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

August 11, 2014

Ms. Karen Fili
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company, Minnesota
2807 West County Road 75
Monticello, MN 55362–9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT NRC INTEGRATED AND POWER UPRATE INSPECTION REPORT 05000263/2014003

Dear Ms. Fili:

On June 30, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Monticello Nuclear Generating Plant. On July 16, 2014, the NRC inspectors discussed the results of this inspection with you and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented two findings of very low safety significance (Green) during this inspection. Both of these findings involved violations of NRC requirements. Further, inspectors documented one licensee-identified violation, which was determined to be of very low safety significance in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the subject or severity of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, 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, United States Nuclear Regulatory Commission, Washington, DC 20555–0001; and the Resident Inspector Office at the Monticello Nuclear Generating Plant.

Additionally, if you disagree with a 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 Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding,” 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 Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Kenneth Riemer, Branch Chief
Branch 2
Division of Reactor Projects

Docket No. 50–263
License No. DPR–22

Enclosure:
IR 05000263/2014003;
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:	05000263/2014003
Licensee:	Northern States Power Company, Minnesota
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello, MN
Dates:	April 1 through June 30, 2014
Inspectors:	P. Zurawski, Senior Resident Inspector P. Voss, Resident Inspector K. Stoedter, Prairie Island Senior Resident Inspector P. LaFlamme, Prairie Island Resident Inspector M. Phalen, Senior Health Physicist
Approved by:	K. Riemer, Branch Chief Branch 2 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000263/2014003; 04/01/2014—06/30/2014; Monticello Nuclear Generating Plant, Surveillance Testing and Other Activities.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green findings were identified by the inspectors. The findings were considered non-cited violations of NRC regulations. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," effective date January 1, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process" Revision 5, dated February 2014.

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," when the licensee failed to comply with the appropriate American Society of Mechanical Engineers (ASME) Code requirements during implementation of the temporary modification (TMOD) of the recirculation pump seal coolers. Specifically, the licensee failed to meet the ASME Code, Section III, Subsection NB 3671.3, Class I piping design requirements for the end cap joint design on the vent line in this TMOD.

  The inspectors determined that the performance deficiency was more than minor, and a finding because it was associated with the Design Control attribute of the Initiating Systems Cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the licensee inadequately designed the vent line end cap such that the design was non-compliant with ASME Code, Section III, Subsection NB 3671.3 requirements and, therefore, potentially challenged plant stability. The inspectors reviewed Attachment 0609.04, "Initial Characterization of Findings," Table 3—SDP Appendix Router. The inspectors answered 'Yes' to all of the questions in Sections A through E of Table 3, and, therefore, the finding was evaluated using the SDP in accordance with IMC 0609, "The Significance Determination Process (SDP) for Shutdown Operations," Appendix G, Attachment 1, Exhibit 2, "Initiating Events Screening Questions." The inspectors answered all the questions in Exhibit 2 and determined that this finding did not increase the likelihood of a plant initiating event during shutdown operations nor did it affect any shutdown safety functions. Therefore, the finding was determined to have very low safety significance. This finding has a cross-cutting aspect in the area of Human Performance, Avoid Complacency, because the licensee failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Specifically, the licensee failed to recognize the latent issue concerning appropriate reactor coolant system pressure boundary identification and subsequent ASME Code piping design requirements for piping systems associated with this TMOD (H.12). (Section 4OA5.1)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," on May 7, 2014, for the licensee's failure to ensure that activities affecting quality were prescribed by documented procedures of a type appropriate to the circumstances. Specifically, the site changed Procedure 0255–02–III, "SBLC [standby liquid control] Quarterly Pumps and Valve Test," to include allowances for starting the safety-related SBLC pumps and adjusting a throttle valve to achieve the desired pump discharge pressure prior to performance of in-service testing, actions which, without evaluation, constituted unacceptable preconditioning.

  The inspectors determined that the licensee's failure to ensure the SBLC pump and valve test surveillance procedure was appropriate to the circumstances was a performance deficiency requiring evaluation. The inspectors screened the performance deficiency and determined that the issue was more than minor because it adversely impacted the Mitigating Systems Cornerstone attribute of Procedure Quality, and affected the cornerstone objective to ensure the availability, reliability, and capability that respond to initiating events to prevent undesirable consequences (i.e., core damage). In addition, if left uncorrected, it had the potential to lead to a more significant safety concern. Specifically, proceduralizing actions which could constitute unacceptable preconditioning, such as manipulating the physical condition of a structure, system or component (SSC) before or during TS surveillance or ASME Code testing, could mask the actual as-found condition of the SSC and result in an inability to verify the operability of the SSC.

  The inspectors determined that this finding was of very low safety significance because each question listed in IMC 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," was answered 'No'. The inspectors concluded that this finding was cross-cutting in the Human Performance, Change Management aspect, because the licensee did not use a systematic process for evaluating and implementing change so nuclear safety remains the overriding priority. Specifically, revising procedures to allow the SBLC pump to be started for test configuration flow adjustments immediately prior to a surveillance test, without an evaluation of preconditioning acceptability, could mask the ability to detect degraded equipment performance (H.3). (Section 1R22)

- Violations of very low safety or security significance or Severity Level IV that were identified by the licensee have been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program (CAP). These violations and CAP tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

At the end of the 1st quarter 2014, Monticello was holding at 1775 megawatts thermal (MWt) (the previously licensed 100 percent power), pending additional review of extended power uprate (EPU) test data. With two exceptions during the second quarter 2014, Monticello operated at or near, its previously licensed power limit of 1775 MWt. On May 24, 2014, power was lowered to approximately 1480 MWt to facilitate control rod pattern adjustments. The unit was returned to approximately 1775 MWt on May 25, 2014. On June 28, 2014, power was reduced to approximately 1560 MWt to facilitate maximum extended load line limit plus (MELLA+) testing. The unit was returned to approximately 1775 MWt on June 29, 2014.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate Alternating Current 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:

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

- 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;
- 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;

- 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
- 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 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.

.2 Readiness For Impending Adverse Weather Condition—Heavy Rainfall/External Flooding Conditions

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the expected flooding conditions based on predicted rainfall and rises in local river and lake levels. The evaluation included a review to check for deviations from the descriptions provided in the Updated Safety Analysis Report (USAR) for features intended to mitigate the potential for flooding. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during the predicted flood conditions or allow water ingress past a barrier. The inspectors also walked down underground bunkers/manholes subject to flooding that contained multiple train or multiple function risk-significant cables. The inspectors also reviewed the abnormal operating procedure and compensatory measures for mitigating the expected flooding conditions to ensure they could be implemented as written. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in 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:

- 13 emergency service water (ESW) during 14 ESW work;
- 11 emergency diesel generator (EDG) during 12 EDG maintenance; and
- High pressure core spray (HPCI) with condensate demineralizer trouble.

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, 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 three partial system walkdown samples as defined 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 23–B; intake structure corridor;
- Fire Zone 03–A; recirculation motor generator set room;
- Fire Zone 01–B; 11 residual heat removal (RHR)/core spray room;
- Fire Zone 03–B; standby liquid control (SBLC) area (962’ Rx bldg. East); and
- Fire Zone 01–E; HPCI room.

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.

1R06 Flooding (71111.06)

.1 Underground Vaults

a. Inspection Scope

The inspectors selected underground bunkers/manholes subject to flooding that contained cables whose failure could disable risk-significant equipment. The inspectors determined that the cables were not submerged, that splices were intact, and that appropriate cable support structures were in place. In those areas where dewatering devices were used, such as a sump pump, the device was operable and level alarm circuits were set appropriately to ensure that the cables would not be submerged. In those areas without dewatering devices, the inspectors verified that drainage of the area was available, or that the cables were qualified for submergence conditions. The inspectors also reviewed the licensee's corrective action documents with respect to past submerged cable issues identified in the CAP to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following underground bunkers/manholes subject to flooding:

- CP100; 1AR feeder and control;
- 2MH03; RHR, core spray, safety relief valve (SRV), and alternative shutdown system;
- NMH309; 1AR feeder; and
- NMH310; 1AR feeder and control.

Specific documents reviewed during this inspection are listed in the Attachment to this report. This inspection constituted one underground vaults sample as defined in IP 71111.06–05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On April 24 and May 1, 2014, the inspectors observed a crew of licensed operators during licensed operator requalification training 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. Specifically, the inspectors observed crew performance of external flood mitigation tabletop exercises which were completed as a corrective action for an NRC-identified Notice of Violation. 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;
- correct use and implementation of abnormal and emergency procedures;
- 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 simulator sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On June 21, 2014, the inspectors observed operators performing turbine generator monthly testing in the control room. This was an activity that required heightened awareness or was related to increased risk. 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 procedures;
- control board manipulations; and
- oversight and direction from supervisors.

The performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

- Plant protection system—reactor protection system (RPS); and
- Reactor core isolation cooling (RCIC).

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 two quarterly maintenance effectiveness samples 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:

- 2R–1R bus 12 ammeter indicating no current during transfer;
- Dry storage canister (DSC) No. 16 high risk weld inspection/repairs;
- Circulating water basin level transmitter diverging trend;
- MELLA+ license requirement implementation activities;
- 12 EDG maintenance—yellow risk; and
- Condensate demineralizer trouble.

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 during this inspection are listed in the Attachment to this report. These maintenance risk assessments and emergent work control activities constituted six samples as defined in IP 71111.13–05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- DSC No. 16 weld repair activities;
- 11 SBLC inoperable;
- Non-conservative alternate nitrogen system required pressure calculation; and
- Strain gauge configuration challenge to steam dryer stress acceptance criteria.

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.

These operability inspections constituted four samples as defined in IP 71111.15–05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modifications:

- Circulating water basin level trip bypass modification (temp); and
- EDG transfer logic compensatory measure (temp).

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 systems. 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; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. 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 are listed in the Attachment to this report.

This inspection constituted two temporary modification samples 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:

- SRV 'F' tailpipe differential pressure trip unit replacement;
- SRV 'H' low set tailpipe interlock trip unit replacement;
- RPS channel 'A' contact burnishing;
- RCIC torus suction outboard valve; and
- P77 EDG fuel oil service pump.

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.

These inspections constituted five PM testing samples as defined in IP 71111.19–05.

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:

- 0143; drywell-torus monthly vacuum breaker check (Routine);
- 0301; safeguard bus voltage protection relay unit functional test (Routine);
- 0255–02–II; SBLC quarterly pump and valve tests (in-service test (IST)); and
- 0187–02; 12 EDG/12 ESW quarterly pump and valve test (Routine).

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

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were clearly stated, demonstrated operational readiness, and were 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 and one IST sample, as defined in IP 71111.22, Sections–02 and–05.

b. Findings

Inadequate Standby Liquid Control Quarterly Pump and Valve Test Due to Proceduralized Unacceptable Preconditioning

Introduction

The inspectors identified a finding of very low safety significance and a NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," on May 7, 2014, for the licensee's failure to ensure that activities affecting quality were prescribed by documented procedures of a type appropriate to the circumstances. Specifically, the site changed Procedure 0255–02–III, "SBLC Quarterly Pumps and Valve Test," to include allowances for starting the safety-related SBLC pumps and adjusting a throttle valve to achieve the desired pump discharge pressure prior to performance of IST, actions which, without evaluation, constituted unacceptable preconditioning.

Description

During the period of May 5–7, 2014, the inspectors observed activities associated with SBLC quarterly pumps and valve test 0255–02–III, performed to ensure the SBLC system met TS 3.1.7 and ASME Operation and Maintenance (OM) Code requirements. On May 5, 2014, Procedure 0255–02–III was performed and the 11 SBLC pump measured flow exceeded the OM Code required flow limit. The licensee documented this anomaly via CAP 1429453, and added enhancements to the 0255–02–III procedure via a temporary change. The enhancements included adding cautionary statements, including: "Delay between opening/closing DM–56 following starting/stopping the pump and stopwatch can impact test results," and to quickly open valve DM–56. The test was re-performed on May 6, 2014, using the revised 0255–02–III procedure. However, because operators were being very deliberate during this test to ensure pump discharge pressure was carefully brought into band, it required approximately 50 seconds to adjust throttle valve CV–2395. Since this length of time had the potential to impact the test results, the licensee initiated CAP 1429605, which resulted in a May 6, 2014, revision to Procedure 0255–02–III to improve the repeatability of the test results. These procedure changes included first starting the SBLC pump, adjusting throttle valve CV–2395 to achieve the desired pump discharge pressure, stopping the pump and draining the SBLC test tank. After completion of these activities, the test procedure required the pump to be re-started with CV–2395 already throttled in the correct position and proceeding with the IST steps to ensure TS and OM Code requirements were met.

On May 7, 2014, immediately after the licensee's pre-job briefing, but prior to performance of the test using the May 6, 2014, procedure revision, the inspectors questioned the licensee on their basis for why the procedure change would not represent preconditioning, and if so, whether an evaluation of the preconditioning had been completed. In response to the inspectors questions, the licensee determined that during preparation of the procedure change, the IST engineer (procedure change preparer), backup IST engineer, system engineer and program supervisor reviewed the examples of acceptable and unacceptable preconditioning in Administrative Work Instruction 4 AWI–09.04.01, IST Program, Figure 5.13, "Preconditioning of Components," and determined that the procedure revision would constitute acceptable preconditioning. However, the acceptable preconditioning conclusion and the basis were not documented on Form 3851 as delineated by Procedure 4 AWI–09.04.01.

Inspectors further reviewed NRC Technical Guidance 9900, "Preconditioning," that defines unacceptable preconditioning as "the alteration, variation, manipulation, or adjustment of the physical condition of an SSC before or during technical specification surveillance or ASME Code testing that will alter one or more of an SSC's operational parameters which results in acceptable test results. Such changes could mask the actual as-found condition of the SSC and possibly result in an inability to verify the operability of the SSC. In addition, unacceptable preconditioning could make it difficult to determine whether the SSC would perform its intended function during an event in which the SSC might be needed. Influencing test outcome by performing valve stroking, preventive maintenance, pump venting or draining, or manipulating SSCs does not meet the intent of the as found testing expectations…" Licensee Procedure CD 5.5, "Inservice Testing Standard," includes similar guidance.

Technical Specification Surveillance Requirement (SR) 3.1.7.7 states, "Verify each pump develops a flow rate ≥ 24 gpm at a discharge pressure ≥ 1275 psig." Procedure 0255–02–III echoes the purpose as demonstrating SBLC pump and valve operability and states, "The test demonstrates each SBLC pump is capable of delivering greater than or equal to 24 gpm at a discharge pressure of greater than or equal to 1275 psig as required by TS SR 3.1.7.7. It further states, "This test satisfies IST Program Surveillance requirements…" Based on review of this information, the inspectors determined that the licensee's failure to document the May 6, 2014, 0255–02–III procedure changes as incorporating acceptable preconditioning, resulted in a procedure that contained unacceptable preconditioning and consequently not appropriate to the circumstance per NRC guidance.

The licensee documented this inspector-identified issue via CAP 1429784. Corrective actions included determining that the revised procedure steps were acceptable preconditioning and documenting those conclusions per 4 AWI–09.04.01 on Form 3851 prior to performing the surveillance.

Analysis

The inspectors determined that the licensee's failure to ensure the SBLC pump and valve test surveillance procedure was appropriate to the circumstances was a performance deficiency requiring evaluation. 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 adversely impacted the Mitigating Systems Cornerstone attribute of Procedure Quality, and affected the

cornerstone objective to ensure the availability, reliability, and capability that respond to initiating events to prevent undesirable consequences (i.e., core damage). In addition, if left uncorrected, it had the potential to lead to a more significant safety concern. Specifically, proceduralizing actions which could constitute unacceptable preconditioning, such as manipulating the physical condition of a SSC before or during TS surveillance or ASME Code testing, could mask the actual as-found condition of the SSC and result in an inability to verify the operability of the SSC.

The inspectors utilized IMC 0609, Attachment 0609.04, "Initial Characterization of Findings," and concluded that the significance of this finding be determined using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors determined that this finding was of very low safety significance because each question listed in IMC 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," was answered 'No'.

The inspectors concluded that this finding was cross-cutting in the Human Performance, Change Management aspect because the licensee did not use a systematic process for evaluating and implementing change so nuclear safety remains the overriding priority. Specifically, revising procedures to allow the SBLC pump to be started for test configuration flow adjustments immediately prior to a surveillance test, without an evaluation of preconditioning acceptability, could mask the ability to detect degraded equipment performance (H.3). (Section 1R22)

Enforcement

Title 10 CFR, Part 50, Appendix B, Criterion V states, in part, that activities affecting quality shall be prescribed by documented instructions of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, on May 7, 2014, the licensee failed to prescribe a documented instruction that was appropriate to the circumstance for testing to demonstrate SBLC pump and valve operability. Specifically, Procedure 0255–02–III incorporated a testing methodology that inappropriately manipulated the SBLC pump prior to obtaining as-found-data, thus resulting in unacceptable pre-conditioning. Because this violation was of very low safety significance and was entered into the licensee's corrective action program (CAP 1429810), it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000263/2014003–01; Inadequate Standby Liquid Control Quarterly Pump and Valve Test Due to Proceduralized Unacceptable Preconditioning)**

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on May 27, 2014, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed table-top emergency response operations in the Technical Support Center to determine whether the event classification, notifications, and protective action recommendations

were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

This emergency preparedness drill inspection constituted one sample as defined in IP 71114.06–06.

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 performance indicator (PI) for the period from the second quarter 2013 through the first quarter 2014. 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 7, dated August 31, 2013, 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 IRs for the period of April 1, 2013, through March 31, 2014, 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 Reactor Coolant System (RCS) Leakage PI for the period from the second quarter 2013 through the first quarter 2014. 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 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator logs; RCS leakage tracking data; issue reports; event reports and NRC Integrated IRs for the period of April 1, 2013, through March 31, 2014, 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.

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 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 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 and human performance 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. Specifically, the inspectors focused their review on the licensee's actions to address a trend in issues associated with the NRC cross-cutting aspect H.7, "Inadequate Documentation." During Monticello's end-of-cycle performance assessment, the NRC assigned a substantive cross-cutting issue (SCCI) in H.7, to focus the site's attention and corrective actions toward arresting the degrading trend.

Inspectors reviewed the impacts of several issues with this cross-cutting aspect on plant performance and examples where these issues caused plant events. The inspectors' review nominally considered the 6-month period of October 2013 through April 2014, 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 one semi-annual trend inspection sample as defined in IP 71152–05.

b. Findings

No findings were identified. The inspectors reviewed the licensee's root cause evaluation (RCE) and corrective actions focused on addressing this trend. Inspectors determined that the cause evaluation was sufficiently broad in that it considered plant identified data from the CAP in addition to the NRC findings that resulted in the SCCI. The inspectors observed that site personnel had trending information which suggested the actions they had put in place had made some impact toward reducing the number of issues caused by cross-cutting aspect H.7. During their RCE review, inspectors questioned whether the identified root cause and associated corrective actions to prevent recurrence were too focused on operations procedure issues when the findings associated with the SCCI were more broadly scattered among different site organizations. The inspectors noted that some additional actions developed in connection with the contributing causes were intended to address H.7 issues more broadly, outside of operations. The inspectors did not identify any findings of significance.

.4 Selected Issue Follow-Up Inspection: Surveillance Requirement 3.3.3.1.2 Inadequate Surveillance Procedure Issue Resolution

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized a corrective action item documenting NRC identified concerns regarding surveillance procedure testing of containment isolation valves and their post-accident monitoring indications. Specifically, inspectors identified that the licensee had been incorrectly crediting a surveillance test which was intended to perform a channel calibration of containment isolation valve position indication on a 2-year frequency. This test was intended to meet the requirements of SR 3.3.3.1.2 for post-accident monitoring. The inspectors identified that the surveillance procedure credited steps performed on a quarterly basis to open and close the valves from the control room and observe the position indication on the control room panel. The inspectors questioned whether these actions met the definition of "channel calibration" as defined in the TSs.

Technical Specifications state, in part, that "a CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST." The inspectors also noted that the TS Bases for SR 3.3.3.1.2 stated, in part, that "a channel calibration is performed approximately at every refueling. Channel calibration is a complete check of the instrument loop, including the sensor. The test verifies the channel responds to the measured parameter with the necessary range and accuracy. For PCIV [primary containment isolation valve] position indication, the channel calibration is a comparison of a local visual check to remote control room indication to verify the PCIV's indicated position agrees with the actual position."

The inspectors questioned whether the procedure as written appropriately calibrated the full logic to ensure that the position indication lights reflected actual valve position. The licensee's evaluation agreed that additional actions were required to fully credit SR 3.3.3.1.2. The inspectors determined that this represented an inadequate

surveillance procedure. Further licensee investigation identified that there were other procedure actions that were performed on a 2-year frequency to satisfy requirements of the IST program. The inspectors noted that these procedure steps appeared to take the necessary actions to verify valve position either by local means or by reliance on other plant parameters in addition to position indication.

After further evaluation, the inspectors concluded that while the improper crediting of the SRs represented a procedural inadequacy, the licensee had other procedure steps in place to perform the required tests at the required frequencies. As a result, the inspectors concluded that this issue was not more than minor. Inspectors reviewed several evaluations performed by the licensee, as well as the corrective actions associated with this issue. The inspectors confirmed that corrective actions were in place to fix the improper crediting of a SR in the procedure, in addition to various other procedural deficiencies. Inspectors also verified that an appropriate extent of condition for other similar procedures was planned, and that the extent of condition was appropriately characterized in the corrective action process.

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

b. Findings

No findings were identified.

.5 Selected Issue Follow-Up Inspection: Use of Non-Safety-Related Parts in Structures, Systems, or Components

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized corrective action items documenting the potential selection and use of non-safety-related (NSR) parts in safety-related components. Inspectors review focused on three primary corrective action documents, CAPs 1407438, 1407369 and 1410037. Inspectors determined CAP 1407369 had been generated by the licensee based on operating experience review of a Prairie Island issue related to NSR gasket installation in place of safety-related gaskets. Corrective action documents (CAPs 1407438 and 1410037) had been generated in regard to an adverse assessment finding documented by the licensee's Nuclear Oversight Department. Inspectors evaluated these, along with numerous other CAPs, to determine the applicability and extent of NSR part use in safety-related components at Monticello. Inspectors determined through review of licensee condition and RCEs, the licensee's scope of review encompassed a period of 2009 to present. Through review of these documents, inspectors determined the licensee concluded several occurrences of NSR parts use in either safety-related limited volume systems without make-up capability or other applications where the part performed a safety-related function. For these cases, the licensee documented each issue in its CAP and through evaluation concluded the safety-related components impacted by the use of NSR parts are operable but non-conforming. The licensee has established actions to replace the identified NSR parts. The inspectors documented a licensee-identified violation of NRC requirements due to the failure to adhere to requirements associated with the establishment of measures for the selections of parts that are essential to safety-related functions of SSCs.

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

b. Findings

Enforcement aspects of this licensee-identified finding are discussed in Section 4OA7 of this inspection report.

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

.1 (Closed) Licensee Event Report 05000263/2014–001–00: Primary System Leakage Found in Recirculation Pump Upper Seal

This event, which occurred between August 9, 2013, and January 17, 2014, involved the discovery of reactor coolant pressure boundary (RCPB) leakage into the reactor building closed cooling water system (RBCCW). As a result of indications of primary water leakage from a recirculation seal cooler into RBCCW, the licensee took action to shut down the reactor in accordance with TS 3.4.4, "RCS Operational Leakage." Following the shutdown, the licensee performed testing to determine the cause of the RCPB leakage. The source of the in leakage into the RBCCW system was determined to be from the reactor recirculation system through a coil failure in the No. 12 reactor recirculation pump upper seal heat exchanger. The licensee measured the leakage during subsequent testing activities, and determined that the leakage was 0.215 gpm.

The licensee performed an equipment cause evaluation and concluded that the apparent cause for the RCPB leak was that there was no established preventative maintenance strategy in place to periodically check the condition of the heat exchanger or replace it. The licensee also performed a failure analysis, which indicated that the heat exchanger leaked due to the slow formation of a crack on the external surface of the stainless steel coil. Specifically, the licensee determined that RBCCW chemistry chlorides concentrated on the coil surface along with scale due to unrecognized boiling occurring in the lower region of the No. 12 reactor recirculation pump upper seal heat exchanger coil. The licensee concluded that the crack formed in the coil due to the unexpected high concentration of chlorides that formed on the coil surface, combined with a susceptible material and stress, resulting in intergranular stress corrosion cracking (IGSCC). Corrective actions included development of a maintenance plan for heat exchangers under similar conditions, and modifications to bypass the susceptible No. 11 and No. 12 recirculation seal coolers and route all cooling through the remaining seal cooler on each pump.

During review of a related RCE (CAP 01415802) on the organizational issues which led to the delayed recognition of the existence of RCS pressure boundary leakage, inspectors raised questions about the corrective actions associated with the causal evaluation. Specifically, inspectors were concerned that the corrective action to prevent recurrence of the organizational and operability evaluation process deficiencies may not prevent recurrence. Inspectors discussed their concerns with the licensee, and the site initiated a CAP to evaluate these concerns. The inspectors documented an NRC identified NCV and a self-revealed NCV in Monticello Integrated IR 05000263/2014–002 relating to this issue and the organizational issues associated with this event. Documents reviewed are listed in the Attachment to this report. This licensee event report (LER) is closed.

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

.2 (Closed) Licensee Event Reports 05000263/2014–002–00: Torus to Drywell Vacuum Breaker Did Not Indicate Closed During Testing and LER 05000263/2014–003–00: Torus to Drywell Vacuum Breaker Dual Indication During Testing

These events, which occurred on February 7, 2014, and again on February 11, 2014, involved the failure of a drywell to torus vacuum breaker to indicate full closed during testing activities. Specifically, on two separate occasions, during surveillance test cycling of vacuum breaker AO–2382A, the vacuum breaker showed dual indication when full closed indication was expected. In both instances, the vacuum breaker was declared inoperable. In response to the unexpected indication, operators used an alarm response procedure to cycle the valve until it indicated full closed. Both events were reported under the reporting requirements of 10 CFR 50.72 and 50.73 for discovery of events or conditions that could have prevented systems from performing their safety function.

The licensee's analysis determined that the most probable cause of the condition was inconsistent operation of the valve limit switches. Based on failure modes and effects analysis, the licensee concluded that the closed limit switches did not consistently indicate closed when the valve was, in fact, closed. As an additional input to the causal analysis, the licensee created an action to inspect vacuum breaker AO–2382A, its limit switches, and close indication circuit during the next refueling outage to confirm the results of the failure mode and effects. As part of normal baseline outage inspection activities, inspectors plan to observe or review the results of this activity when it occurs as additional validation of the licensee's conclusions.

The inspectors reviewed the corrective actions associated with the events. Corrective actions included creating a WO to inspect AO–2382A during the refueling outage, and changing the surveillance testing methodology to require performance of an alternate surveillance test described in the TS bases for cases when vacuum breaker indication is unreliable. During their review, the inspectors questioned whether the action specified in the LERs to inspect the vacuum breaker would alone correct existing equipment deficiencies to prevent recurrence. The licensee documented the inspectors' concerns in CAP 1437087. Subsequently, the licensee created an action to perform additional evaluation if the suspected cause is not confirmed. In addition, the site revised existing actions to include repair of the malfunctioning limit switches if the suspected cause is confirmed. The licensee also took action to correct the inappropriate tracking of the existing corrective action.

The inspectors documented three NCVs regarding preconditioning, the discovery of inadequate refueling outage post-maintenance tests (PMTs) of all eight vacuum breakers, and an inadequate operability recommendation in Monticello Integrated IR 05000263/2014–002 relating to these events. Documents reviewed are listed in the Attachment to this report. These LERs are closed.

This event follow-up review constituted two samples as defined in IP 71153–05.

.3 (Closed) Licensee Event Report 05000263/2014–004–00: Time to Energize Loads Greater than Surveillance Requirement

This event, which occurred on February 10, 2014, involved the licensee's discovery that the EDGs may not energize permanently connected loads within the time required by TSs. Specifically, given an actual or simulated loss of offsite power signal generated by only the TS, "Loss of Power Instrumentation," Table 3.3.8.1–1, "Function 1 Loss of Voltage Relays," the EDGs would not energize permanently connected loads until 10.2 or 10.34 seconds for Division I and Division II, respectively which exceeds the 10 second limitation of Monticello TS SR 3.8.1.12. The licensee determined that previous surveillances had credited the use of a NSR relay to actuate to bypass the affected 10 second transfer delay relays to ensure that the 10 second transfer limitation could be met.

The licensee determined that the apparent cause of the event was due to improper verification and validation on the part of licensee personnel involved in the transition from custom TSs to the improved TSs, when SR 3.8.1.12 for the EDGs was changed to include new standard wording. Specifically, the licensee failed to recognize that the new wording altered the acceptance criteria of the SR such that different testing and calibration setpoints were necessary for the loss of voltage transfer logic. Both EDGs were declared inoperable at the time of discovery, and corrective actions included changing the set points for the applicable relays to allow the 10 second acceptance criteria to be met, and revising the surveillance procedure to implement the correct testing for SR 3.8.1.12. The inspectors documented a licensee-identified NCV for this issue in the Monticello Integrated IR 05000263/2014–002. Documents reviewed are listed in the Attachment to this report. This LER is closed.

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

.4 (Discussed) Licensee Event Report 05000263/2014–007–00: Non-Compliance with Technical Specification 3.4.9, "Reactor Coolant System Pressure and Temperature Limits"

This event, which occurred on April 14, 2014, involved the licensee's discovery of a previously unrecognized failure to meet the requirements of TS 3.4.9. While performing an operating experience review, plant personnel discovered that on seven occasions during the previous 3 years, reactor pressure vessel (RPV) pressure had been lowered below 0 psig during plant startup activities. Specifically, between May 22, 2011, and February 5, 2014, a vacuum of approximately–3 psig had been drawn on the RPV six times, and in one case a vacuum of–17.5 psig was drawn, which violated the limits specified in the Pressure Temperature Limits Report (PTLR). This represented a violation of TS 3.4.9, "Reactor Coolant System Pressure and Temperature Limits." The licensee performed a causal evaluation for this issue and determined that the cause of the failure to enter the appropriate TS actions was that station personnel did not recognize a vacuum was drawn on the RPV and the implications for compliance with pressure/temperature curves.

The licensee evaluated the impacts of the partial vacuum on the structural integrity of the RPV. This evaluation examined several potential concerns that would be associated with violated the temperature and pressure limits specified in the PTLR. As a result of this analysis, the licensee concluded that the reactor vessel was not damaged by having

a partial vacuum and had significant margin to collapse. The inspectors reviewed this analysis and did not identify any concerns. However, the inspectors identified an Unresolved Item (URI) associated with the licensee's operation outside of the region defined in the PTLR, which is documented in the section below. The licensee entered this issue into their corrective action program as CAP 01425020 and CAP 01427529. Corrective actions included an action to evaluate and revise the PTLR limits and submit the changes for NRC review. Documents reviewed are listed in the Attachment to this report. This LER will remain open pending disposition of the URI.

Findings

An URI associated with TS 3.4.9, "Reactor Coolant System Pressure and Temperature Limits," was identified. Technical Specification 3.4.9 requires, in part, that RCS pressure and RCS temperature shall be maintained within the limits specified in the PTLR, which requires that RCS pressure remain at or above 0 psig. Between May 22, 2011, and February 5, 2014, Monticello RCS pressure was decreased below 0 psig several times during reactor startup activities. During an operating experience review in April 2014, the licensee noted that a vacuum of approximately–3 psig had been drawn on the RPV six times, and in one case a vacuum of–17.5 psig was drawn, which was outside of the limits specified in the PTLR. The licensee's analysis showed that there was no impact on RPV integrity due to the existence of the partial vacuum conditions. The licensee entered this issue into the CAP and initiated action to revise the PTLR limits and submit them for NRC review.

Inspectors reviewed the results of the licensee's operating review and decided that additional information was needed, including insights from NRR's more generic resolution to industrywide issues regarding TS PTLR limits, to determine whether a performance deficiency exists. **(URI 05000263/2014003–02; Operation Outside of Reactor Coolant System Pressure and Temperature Limits)**

This event follow-up review did not constitute a sample as defined in IP 71153–05.

4OA5 Other Activities

.1 Power Uprate Related Inspection Activities (71004)

a. Inspection Scope

During this inspection period, the inspectors observed several activities related to the power uprate amendment. Specific activities are documented below, and as referenced:

- Section 1R13–This section documents specific inspector reviews of EPU activities associated with Mella+ license requirement implementation activities.

b. Findings

No findings were identified.

.2 Failure to Comply with American Society of Mechanical Engineers Code Piping Design Requirements

Introduction

The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," when the licensee failed to comply with the appropriate ASME Code requirements during implementation of the temporary modification (TMOD) of the recirculation pump seal coolers. Specifically, the licensee failed to meet the ASME Code, Section III, Subsection NB 3671.3, Class I piping design requirements for the end cap joint design on the vent line in this TMOD.

Description

The seal coolers for each RCP provide cooling for the RCP seals through the safety-related recirculation seal injection piping. The purpose of the TMOD described Engineering Change (EC) 23490, “Operate 12 Reactor Recirculation Pump With One Seal Water Heat Exchanger,” was to operate the RCP P–200B with only one seal cooler as opposed to the original configuration of two seal coolers for that RCP. The TMOD configuration included a cross-tie line that connected both seal coolers for the RCP. This configuration diverted RCS flow from the seal cooler that is no longer in-service to the only seal cooler in-service that is cooling the RCP. As part of this TMOD cross-tie line, the licensee also included a one inch branched connection vent line off the cross-tie line. This vent line lead to a single globe valve and ended in a threaded pipe cap. This vent line had RCS flow through it and therefore, constituted as RCS pressure boundary piping.

The inspectors reviewed EC 23490, Revision 1, and identified that the licensee did not classify the RCS pressure boundary out to the threaded cap end of the TMOD cross-tie vent line. The licensee only considered the length of piping up to the single globe valve as part of the RCS pressure boundary and, therefore, ASME Code Class 1 piping. Title 10 CFR, Part 50.2, defines the RCS pressure boundary as all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the reactor coolant system, or

(2) Connected to the reactor coolant system, up to and including any and all of the following:

> (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment; and
>
> (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment.

The cross-tie line and connecting vent line between the seal coolers for this TMOD did not penetrate the primary reactor containment. The TMOD vent line had only one isolation valve (globe valve) along its length before ending in a threaded pipe cap. Therefore, the length of pipe immediately after that single valve on the vent line up to the threaded pipe cap was considered RCS pressure boundary in accordance with the

definition of RCS pressure boundary described in 10 CFR 50.2 (2) (ii). Furthermore, the inspectors identified that 2007 ASME Code, Section III, “Rules for Construction of Nuclear Facility Components,” Subsection NB 3671.3, prohibits the use of a threaded connection as the only pressure boundary seal for RCS pressure boundary piping. A failure of the threaded pipe cap with only one isolation valve along the length of the vent line could result in a potential unisolable leak of RCS fluid from the vent line. That could result in diversion of RCS flow from the only in-service seal cooler and potentially result in an inoperable RCP, which could challenge the operating plant stability.

Subsequent to the inspector's identification of the vent line's design deficiency, the licensee made revisions to EC 23490 and physical changes to the TMOD to incorporate the RCS pressure boundary out the vent line end cap and the end cap was socket welded. This socket welded end cap did not utilize a threaded joint design and was in compliance with ASME Code Section III design requirements for ASME Code, Class I piping.

Analysis

The inspectors determined that the failure to demonstrate the adequacy of design of the TMOD vent line by not meeting the applicable ASME Code requirements was a performance deficiency.

The inspectors determined that the performance deficiency was more than minor, and a finding because it was associated with the Design Control attribute of the Initiating Systems Cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the licensee inadequately designed the TMOD vent line such that the design was non-compliant with ASME Code, Section III, Subsection NB 3671.3, requirements and therefore, potentially challenged plant stability.

The inspectors reviewed Attachment 0609.04, "Initial Characterization of Findings," Table 3 - SDP Appendix Router. The inspectors answered 'Yes' to all of the questions in Sections A through E of Table 3 and, therefore, the finding was evaluated using the SDP in accordance with IMC 0609, "The Significance Determination Process for Shutdown Operations," Appendix G, Attachment 1, Exhibit 2, "Initiating Events Screening Questions." The inspectors answered all the questions in Exhibit 2 and determined that this finding did not increase the likelihood of a plant initiating event during shutdown operations nor did it affect any shutdown safety functions. Therefore, the finding was determined to have very low safety significance (Green).

This finding has a cross-cutting aspect in the area of Human Performance, Avoid Complacency area, because the licensee failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Specifically, the licensee failed to recognize the latent issue concerning appropriate RCS pressure boundary identification for small bore piping systems associated with the TMOD (H.12).

Enforcement

Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of the design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

The 2007 ASME Code, Section III, Subsection NB 3671.3, for Class 1 piping states, in part, that "Threaded joints in which the threads provide the only seal shall not be used."

Contrary to the above, on January 31, 2014, the licensee failed to demonstrate the design adequacy of the vent line connected to the safety-related RCP seal injection piping system. Specifically, the licensee designed the vent line, which provided venting of the safety-related RCP seal injection piping system using only a threaded joint wherein the threads provided the only seal for this RCS pressure boundary piping. This design was not in compliance with the 2007 ASME Code, Section III, Subsection NB 3671.3, requirements for a threaded end cap design on Class 1 piping.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, CAPs 01416862 and 01417186, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. As part of their immediate corrective actions, the licensee revised the TMOD vent line end cap design to comply with the ASME Code, Section III, Class 1 piping design requirements. **(NCV 05000263/2014003–03; Failure to Comply with ASME Code Piping Design Requirements)**

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 16, 2014, the inspectors presented the inspection results to Karen Fili, Site Vice President, 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.

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.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) or Severity Level IV was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV:

- Title 10 CFR 50, Appendix B, Criterion III, “Design Control,” requires measures be established for the selection and review for suitability of application of material, parts, equipment, and processes that are essential to the safety-related functions of the SSCs. Contrary to this requirement, the licensee identified that from 2009 until November 20, 2013, NSR parts were installed in safety-related components.

During operating experience review, the licensee identified the potential for NSR parts, such as gaskets and packing to be installed in safety-related systems. The licensee conducted an evaluation from 2009 forward to determine the use of both “soft” and “hard” NSR parts in limited volume safety-related systems or other applications where it performs a safety-related function. Eight occurrences of “soft” NSR parts installed in limited volume safety-related systems (SBLC, diesel fuel oil, alternate nitrogen, and HPCI) were identified. Four occurrences of “hard” NSR parts installed in limited volume safety-related systems (alternate nitrogen, high pressure core spray, and RCIC) were identified. Lastly, seven occurrences were identified by the licensee where “hard” NSR parts (key, fuses, relay, seal kit, and sealant) were installed in other applications where it performs a safety-related function. The licensee entered this issue into its corrective action program, CAPs 1407369 and 1410037, and through evaluation has determined the safety-related components impacted by the use of NSR parts are operable but non-conforming. The licensee has established actions to replace the identified NSR parts.

The inspectors determined that this issue was more than minor because, if left uncorrected, the installation of parts/materials which fail to meet requirements could lead to subsequent part failure and adverse impact the ability of safety-related equipment to perform its safety function. Specifically, the licensee failed to establish measures for the selection of parts that are essential to the safety-related functions of SSCs. As a result, the licensee installed NSR parts in safety-related systems.

The inspectors determined that this issue was associated with the Mitigating Systems cornerstone and could be evaluated using the SDP. The inspectors used IMC 0609, Attachment 0609.04, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” dated June 19, 2012, and concluded that this issue was of very low safety significance (Green) because the finding was associated with a deficiency that only affected the design or qualification of the specific SSCs.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Fili, Site Vice President
P. Gardner, Director of Site Operations
H. Hanson, Jr., Plant Manager
P. Albares, Operations Manager
M. Lingenfelter, Director of Engineering
K. Jepson, Recovery Manager
S. Mattson, Maintenance Manager
K. Petersen, Chemistry Manager
T. Hedges, Acting Radiation Protection Manager
D. Collins, Regulatory Affairs Manager (Interim)

Nuclear Regulatory Commission

K. Riemer, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000263/2014–003–01	NCV	Inadequate Standby Liquid Control Quarterly Pump and Valve Test Due to Proceduralized Unacceptable Preconditioning (Section 1R22)
05000263/2014–003–02	URI	Operation Outside of Reactor Coolant System Pressure and Temperature Limits (Section 4OA3.4)
05000263/2014–003–03	NCV	Failure to Comply with ASME Code Piping Design Requirements (Section 4OA5.1)

Closed

05000263/2014–003–01	NCV	Inadequate Standby Liquid Control Quarterly Pump and Valve Test Due to Proceduralized Unacceptable Preconditioning (Section 1R22)
05000263/2014–001–00	LER	Primary System Leakage Found in Recirculation Pump Upper Seal Heat Exchanger (Section 4OA3.1)
05000263/2014–002–00	LER	Torus to Drywell Vacuum Breaker did not Indicate Closed During Testing (Section 4OA3.2)
05000263/2014–003–00	LER	Torus to Drywell Vacuum Breaker Dual Indication During Testing (Section 4OA3.2)
05000263/2014–004–00	LER	Time to Energize Loads Greater Than Surveillance Requirement (Section 4OA3.3)
05000263/2014–003–03	NCV	Failure to Comply with ASME Code Piping Design Requirements (Section 4OA5.1)

Discussed

05000263/2014–007–00	LER	Non-Compliance with Technical Specification 3.4.9, “Reactor Coolant System Pressure and Temperature Limits” (Section 4OA3.4)

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.

Section 1R01

CAP 1388725; NRC Questions Surrounding Material in Subyard and Wind
1478; External Flood Surveillance; Revision 9
A.6; Acts of Nature; Revision 49
4 AWI-04.02.01; Revision 22

Section 1R04

2118; Plant Prestart Checklist HPCI System; Revision 16
2124; Plant Prestart Checklist Diesel Generators and Fuel Oil System; Revision 9
2154-10; HPCI System Prestart Valve Checklist; Revision 33
2154-34; ESW System Prestart Valve Checklist; Revision 28
2154-35; HPCI Hydraulic Control and Lubrication System Prestart Valve Checklist; Revision 8
2200; Plant Prestart Checklist—ESW System; Revision 2
B.08.01.04-01; ESW—Function and General Description of System; Revision 5
B.08.01.04-02; ESW—Description of Equipment; Revision 4
B.08.01.04-03; ESW—Instrumentation and Controls; Revision 11
B.08.01.04-05; ESW—System Operation; Revision 27
CAP 1345008; Leaking Boundary Valve Prevents Performance of PM
CAP 1370541; 13 ESW Pipe has Pit Below Minimum Wall
CAP 1401616; ESW 73-1 Failed Required IST Acceptance Criteria of 0187-01
NH-36041; Service Water System; Revision 103
NH-36249; P&ID (Steam Side) HPCI System; Revision 80
NH-36250; P&ID (Water Side) HPCI System; Revision 82
NH-36664; RHR Service Water and ESW Systems; Revision 85
NH-36665; Service Water System and Make-up Intake Structure; Revision 96
Operations Manual B.03.02-01; HPCI: Function and General Description of System; Revision 12
WO 463436; ESW 14-1, Repair Leaking Valve; April 21, 2014

Section 1R05

4 AWI-08.01.00; Fire Protection Program Plan; Revision 16
4 AWI-08.01.01; Fire Prevention Practices; Revision 43
CA-05-084; Combustible Loading Calculation; Revision 2
CAP 1427455; NRC Identified Leakage through Hatch-5/RB
EWI-08.23.01; Review of Proposed Changes to the Fire Protection Program; Revision 4
Strategy A.3-01-B; Fire Zone 1-B—No. 11 RHR and Core Spray Pump Room; Revision 4
Strategy A.3-01-E; Fire Zone 01-E – HPCI Room – Reactor Building Elevation 962'; Revision 7
Strategy A.3-03-A; Fire Zone 03-A – Recirculation MG Set Room; Revision 06
Strategy A.3-03-B; Fire Zone 03-B – SBLC Area (962' Rx Building East); Revision 11
Strategy A.3-23-B; Fire Zone 23-B—Intake Structure Corridor; Revision 7

Section 1R06

NF-74413-4; Underground Services Electrical Power; Revision 92
NF-74413-6; Underground Services of Division II Cable Raceway System; Revision 79
WO 481640; Underground Vaults for Water Inspection; Revision 0

Section 1R11

1040-01; Turbine – Generator; Revision 81
8300-02; External Flooding Protection Implementation to Support A.6 Acts of Nature; Revision 0
A.6; Acts of Nature; Revision 48
CAP 1435679; Possible Babbit Material on T-40 Screen
MT-LOR-14B-007L Training Presentation; A.6 External Flooding "Time Zero" to PMF Ops Training; April 24, 2014
MT-LOR-14B-007L; Licensed Operator Requalification Training—A.6 External Flooding; Revision 1

Section 1R12

CAP 1337479; RCIC Low Steam Pressure Switch Contact Resistance Concern; May 14, 2012
CAP 1355882; Blown Fuse, Drywell High Pressure Scram When Replacing Test Light; October 22, 2012
CAP 1358922; Received RCIC Equipment Area Alarm A-16 during RCIC Testing; November 13, 2012
CAP 1359114; Major Step Change in Noise Coming from Relay 16A-K50A; November 14, 2012
CAP 1365355; Pickup Voltage Adjustment Issues for HFA Relays; January 3, 2013
CAP 1365700; Unexpected Relay/Test Fixture Response during 0006 Test; January 11, 2013
CAP 1367241; Relay May Need Burnishing; January 24, 2013
CAP 1367678; Reactor Water Cleanup Isolation Time Delay Relay not Adjustable to As-Left; January 24, 2013
CAP 1373643; Unexpected System Response While Repositioning Mode Switch; March 8, 2013
CAP 1374470; Unexpected Lifted Lead; March 14, 2013
CAP 1374841; 16A-K7A Relay Failed to Energize during 0500 Test; March 16, 2013
CAP 1374862; Apparent MO-2076 Overthrust in Closed Direction During As-Found Test; March 16, 2013
CAP 1376613; MO-2076 Has Running Load Anomaly in Open Stroke; March 28, 2013
CAP 1376941; Left Auxiliary Contact on Scram Contactor 5A-K13A Not Operating; March 30, 2013
CAP 1379026; RCIC-9 Failed Inspection; April 16, 2013
CAP 1382194; 16A-K81C Timing Out of Spec High for Reactor Water Cleanup High Flow Trip; May 8, 2013
CAP 1382197; 16A-K81D Timing Out of Spec High for Reactor Water Cleanup High Flow Trip; May 8, 2013
CAP 1384890; Reactor Protection System Test Fix Light on When Parallel Contacts Closed; May 31, 2013
CAP 1385048; SV-2082B Blowing Air Out Vent; June 1, 2013
CAP 1390367; RCIC Speed Indicator Did Not Show Speed on Startup; July 17, 2013
CAP 1391708; RCIC Null Voltage Out of As-Found Tolerance; July 28, 2013
CAP 1393177; Installed Fuse is Different from Tech Manual Recommendation; August 12, 2013

CAP 1393742; 16A-K1C Relay Making Unusual Noise; August 16, 2013
CAP 1394918; Contacts on Relay 5A-K1H Need to be Burnished; August 28, 2013
CAP 1395598; High Pressure Scram Relay Contact Issue 5A-K5C 1-2; September 4, 2013
CAP 1396198; Potential Adverse Trend with Contact Resistance during Surveillances; September 9, 2013
CAP 1396323; Leakage Noted in Vicinity of AO-13-22 in Steam Chase; September 10, 2013
CAP 1397762; AO-13-22 Leakage not Recorded in Accordance with EWI-08.06.04; September 20, 2013
CAP 1397779; AO-13-22 Operability Recommendation Lacked Content; September 20, 2013
CAP 1401099; AO-13-22 Leakage Appears to have Increased; October 11, 2013
CAP 1405840; Potential through Wall Leak on Steam Line to Turbine Regulator; November 11, 2013
CAP 1405926; SI-7321 Failed to Indicate RPMs during RCIC Run; November 12, 2013
CAP 1406182; Found SY-7321 RCIC (Frequency Converter) Loose on Its Mount Base; November 13, 2013
CAP 1406235; SI-7321 (RCIC Speed Indicator) has Failed Twice in Four Months; November 13, 2013
CAP 1418253; DPIS-13-83 RCIC High Steam Flow Out of As-Found Criteria; February 10, 2014
CAP 1418298; Switches Discovered Outside of As-Found Tolerance; February 10, 2014
CAP 1418317; CV-2848 Outside Close Trend Band; February 10, 2014
Monticello Maintenance Rule Program System Basis Document; RCIC; Revision 2
Monticello Maintenance Rule Program System Basis Document; RPS; Revision 1
Operations Manual B.02.03-01; RCIC – Function and General Description of System; Revision 5
Operations Manual B.05.06-01; Plant Protection System; Revision 13

Section 1R13

12751-MNGP-OPR-01; Spent Fuel Cask Welding: 61BT/BTH NUHOMS Canisters; Revision 0
12751-MNGP-QP-9.201; Visual Weld Examination; Revision 0
12751-MNGP-QP-9.202; Color Contrast Liquid Penetrant Examination Using the Solvent Removable Method; Revision 1
8216-01; Mella+ Dynamic Test; Revision 0
B.06.06-04; Condensate Demineralizer System; Revision 27
CAP 1426595; Upon Close of 2R Xfrmr 12 Bus Bkr no Current Response on Amp Meter
CAP 1428591; NRC Question – Penetration Seal 11 EDG Room
CAP 1428678; Unexpected High Temp Alarm on No. 12 EDG Following Shutdown
CAP 1429356; No. 11 CW Pump Bay Level Divergence
CAP 1430618; C-80, Multiple Bad Signal Alarms on “B” and “C” Cond. Demins; May 14, 2014
EC 18624; Spent Fuel Loading DSCs 6A-10B; Revision 4
EC 24075; Pneumatic Block for Condensate Demineralizer Effluent Valves; June 14, 2014
ISP-NIP-0586; 2-of-4 Voter Channel Functional Test; Revision 6
ISP-NIP-0587; 2-of-4 Voter Time Response Test for 2-of-4 Voter-3; Revision 5
NH-36489; Circulating Water System; Revision 83
NH-36666; P&ID Screen Wash, Fire, and Chlorination System Intake Structure; Revision 89
NX-236555-112; Pneumatic Tubing Detail; Revision 0
ODMI 1429356; 11 CWP Low Basin Level Instrument Divergence; May 6, 2014
QF0501; 10 CFR 50.59 Screening – Pneumatic Block for Condensate Demineralizer Effluent Valves; Revision 0

Trouble Shooting Plan—Diverging 11 Circulating Water Pump Basin Level Transmitter; May 6, 2014
Troubleshooting Plan—2R to12 Bus Ammeter Indicating no Current during Transfer; April 13, 2014
USAR 11.7; Condensate Demineralizer System; Revision 28
WAP-3; TriVis Welding Administrative Procedure Control of Filler Metal; Revision 4
WO 464956-76; DSC-16 Weld 4 Minor and if Required Major Weld Repairs Includes VT and PT of Weld 5; Revision 3
WO 464956-76; DSC-16 Weld 4 Repairs Includes VT and PT; Revision 4
WO 488589-08; Convert to MELLA+ TS, TRM, and COLR with 2-of-4 Voter CDA Jumper Removal and APRM/OPRM Setpoints/settings Change; Revision 1

Section 1R15

0255-02-III; SBLC Quarterly Pump and Valve Tests; Revision 56
12751-MNGP-OPR-01; Spent Fuel Cask Welding: 61BT/BTH NUHOMS Canisters; Revision 0
12751-MNGP-QP-9.201; Visual Weld Examination; Revision 0
12751-MNGP-QP-9.202; Color Contrast Liquid Penetrant Examination Using the Solvent Removable Method; Revision 1
2010-01; Alternate Nitrogen System Data; Revision 3
2010-OWI-02.03; Operator Rounds—Turbine Building East; Revision 60-B
2212; Plant Startup Checklist Alternate Nitrogen System; Revision 8
3851; Monticello IST Preconditioning Evaluation; Revision 0
3851; Monticello IST Preconditioning Evaluation; Revision 1
4 AWI-09.04.01; IST Program; Revision 45
ACE 1429810; 0255-02-III Revised without Documenting Preconditioning Evaluation per 4 AWI-09.04.01; No Date
CA-94-017; Calculation of Alternate Nitrogen System Supply Pressure and Spare Bottle Inventory; Revision 9
CAP 1382763; TS SR 3.5.1.3 Min Required Value for AN2 May be Non-conservative
CAP 1420906; Non-conservative Error Found in Calc. 94-017; Revision 8
CAP 1425172; Strain Gauge Location Configuration Question on C MSL
CAP 1425882; Non-conservative Temp Used in 94-017-R8 for AN2 System
CAP 1429453; Operability Evaluation
CAP 1429453; Out-of-Spec Value during 0255-02-III
CAP 1429810; Preconditioning Evaluation was not Documented
EC 18624; Spent Fuel Loading DSCs 6A-10B; Revision 4
EC 23928; Evaluation of Effects on Steam Dryer Design Function due to Plant Operation through 1864 MWT; May 19, 2014
Operations Manual B.08.04; Alternate Nitrogen System; B.08.04.03-05; Alternate Nitrogen System—System Operations; Revision 17
OPR 1425172; Unable to Verify Steam Dryer EPU Stress Ratios; Revision 0
OPR 1425882; Past Operability Evaluation—AN2 System; April 16, 2014
OSP-AN2-0567; Monitor ADS Pneumatic Supply; Revision 5
QF0565; Maintenance Rule Functional, MSPI, and Equipment Reliability Clock Reset Failure; May 13, 2014
WAP-3; TriVis Welding Administrative Procedure Control of Filler Metal; Revision 4
WO 464956-76; DSC-16 Weld 4 Minor and if Required Major Weld Repairs includes VT and PT of Weld 5; Revision 3
WO 464956-76; DSC-16 Weld 4 Repairs Includes VT and PT; Revision 4

Section 1R18

01429621-03; Operations Memo—B.06.04-05 Circulating Water System Operation; Revision 0
C.6-006-C-10; Circ Wtr Pp P-100B Trip; Revision 3A
C.6-006-C-19; Circ Wtr Pp P-100A Low Basin Level; Revision 4A
C.6-006-C-21; Intake Structure E Basin Low Level; Revision 0A
C.6-006-C-31; Intake Structure W Basin Low Level; Revision 7A
CAP 1418321; SR 3.8.1.12 was not Met by TS LOP Instr Alone
CAP 1419309; EDG Compensatory Measure Implementation Issues
CAP 1429356; No. 11 CW Pump Bay Level Divergence
EC 24035; Bypass P-100A 11 CWP Low Basin Level Trip; May 5, 2014
EDG Timing Relay Issue Timeline; March 10, 2014
FG-E-SE-03; 50.59 Resource Manual; Revision 5
FP-E-SE-03; 10 CFR 50.59 and 72.48 Processes; Revision 6A
FP-OP-OB-01; Operator Burden Program; Revision 5
FP-OP-OL-01; Operability/functionality Determination; Revision 13
FP-OP-SC-02; Alternate Plant Configuration Control; Revision 0
NE-36394-2; Circulating Water Pump ACB No. 152-305 P-100A; Revision 77
NH-36489; Circulating Water System; Revision 83
NH-36666; P&ID Screen Wash, Fire, and Chlorination System Intake Structure; Revision 89
QF1139; Contingent Operator Action Guidance; Revision 2
QF1143 Alternate Plant Configuration Sheet; Relay Timing—AR 1418321 and C/O 57087; February 10, 2014
SCR-14-0073; EDG Transfer Circuit Compensatory Activity; Revision 0
SCR-14-0218; EC 24035—Bypass P-100A 11 CWP Low Basin Level Trip; Revision 0

Section 1R19

1070; RCIC Flow Control System Dynamic Test Procedure; Revision 19
CAP 1429855; Rosemount 710 MTUs in SRV LLS Subject to Part 21
CAP 1430043; P-77 Lost its Prime While Performing the Cable Cutover Task
CAP 1433996; 5A-K11C RPS Test Fixture Light on, Performance: ISP-NIP-0586
CAP 1434043; Burnish Contacts of Relays 5A-K11G and 5A-K11D
EC-24179; P-77 Diesel Oil Service Pump, Low Discharge Pressure and Flow
Instrument Calibration Worksheet; DPT-4060A SRV F Tailpipe D/P; May 7, 2014
NE-93576; Single Line Diagram 480V MCC B34; Revision 78
NH-36051; P&ID Diesel Oil System; Revision 81
NX-78-67-7; RPS; Revision 79
Operations Manual B.03.03-01; Reactor Pressure Relief—Function and General Description of System; Revision 13
Operations Manual B.03.03-03; Reactor Pressure Relief—Instrumentation and Controls; Revision 7
Operations Manual B.03.03-06; Reactor Pressure Relief—Figures; Revision 8
WO 449182-01; SRV F Tailpipe D/P—Replace Transmitter; May 7, 2014
WO 449182-02; SRV F Tailpipe D/P—PMT; May 7, 2014
WO 449189-02; SRV H Low Low Set Tailpipe D/P—Replace Transmitter; May 6, 2014
WO 449189-03; SRV H Low Low Set Tailpipe D/P—Post Maintenance Test; May 6, 2014
WO 458885; 1070 RCIC Flow Control System Dynamic Test Procedure; May 12, 2014
WO 492209; Channel A2 Relays—Burnish Contacts; June 11, 2014
WO 503246; P-77 Diesel Oil Service Pump Not Providing Flow to T-45A; June 8, 2014

WO 503246-04; OPS – P-77 PMT; June 9, 2014
WR 103302; P-77 Lost its Prime While Performing the Cable Cutover Task; May 9, 2014

Section 1R22

0143; Drywell-Torus Monthly Vacuum Breaker Check; Revision 39
0187-02; 12 EDG/12 ESW Quarterly Pump and Valve Tests; Revision 85
0213; Drywell to Suppression Chamber Vacuum Breaker Leakage Operational Check; Revision 012A
0255-02-III; SBLC Quarterly Pump and Valve Tests; Revision 56
0301; Safeguard Bus Voltage Protection Relay Unit Functional Test; Revision 38
3851; Monticello IST Preconditioning Evaluation; Revision 0
3851; Monticello IST Preconditioning Evaluation; Revision 1
4 AWI-09.04.01; IST Program; Revision 45
ACE 1429810; 0255-02-III Revised without Documenting Preconditioning Evaluation per 4 AWI-09.04.0; No Date
CAP 1417977; Failure of Drywell-torus Vacuum Breaker to Close
CAP 1417977; Failure of Drywell-torus Vacuum Breaker to Close
CAP 1428678; Unexpected High Temp Alarm on No. 12 EDG Following Shutdown
CAP 1429453; Operability Evaluation
CAP 1429453; Out-of-Spec Value during 0255-02-III
CAP 1429810; Preconditioning Evaluation was not Documented
CAP 1432547; NRC Resident Questions Critical Step Consistency
ODMI 1417977; Failure of Drywell-torus Vacuum Breaker to Close
QF0565; Maintenance Rule Functional, MSPI, and Equipment Reliability Clock Reset Failure Evaluation –Torus-DW Vacuum Breaker; March 12, 2014
QF0565; Maintenance Rule Functional, MSPI, and Equipment Reliability Clock Reset Failure; May 13, 2014
WO 486467-01; 12 EDG/12 ESW Quarterly Pump and Valve Tests; April 29, 2014
WO 488421; No. 031 Safeguard Bus Voltage Protection Relay Unit Functional Test; May 29, 2014

Section 1EP6

Monticello Emergency Plan Table-Top Drill w/DEP May 13, 2014 – June 10, 2014; Revision 0

Section 4OA1

0533; Containment Sump Flow Measurement Instrumentation; Revision 22
CAP 1416664; RCS Spec Activity – Misreporting of per Find for 2013
CAP 1433055; NRC Resident Inspector Requests Information on Indicator
CAP 1433144; QF0445 Did Not Reflect MT Expectation and Cross Referenced
EC 23635; Main Exhaust Plenum Airlock Breach SSFF Evaluation; Revision 0
FP-PA-PI-02; NRC/INPO/WANO Performance Indicator Reporting; Revision 8
NEI 99-02; Regulatory Assessment PI Guideline; Revision 7
PCR 1426303; QF0445—Add Comment from NEI 99-02 to Each KPI Page
QF0445; NRC/INPO/WANO Data Collection and Submittal—RCS Total Leakage; 2nd Qtr 2013
QF0445; NRC/INPO/WANO Data Collection and Submittal—RCS Total Leakage; 3rd Qtr 2013
QF0445; NRC/INPO/WANO Data Collection and Submittal—RCS Total Leakage; 4th Qtr 2013
QF0445; NRC/INPO/WANO Data Collection and Submittal—RCS Total Leakage; 1st Qtr 2014

QF0445; NRC/INPO/WANO Data Collection and Submittal—SSFF; 1st Qtr 2014
QF0445; NRC/INPO/WANO Data Collection and Submittal—SSFF; 2nd Qtr 2013
QF0445; NRC/INPO/WANO Data Collection and Submittal—SSFF; 3rd Qtr 2013
QF0445; NRC/INPO/WANO Data Collection and Submittal—SSFF; 4th Qtr 2013
RCS Leakage Log Entries; April 1, 2013 through March 31, 2014

Section 4OA2

0255-10-IA-1; PCIV Exercise; Revision 42
2167-01; Startup Checklist Transition from Mode 3 to Mode 2; Revision 23
CAP 1366595; NSR Gaskets May Be Installed in Place of SR in the Plant
CAP 1374981; Incorrect Cable Cut during Demolition per WO 00416771-05, "A" RHR Cable Cut Instead of CSP
CAP 1378744; E SRV Low-low Set Tailpipe dP Root Valve Found Closed
CAP 1386536; Breaker 152-101 Fault Resulting in LONOP and INOP Both EDGs
CAP 1394877; Recirc Runback No. 12 Recirc MG Set
CAP 1397500; RETS/REMP NRC 2013: Dispersion Parameters in ODCM-APP-A
CAP 1399730; NRC Findings in H.7 (Documentation) Crosscutting Aspect; Revision 2
CAP 1399730; NRC Findings in H.7 (Documentation) Crosscutting Aspect; Revision 1
CAP 1404263; NRC Question on SR 3.3.3.1.2
CAP 1404581; NRC Question Regarding TS SR 3.3.3.1.2 for Function 6
CAP 1406283; Failure to Evaluate the Effects of the HPCI Steam Isolation Valve Closure Time Increase
CAP 1407369; PI AR01366595 NSR Gaskets Installed in Place of SR Gaskets
CAP 1410037; Rapid OE: NOS AAF-Non-SR Parts Installed into SR Applications
CAP 1410157; NRC Question not Resolved by Date Committed
CAP 1411612; Non-conforming Gasket Installed on RV-11-39A; SBLC RV
CAP 1411617; Non-conforming Gasket Installed on RV-11-39B; SBLC RV
CAP 1411618; Non-conforming Gasket Installed on XP-3-1 SBLC Discharge Check
CAP 1411620; Non-conforming Gasket Installed on XP-3-2 SBLC Discharge Check
CAP 1411621; Non-conforming Gasket Installed on P-11 FO Transfer Pump
CAP 1411984; Additional Requests from NRC on NSR Gasket Issue
CAP 1414015; Non-conforming Gasket Installed on Inlet of RV-4236
CAP 1414038; Non-conforming Gasket Installed on AI-599
CAP 1414087; Non-conforming Gasket Installed on AI-687
CAP 1414091; Non-conforming Gasket Installed on PCV-4214
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CAP 1414992; Non-conforming Pressure Regulator Installed on CV-2104
CAP 1415000; Non-conforming Diaphragm Installed in CV-2065
CAP 1416177; Document EOC Result for Tier 2 of EOC for RCE 01407438
CAP 1416402; Safety Related Gasket Requirement Clarification Needed
CAP 1416791; Potential NSR Key Installed in MO-2034
CAP 1416833; NSR Fuses Installed in MCC B3345
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CAP 1416848; Question from NRC Region on use of NSR Gaskets in Drywell
CAP 1416922; NRC Request for Documentation
CAP 1417088; NSR Relay Installed in 14A-K2A
CAP 1417093; NSR Seal Kits Installed in AO-1825A/B
CAP 1417173; NSR Sealant Potentially used on the 11 and 12 EDG
CAP 1417174; NSR Sealant Potentially used on the HPCI Turbine

CAP 1427455; Water Observed Dripping in 'A' RHR Room by NRC
CAP 1434257; RCE Corrective Action not Specific
CAP 1434671; Review of RCE EOC Actions for Risk Sensitivity
CAP 1434675; RCE Effectiveness Measure Inadequate
CE 1404581-02; NRC Questions on Procedure 0255-10-IA-1
CE 1410037; NSR Gaskets Installed in SR Components
CSM 2009-007-01; Classification of Subcomponents and Material – Gaskets, Spiral Wound, Corrugated Iron, Metal Jacketed, and Sheet Type; March 21, 2011
CSM-2009-00; Classification of Subcomponents and Material – Gaskets, Spiral Wound, Corrugated Iron, Metal Jacketed, and Sheet Type; December 8, 1999
CSM-2009-005; Classification of Subcomponents and Material – Pressure Seal Gasket; April 1, 2009
EC 23142; PCIV Calibration Requirements; Revision 0
EC 23285; PCIV Calibration Requirements; Revision 0
FP-SC-GEN-02; Requesting Material; Revision 11
WO 494671-01; 11 SLC Pump Discharge Check Valve NSR Bonnet Gasket Replacement
WO 494676-01; 12 SLC Pump Discharge Check Valve NSR Bonnet Gasket Replacement

Section 4OA3

CAP 1415225; Primary System Leakage Determined into RBCCW
CAP 1415802; RBCCW In-leakage Historical Review of Event
CAP 1417977; Failure of Drywell-torus Vacuum Breaker to Close
CAP 1418321; SR 3.8.1.12 was not Met by TS LOP Instr. Alone
CAP 1418471; AO-2382A Torus-to-DW Vac Breaker Closed Indication Anomaly
CAP 1419309; EDG Compensatory Measure Implementation Issues
CAP 1420318; DW-torus Vacuum Breaker Work Performed with Inadequate PMT
CAP 1420597; CAP not Written for Extent of Condition
CAP 1421104; NRC Questions on Torus to Drywell Vacuum Breakers
CAP 1421323; NRC Question Regarding DW-torus Vacuum Breaker PMT
CAP 1421809; OPR for Torus-drywell Vac Breakers Required Multiple Revisions
CAP 1422587; RCPB Leak Resulted in a Condition Prohibited by Tech Specs
CAP 1425020; Procedure Allows Vacuum on RPV Outside of PTLR Limits
CAP 1425443; Six NRC Findings in H.14 Cross-cutting Aspect
CAP 1427529; 60 Day LER Required for PTLR violation
CAP 1434385; CAP 01426035 May have been Closed Inappropriately
CAP 1437087; NRC Question on Tracking Action to Inspect AO-2382A in RFO
EC 23635; Main Exhaust Plenum Airlock Breach SSFF Evaluation; Revision 0
EC 23788; SSFF Evaluation for CAP 01418321; Revision 0
EC 23793; SSFF Evaluation for AR 1417977 and AR 1418471 (PCT Vacuum Breaker Dual Indication during Testing on February 7, 2014, and February 11, 2014); Revision 0
EC 23793; Supplement; Vacuum Breaker SSFF NRC Question Response; April 17, 2014
EC 23962; Structural Integrity of the RPV Under a Vacuum; Revision 0
EDG Timing Relay Issue Timeline; March 10, 2014
FP-OP-OL-01; Operability/functionality Determination; Revision 13
QF1139; Contingent Operator Action Guidance; Revision 2
WO 497230; ELEC- RFO 27 Investigate Repair as Required AO-2382A
WR 100695; RFO 27 Investigate Repair as Required AO-2382A

Section 4OA5

CAP 1416423; Additional Support Needed for P-200A Seal Cooler Line TMod
CAP 1416546; Venting of New Line in EC-23490 Not Adequately Addressed
CAP 1416586; As-found Discrepancy of RBCCW Orifice to 12 Rec Pump Hx
CAP 1416748; Critical Path Work Delays -11 Recirc Pump T-mod Vent Piping
CAP 1416848; Open NRC Question on the Use of NSR Gaskets in the Drywell
CAP 1416862; Open NRC Question on the Use of a Single Vent Valve in the Vent Line Added as a Result of the Temporary Modifications Installed on REC
CAP 1417186; Potential Non-conformance on Rx Coolant Pressure Boundary
CAP 1417375; REC Design Question – Class 600 Valves in Recirc System
EC 23513; Operate 11 Reactor Recirc Pump with One Seal Water Heat Exchanger; Revision 0
EC 23513; Operate 11 Reactor Recirc Pump with One Seal Water Heat Exchanger; Revision 1
EC 23513; Operate 11 Reactor Recirc Pump with One Seal Water Heat Exchanger; Revision 2
OPR 1417186-01; Recirc System Vent Lines; Revision 0
OPS-XR-32-2; PMT, 12 Recirc Pump Inst. Line Vent; Revision 7
T-MOD EC-23490; Operate 12 Recirc Loop with One Heat Exchanger, Operations Acceptance of T-Mod Installation; Revision 0

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
AWI	Administrative Work Instruction
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
DSC	Dry Storage Canister
EC	Engineering Change
EDG	Emergency Diesel Generator
EPU	Extended Power Uprate
ESW	Emergency Service Water
Gpm	Gallons per Minute
HPCI	High Pressure Coolant Injection
IGSCC	Intergranular Stress Corrosion Cracking
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
IST	In-service Test
LER	Licensee Event Report
MELLA+	Maximum Extended Load Line Limit Plus
MWt	Megawatts Thermal
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NSR	Non-Safety-Related
NUMARC	Nuclear Management and Resources Council
OM	Operation and Maintenance
PARS	Publicly Available Records System
PCIV	Primary Containment Isolation Valve
PI	Performance Indicator
PM	Post Maintenance
PMT	Post-Maintenance Testing
psig	Pounds Per Square Inch Gauge
PTLR	Pressure Temperature Limits Report
RBCCW	Reactor Building Closed Cooling Water
RCE	Root Cause Evaluation
RCIC	Reactor Core Isolation Cooling
RCP	Recirculation Pump
RCS	Reactor Coolant System
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SBLC	Standby Liquid Control
SCCI	Substantive Cross-Cutting Issue
SDP	Significance Determination Process
SR	Surveillance Requirement

SRV	Safety Relief Valve
SSC	Structure, System and Component
TMOD	Temporary Modification
TS	Technical Specification
TSO	Transmission System Operator
USAR	Updated Safety Analysis Report
URI	Unresolved Item
WO	Work Order

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Sincerely,

/RA/

Kenneth Riemer, Branch Chief
Branch 2
Division of Reactor Projects

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