Standby Gas Treatment—Function and General Description of System; Revision 9
NX-9288-7-3; MNGP Electrical Schematic—Standby Gas Treatment—Units A & B; Revision L
CAP 1279870; WO 394967 PMT Steps Don't Necessarily Test the Relays
ESP-RPV-0010; Reactor Coolant Pressure Boundary End of Interval System Leakage Test; Revision 0
0255-20-IIC-1; Reactor Coolant Pressure Boundary Leakage Test; Revision 34
0255-20-IIC-2; Reactor Coolant Pressure Boundary Leakage Test; Revision 29
CAP 1283851; RCS Leakage Test – Documented Leakage
CAP 1283748; RCS Leakage Test – MO-2012 Leaking
CAP 1283752; RCS Leakage Test – MO-2374 Leaking
CAP 1283759; RCS Leakage Test – SV-122/10-15 Leaking
CAP 1283763; RCS Leakage Test – SV-122/14-19 Leaking
CAP 1283786; RCS Leakage Test – XR-32-2 Leaking Steady Stream
CAP 1283801; 1-RCI-V-1 Minor Packing Leak
CAP 1283835; RCS Leakage Test – MO-2-53A Packing Leakage
CAP 1283845; MO-2034 Packing Leak Approx 1 Drop per Minute
0086; SBLC Refueling Tests; Revision 37
WO 360594-33; Main Transformer 22KV/345KV Back Feed Testing; Revision 4
8396-01; 14A FWH Shell Side Initial In-Service Leak Test; Revision 0
8396-02; 14B FWH Shell Side Initial In-Service Leak Test; Revision 0
8396-03; 15A FWH Shell Side Initial In-Service Leak Test; Revision 0
8396-04; 15B FWH Shell Side Initial In-Service Leak Test; Revision 0
EC 18174; Documentation of Leak Testing Strategy for Replacement FWN Piping Shell Side Welds

Section 1R20

9006 Reactor Well and Dryer-Separator Storage Pool Draining Procedure; Revision 21
NUC-06.01; Fuel/Core Component Moves Generation; Revision 11
Monticello RF-25 Shutdown Safety Review Report
4 AWI-08.15.03; Risk Management for Outages; Revision 6
1371; Drywell Prestart Inspection; Revision 8

C.1; Startup Procedure; Revision 68A
0112; Safety Relief Valves Operability Check; Revision 34
1057; HPCI Turbine Overspeed Trip Test; Revision 15
1056; RCIC Turbine Overspeed Trip Test; Revision 14
2150; Plant Restart Checklist; Revision 38
FP-S-WHL-01; 10 CFR 26 Scope of Work Hour Limits; Revision 2
FP-S-FMP-01; 10 CFR 26 Fatigue Management Fleet Procedure; Revision 2
FP-S-CWH-01; 10 CFR 26 Calculating Work Hours; Revision 1
FP-S-FAP-01; 10 CFR 26 Fatigue Assessment Procedure; Revision 1
CAP 1241474; Violation of Work Hours Rules under 10 CFR 26
ITAR 1254318; WorkForce Security Schedule Change Request
CAP 1219126; Multiple Security Officers Exceeded 10 CFR 26 Rule
CAP 1266406; Security Officer Exhibited Fatigue While on Post
QF-1725; For Cause Fatigue Assessment—Individual 1; 11/10/2011
QF-1725; For Cause Fatigue Assessment—Individual 2; 11/10/2011
CAP 1270640; Worker Self Declared Fatigue
CAP 1286813; Concern over WorkForce Turnover Time
List of Minimum 10 Hour Break Periods SFM-Officers, LT.'s, and Capt.'s
C.3; Shutdown Procedure; Revision 65
Plant Operating Review Committee, Restart Readiness Review for 2716 – SRV Outage; dated June 25, 2011 & June 26, 2011
CAP 01292007; 12 EDG Test 0187-02B Conflict with Forced Shutdown Start-up
CAP 01291945; CV-3490 Failed to Close
CAP 01291960 & 01292084; Unexpected Annunciator Recirc Pump Motor B Hi Vibes
CAP 01291990; Several Issues Identified by Resident during Inspection
CAP 01291968; Load Limit Clutch Failure to Engage
CAP 01291976; IRM 18 Hi Hi is Low Out of Spec
CAP 01291832; Turbine Hood Spray CV-1269 Manually Closed during S/D
CAP 01291834; Invalid Reactor Pressure Causes Loss of Gardel HB during S/D
CAP 01291867; IRM-18 Does not Indicate Full In on C-05
CAP 01291844; Operator Attempted to Manipulate Incorrect Handwheel
Human Performance Event Investigation Tool for "Operator Attempted to Manipulate Incorrect Handwheel"; dated June 24, 2011
CAP 01292222; PCV-7497B Slow to Open During C.1 Start-up Procedure
CAP 01292207; CV-4174B Leaks from the Stem by Control Knob
AEC Letter to NSP; "RE: License No. DPR-22"; dated February 4, 1974
Calculation No. CA-76-138; "Structural Requalification for New 85 Ton Crane"; Revision 0
Calculation No. CA-05-101; "Evaluation of Reactor Steel Superstructure for 105 Ton Reactor Building Crane"; Revision 3A
Calculation No. CA-05-103; "Reactor Building Superstructure Seismic Response Analysis with 105 Ton Crane"; Revision 0A
Calculation No. CA-05-107; "Structural Seismic Qualification Reactor Building Crane Upgrade for ISFSI"; Revision 0B
NSP letter to AEC; "Submittal of Analyses of the Spent Fuel Shipping Cask Drop Accident"; dated October 1, 1974
NRC letter to NSP; "Request for Additional Information Northern States Power Company Monticello Nuclear Generating Plant Docket No. 50-263"; dated January 31, 1975
NSP letter to NRC; "Response to Request for Additional Information Northern States Power Monticello Nuclear Generating Plant Docket No. 50-263"; dated February 17, 1975
NSP letter to NRC; "Status Report on Plans for Off-Site Shipment of Spent Fuel"; dated May 30, 1975

NSP letter to NRC; “Off Site Shipment of Spent Fuel”; dated January 22, 1976
NSP letter to NRC; “Off Site Shipment of Spent Fuel”; dated February 13, 1976
NSP letter to NRC; “Off-Site Shipment of Spent Fuel”; dated October 27, 1976
NSP letter to NRC; “Design Report for Redundant Reactor Building Crane”; dated November 22, 1976
NRC letter to NSP; “Safety Evaluation by the Office of Nuclear Reactor Regulation [NRR] Supporting Use of 25 ton Spent Fuel Shipping Casks NFS-4 and NAC-1 Northern States Power Company Monticello Nuclear Generating Plant Docket No. 50-263”; dated January 25, 1977
NRC letter to NSP; “Northern States Power Company Monticello Nuclear Generating Plant Docket No. 50-263 “Request for Additional Information”; dated February 11, 1977
NSP letter to NRC; “Responses to Request for Additional Information”; dated February 28, 1977
NRC letter to NSP; “Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Approval of Crane Modifications and Use of 70 Ton Spent Fuel Shipping Cask IF-300 Northern “States Power Company Monticello Nuclear Generating Plant Docket No. 50-263”; dated May 19, 1977
NSP letter to NRC; “Reactor Building Redundant Crane”; dated June 24, 1977
Regulatory Guide 1.104; “Overhead Crane Handling Systems for Nuclear Power Plants”; dated February 1976
AR 01290843, “Spent Fuel Cask Height Restriction is not in USAR”; dated June 16, 2011

<u>Section 1R22</u>

0151-01; Secondary Containment Capability Test; Revision 20
CAP 1282522; Control Room Light Indication on V-EF-20 Didn’t Change on Trip
CAP 1282524; OG-28 Cut Out during SCT Capability Test
1297-01; Secondary Containment Door Interlock Check; Revision 14
WO 394853-01; OPS-SCT, 0151-01 Secondary Containment Capability Test
8136; Secondary Containment Penetrations; Revision 17
8136-04; Secondary Containment Penetration Work Control Checklist
NH-36881; Standby Gas Treatment Flow Diagram; Revision 76
OSP-ECC-0566; Low Pressure ECCS Automatic Initiation and Loss of Auxiliary Power Test; Revision 7
CAP 01284441; Relay 10A-K46A De-energized Prior to 5 Minutes in ECCS Test
CAP 01284261; Relay 10A-K36B Contacts not Open as Expected during ECCS Test
CAP 01284602; Received 94-A-6 Alarm when 12 EDG Loaded Onto Bus
CAP 1287754; Change to Verif. Practice not Processed as Temp Change
AWI-04.04.02; Equipment Positioning, Independent and Concurrent Verification Methods; Revision 21
0081; Control Rod Drive Scram Insertion Time Test; Revision 59
0081A; Control Rod Order List; 2300 Rev 1 Attachment
FP-G-DOC-04; Procedure Processing; Revision 12
0006; Scram Discharge Volume Hi Level Scram Test and Calibration Procedure; Revision 30
Operations Manual B.05.05-02; Reactor Manual Control System (RMCS)—Description of Equipment; Revision 5
NX-7866-74-5; Elementary Diagram Reactor Control System; Revision 8
Operations Manual B.05.05-01; RMCS—Function and General Description of System; Revision 4
Operations Manual B.05.06-02; Plant Protection System—Description of Equipment; Revision 19
Operations Manual B.05.02-01; Rod Worth Minimizer—Function and General Description of System; Revision 1

Operations Manual B.05.05-05; RMCS—System Operation; Revision 16
0000-J; Operations Daily Log – Part J Outplant; Revision 104
0000-H; Operations Daily Log – Part H; Revision 94
1257-B; Reactor Coolant Drywell Leak Rate Check; Revision 1
OWI-02.03; Operator Rounds; Revision 16
2023; Reactor Building Outside—Operator Rounds Attachment to Log; Revision 38

Section 2RS

CAP 1239074, Dose Rates Greater Than 0.25 millrem Per Hour Found Outside Warehouse No. 4; June 26, 2010
CAP 1238088; Steam Jet Air Ejector Barrier Locked with Personnel Working Inside; June 12, 2010
CAP 1250126; Individual Entered Radiologically Controlled Area with Coffee; September 17, 2010
CAP 1250156; ALARA Effectiveness KPI is Below Expectations; September 17, 2010
CAP 1251220; Contract Radiation Protection Technician Reaching Across Radiologically Controlled Area Boundary without Surveying; September 24, 2010
CAP 1257138; Worker Wearing another Worker's Thermoluminescent Dosimeter; November 4, 2010
CAP 1258780; Radiation Protection Technician Entered Onto C/D Scaffold Without Lock Out on Turbine Building Crane; November 16, 2010
CAP 1260297; Individual Alarmed the Argos Monitor But Did Not Contact Radiation Protection; November 26, 2010
CAP 1261528; Dose Alarm Received During High Radiation Boundary Checks; December 2010
CAP 1267451, Radiation Protection Stopped Work on WO 366810; Install New Condensate Demineralizer Crane; March 9, 2011
CAP 1268006; Improper Radworker Practice Observed; December 15, 2010
CAP 1269644; Radworker Practices Not Meeting Standards of Industry Excellence; February 6, 2011
CAP 1273202; Piping on Tag-Out Not Completely Drained of Water; March 1, 2011
CAP 1273886; Unexpected Dose Rate Alarm in Drywell; March 6, 2011
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CAP 1274777; Individual Observed Working in Class One in Condenser Contaminated Area; March 11, 2011
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CAP 1275528; System Breach of Contaminated System with Not Radiation Protection Technician Present; March 16, 2011
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CAP 1277975; Personnel Contamination Event Number 14; March 29, 2011
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CAP 1279872; Unexpected High Airborne Radioactivity Stooped Work on Main Steam Isolation Valve; April 9, 2011
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CAP 1281076; Received Unexpected Dose rate Alarms While Moving Old Low Power Range Monitors; April 16, 2011
CAP 1281878; Contamination Spread in Reactor Water Clean-up Room When Testing MO-2398; April 21, 2011
CAP 1283115; Untimely Control Room Notification of High Airborne Refuel Floor Evacuation; April 30, 2011
CAP 1283116; Access to Refuel Floor Not Effectively Restricted After Evacuation; April 30, 2011
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2RS2; Occupational ALARA Planning and Controls (71124.02)
FP-RP-JPP-01; Radiation Protection Job Planning; Revision 8
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CAP 1251207; Station ALARA Committee Meeting Shortfalls; September 24, 2010
CAP 1264673; Executive Site ALARA Committee Meeting Attendance Issue; December 30, 2010
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QF-1207; Radiological Work Assessment Form-ALARA Review Checklist-Replacement of Condensate Demineralizers; January 25, 2011
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QF-1223; Work in Progress Review; Replacement of Condensate Demineralizers; February 10, 2011
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WO 319048; Investigate; Repair Inboard Main Steam Isolation Valve; March 17, 2011

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0000-J; Operations Daily Log – Part J Outplant; Revision 103
LER 2010-006; Mode Change with PCIVs Inoperable
LER 2010-004; Secondary Containment Briefly Inoperable Due to Simultaneous Opening of Airlock Doors
LER 2010-003; Secondary Containment Briefly Degraded
LER 2010-002; Secondary Containment Briefly Degraded
CAP 01272291; FAQ 11-02 – MSPI Basis Document Updates

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CAP 01285690; Items Identified by NRC on RFO 25 DW Closeout Inspection
Operator Burden Reports (various dates)
Outage Mode Change Reports (various dates)
HELB Barrier Status Reports (various dates)
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CAP 00826605-03; GE Part 21 Notice (SC05-03) – Potential to Exceed Low Pressure TS Safety Limit
CAP 01253620-08; Track Resolution of Design Basis for ADS Bypass Timer
CAP 01079705-06; Develop a Licensing Basis for EDG Tornado SSCs Unprotected
CAP 01173169-05; Revise Technical Specifications for Core Spray
CAP 01177862-05; Figure 3.4.9-1 of Technical Specifications is Non-Conservative
CAP 01237578-10; Track Status of LAR Submittal – Update PT Curves
CAP 01253620-08; Track the Revision of this Issue: 1973 HELB
CAP 01263913-02; MO-2035, Incorrect Voltage Used in MOV Calculation

CAP 01292013; Items Found During Final Drywell Inspection by NRC
Trend CAPs; dated January through April 2011
SAR 01249160; Site Roll-UP Meeting Results – 1st and 2nd Quarter 2010
Department Roll-Up Meeting Results - RP-Chem; 3rd Quarter 2010
SAR 01264947; Department Roll-Up Meeting Results – Maintenance; 4th Quarter 2010
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1076; reactor Dynamics Test Procedure; Revision 14
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A.7-TSG-03; Plant Status Assessment; Revision 6
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A.7-TSG-06; Abnormal System Configuration; Revision 4
5790-001-01; Emergency Response Organization; Revision 86
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2261; EOP/SAMG Impact Assessment Checklist; Revision 0
4 AWI-05.08.00; Fleet Modification Process Implementation; Revision 12
QF-0515A (FP-E-MOD-04); Design Input Checklist (Part A- Engineering Programs and Departmental Reviews); Revision 15
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A.2-210; Engineering Support in the TSC; Revision 10
CAP 1276087; IERL1-11-1: Training Improvement on Use of SAMG/EDMG in EP
CAP 1276567; IERL1-11-1: SAMG/EDMG Training Improvements Noted
MT-M7429L-005; SAMG Table Top Exercise Training Record; April 27, 2011
MT-BEP; Emergency Plan Training Program Description; Revision 5
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MT-BEP-SAM-005L; SAMG Table Top Exercise Lesson Plan
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CAP 1286843; Items from NRC Inspection TI-184 (SAMGs)
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CAP 01252363; NRC-Identified that HELB Isolation Calculations Appeared Inconsistent with the TS and USAR; October 1, 2010
Docketed letter dated July 29, 1974 from Northern States Power to the NRC submitting report on Postulated Pipe Failures Outside Containment
CA-00-82; HPCI Line Break Mass Release Analysis for the Elimination of the 150,000 lbm/hr Time Delayed Isolation; June 14, 2000
CA-96-078; HPCI HELB in the HPCI Building Analysis Module A.1.4; Revision 1
CA-96-079; RCIC HELB in the RCIC Room Analysis Module A.1.2; September 12, 1996
CA-96-0175; RWCU HELB in the HX area with Automatic Isolation, Rerate, Analysis Module R.1.3-2; January 7, 1997

LIST OF ACRONYMS USED

AC	Alternating Current
ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access Management System
AEC	Atomic Energy Commission
ALARA	As-Low-As-Is-Reasonably-Achievable
APRM	Average Power Range Monitor
ARM	Area Radiation Monitor
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient without Scram
CAP	Corrective Action Program
CDE	Consolidated Data Entry
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CRV	Control Room Ventilation
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
EPRI	Electric Power Research Institute
FAQ	Frequently Asked Questions
FIN	Finding
FSAR	Final Safety Analysis Report
HCU	Hydraulic Control Unit
HELB	High Energy Line Break
HPCI	High Pressure Coolant Injection
HWC	Hydrogen Water Chemistry
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IST	Inservice Test
kV	Kilovolt
LLRT	Local Leak Rate Testing
LOCA	Loss-of-Coolant Accident
MPR	Mechanical Pressure Regulator
MSIV	Main Steam Isolation Valve
MNGP	Monticello Nuclear Generating Plant
MSPI	Mitigating Systems Performance Indicator
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NUMARC	Nuclear Management and Resources Council
OBD	Operable But Degraded
OBE	Operational Basis Earthquake
OBN	Operable But Nonconforming
ODMI	Operational Decision Making Instruction
OpESS	Operating Experience Smart Sample
OSP	Outage Safety Plan
OWA	Operator Workaround

PARS	Publicly Available Records System
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PM	Planned or Preventative Maintenance
PMT	Post-Maintenance Testing
PRA	Probabilistic Risk Assessment
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RFO	Refueling Outage
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SAMG	Severe Accident Management Guideline
SBGT	Standby Gas Treatment
SBLC	Standby Liquid Control
SBO	Station Blackout
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SRV	Safety Relief Valve
SSC	Structures, Systems, and Components
TS	Technical Specification
TSO	Transmission System Operator
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
URI	Unresolved Item
WBC	Whole Body Count
WO	Work Order

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Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

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Letter to T. O'Connor from K. Riemer dated August 8, 2011

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT
INTEGRATED INSPECTION REPORT 05000263/2011003

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