



**UNITED STATES
NUCLEAR REGULATORY COMMISSION**

REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

December 30, 2013

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
EVALUATIONS OF CHANGES, TESTS, AND EXPERIMENTS, PERMANENT
PLANT MODIFICATIONS BASELINE INSPECTION, AND POWER UPRATE
INSPECTION REPORT 05000263/2013007**

Dear Ms Fili:

On November 22, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications inspection at your Monticello Nuclear Generating Plant. The enclosed inspection report documents the inspection results which were discussed November 22, 2013, with Ms. K. Fili and other members of your staff.

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

Four NRC identified findings of very low safety significance (Green) were identified during this inspection. The four findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because the issues were entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, - Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Monticello Nuclear Generating Plant. In addition, 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.

K. Fili

-2-

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at <http://www.nrc.gov/reading-rm/adams.html> (the Public Electronic Reading Room).

Sincerely,

/RA by R. A. Langstaff for/

Robert C. Daley, Chief
Engineering Branch 3
Division of Reactor Safety

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

Enclosure: Inspection Report 05000263/2013007;

w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ™

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

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

Report No: 05000263/2013007

Licensee: Northern States Power Company, Minnesota

Facility: Monticello Nuclear Generating Plant

Location: Monticello, MN

Dates: November 4 - 8, 2013, and
November 18 - 22, 2013

Inspectors: Andrew Dunlop, Senior Reactor Engineer, Lead
George Hausman, Senior Reactor Engineer
Ijaz Hafeez, Reactor Engineer

Approved by: Robert C. Daley, Chief
Engineering Branch 3
Division of Reactor Safety

Enclosure

SUMMARY

IR 05000263/2013007; 11/04/2013 – 11/22/2013; Monticello Nuclear Generating Plant; Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications, Component Design Basis Inspection.

This report covers a two-week announced baseline inspection on evaluations of changes, tests, or experiments, permanent plant modifications, and power uprate. The inspection was conducted by Region III based engineering inspectors. Four Green findings were identified by the inspectors. Four of the findings were considered Non-Cited Violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (i.e. greater than Green, or Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0310, "Components Within the Cross Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated June 7, 2012. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to evaluate the effects of increasing the high pressure coolant injection (HPCI) steam isolation outboard valve allowed closure time from 40 to 50 seconds in several documents. Specifically, the licensee failed to evaluate the effect of the increased allowed closure time for MO-2035 in several analyses. The licensee entered this issue into their Corrective Action Program, where the licensee is reviewing the impact of increasing the allowed closure time for MO-2035 on high energy line break (HELB) calculations and will revise the applicable analyses and documentation as required. A preliminary analysis using actual stroke and delay times for MO-2035 verified the 55 seconds used in the analysis was still bounding.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low safety significance (Green) because the finding was a design deficiency that did not result in a loss of operability or functionality. This finding has a cross-cutting aspect in the area of Human Performance, Resources because the licensee did not have complete, accurate, and up-to-date design documentation. Specifically, the licensee failed to revise all affected design documentation when the HPCI steam isolation outboard valve allowed closure time was increased from 40 seconds to 50 seconds. [H.2(c)] (Section 1R17.2.b(1))

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to verify that required analysis was in-place prior to superseding CA-00-082. Specifically,

the licensee failed to recognize that the superseded calculation contained required analysis that was not verified in other current calculations. The licensee entered this issue into their Corrective Action Program where the licensee performed a preliminary analysis that verified the HPCI HELB was still bounded by the main steam line break analysis and to ensure that the analysis will be restored consistent with the provisions of CA-00-082 and License Amendment 117.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was a design deficiency that did not result in a loss of operability or functionality. This finding has a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee did not use human error prevention techniques, such as self and peer checking, to ensure that work activities were performed safely. Specifically, the licensee failed to recognize that the superseded calculation contained required analysis that was not verified to be in other current calculations. [H.4(a)] (Section 1R17.2.b(2))

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to maintain seismic qualification of safety-related undervoltage (UV) relay 27-43A, where the UV relay's coil was replaced without proper analysis and documentation. Specifically, the licensee did not ensure there was proper test analysis and documentation in-place that specified the requirements to allow replacement of the UV relay's coil to maintain its seismic qualification. The licensee entered this finding into their Corrective Action Program to address the cause that lead to this issue. The relay had previously been replaced with a qualified component prior to this inspection.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low safety significance (Green) because the finding was a design deficiency that did not result in a loss of operability or functionality. This finding has a cross-cutting aspect in the area of Human Performance, Decision-Making because the licensee did not make safety significant decisions using a systematic process, especially when faced with unexpected plant conditions, to ensure safety is maintained. Specifically, the licensee failed to recognize that to maintain seismic qualification, proper analysis and documentation must be in-place to identify those components that are authorized to be replaced without invalidating the seismic qualification analysis. [H.1(a)] (Section 1R17.2.b(3))

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure the emergency diesel generator (EDG) fuel oil system original design met the single failure criteria with respect to having two safety-related pumps that were physically separated and provided with independent piping and safety-related power source. The licensee entered this finding into their Corrective Action Program and implemented actions that included separating the fuel oil system into individual trains for each EDG, providing each

pump with safety-related power, and tracking the final resolution of this issue to completion.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low safety significance (Green) based on a Detailed Risk-Evaluation performed by the Senior Reactor Analysts. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3.b.(1))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

.1 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed six safety evaluations performed pursuant to Title 10, *Code of Federal Regulations* (CFR) 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 13 screenings and one applicability determination where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, or experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issue requiring the change, tests or experiment was resolved;
- the licensee conclusions for evaluations of changes, tests, or experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations, and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

This inspection constituted six samples of evaluations and 14 samples of screenings and/or applicability determinations as defined in IP71111.17-04.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed nine permanent plant modifications that had been installed in the plant during the last three years. This review included in-plant walkdowns for portions of the modified residual heat removal service water and 13.8KV switchgear. The modifications were selected based upon risk significance, safety significance, and complexity. The inspectors reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;
- the changes were in accordance with the specified design requirements;
- the procedures and training plans affected by the modification have been adequately updated;
- the test documentation as required by the applicable test programs has been updated; and
- post-modification testing adequately verified system operability and/or functionality.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an Attachment to this report.

This inspection constituted nine permanent plant modification samples as defined in IP 71111.17-04.

b. Findings

(1) Failure to Evaluate the Effects of the HPCI Steam Isolation Valve Closure Time Increase

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to evaluate the effects of increasing the high pressure coolant injection (HPCI) steam isolation outboard valve allowed closure time from 40 to 50 seconds in several documents. Specifically, the licensee failed to evaluate the effect of the increased allowed closure time for MO-2035 in several analyses.

Description: The inspectors reviewed EC 20039, "Logic Change to Provide Margin to MO-2035 and No. 16 Battery." The purpose of the modification was to address the minimum direct current (dc) voltage to HPCI steam isolation outboard valve MO-2035, where HPCI initiation could occur simultaneously with a Group IV isolation. During this event, the dc voltage would be adequate to close and seat MO-2035, but not adequate to stop on torque as required. In addition, the modification addressed low margin on the 250 Volts direct current (Vdc) Battery No. 16.

Normally open valve MO-2035 is on the steam line to the HPCI turbine. This valve and MO-2034 provide primary containment and HPCI steam line break isolation. As a

result of this modification, MO-2035's allowed closure time increased from 40 to 50 seconds. In the event of a high energy line break (HELB) of the HPCI steam line, the bounding closure time assumed with respect to a Group IV isolation to mitigate the HPCI HELB was 55 seconds.

The inspectors' review of SCR-12-0559, "HPCI Logic Change to Provide Margin to MO-2035 and No. 16 Battery," Attachment 1, revealed that the Safety Evaluation Report (SER) by the Office of Nuclear Reactor Regulation related to Amendment No. 117, dated March 7, 2001, stated that "In support of the proposed changes to the HPCI high steam flow isolation, the licensee performed a HPCI line break mass release analysis, Monticello Calculation CA-00-082."

The purpose of CA-00-082, "HPCI Line Break Mass Release Analysis for the Elimination of the 150,000 lbm/hr Time Delayed Isolation," was to develop the analysis for the mass release from a HPCI steam line break for various scenarios to support License Amendment 117 activities. The calculation stated the HPCI mass release was equal to the HPCI mass flow rate times the isolation time. The calculation identified the isolation time as the sum of the logic initiation time (2 seconds), isolation delay time (7 seconds), and valve closure time (40 seconds), for a total isolation time of 49 seconds. As a result of EC 20039, MO-2035's allowed closure time would have increased from 49 to 59 seconds and would have exceeded the bounding or limiting isolation time of 55 seconds used in the HPCI HELB analysis. Therefore, the inspectors were concerned that the effects of the revised MO-2035 stroke time were not adequately evaluated by EC 20039. As a result, the licensee conducted an initial review of HELB calculations and determined that the following analyses were affected by EC 20039:

- CA-00-082, "HPCI Line Break Mass Release Analysis for the Elimination of the 150,000 lbm/hr Time Delayed Isolation,"
- CA-96-078, "High Energy Line HPCI HELB in the HPCI Building,"
- CA-97-042, "High Energy Line HPCI HELB in the Steam Chase,"
- CA-97-146, "GOTHIC HELB Project Verification,"
- CA-10-219, "HPCI HELB Mass and Energy Release Rate for Transient Reactor Pressure,"
- CA-11-046, "Reactor Building HELB Equipment Thermal Lag,"
- EC 17914, "MCC311 and MCC312 Thermal Lag for EPU," and
- EC 14637, "MCC 311 and 312 Thermal Lag."

The licensee initiated Action Request (AR) 01405367, AR01405518, and AR01406283 to review the impact of increasing the allowed closure time for MO-2035 on HELB calculations and will revise the associated documentation as required. A preliminary analysis using actual stroke and delay times for MO-2035 verified the 55 seconds used in the analysis was still bounding.

Analysis: The inspectors determined that failure to evaluate the effects of increasing the HPCI steam isolation outboard valve MO-2035's allowed closure time from 40 to 50 seconds in several analyses was a performance deficiency. Specifically, as part of EC 20039 a number of analyses were not identified as being affected by the increase in allowed closure time. For example, if the increased allowed closure time was included in combination with the assumed process/delay times of 9 seconds for MO-2035

(i.e., 59 seconds), the HELB analysis that assumed 55 seconds closure time would no longer be bounding.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone's attribute of equipment performance and affected the cornerstone's objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the licensee's failure to incorporate the revised HPCI steam isolation allowed closure time into the plant's design basis could result in the creation of non-conservative conclusions and/or documents. Updating the affected documents was necessary to verify whether increasing the allowed closure time remained within the bounds of the HELB analysis.

The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with Inspection Manual Chapter (IMC) 0609, Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors used Exhibit 2 – "Mitigating Systems Screening Questions" for mitigating systems, structures, components and functionality. The performance deficiency affected the design or qualification of a mitigating structure, system, and component (SSC); however, the SSC maintained its operability or functionality as applicable. Specifically, a preliminary analysis using actual stroke and delay times for MO-2035 verified the 55 seconds used in the HPCI HELB analysis was still bounding. Therefore, the inspectors answered "yes" to the Mitigating Systems' Screening Question A.1 in Exhibit 2 and screened the finding as having very low safety significance (Green).

This finding has a cross-cutting aspect in the area of Human Performance, Resources because the licensee did not have complete, accurate, and up-to-date design documentation. Specifically, the licensee failed to revise all affected design documentation with the implementation of EC 20039, when the HPCI steam isolation outboard valve allowed closure time was increased from 40 seconds to 50 seconds.

[H.2(c)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, prior to November 22, 2013, the licensee failed to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions during the implementation of EC 20039. Specifically, the licensee failed to incorporate the correct HPCI steam isolation outboard valve allowed closure time into the plant's design basis. Updating the affected analyses and documents was necessary to verify whether increasing the allowed closure time remained within the bounds of the HELB analysis.

Because this violation was of very low safety significance and it was entered into the licensee's Corrective Action Program as AR01405367, AR01405518, and AR01406283, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The licensee is reviewing the impact of increasing the allowed closure time for MO-2035 on HELB calculations and will revise the associated analyses and documentation as required. (NCV 05000263/2013007-01, Failure to Evaluate the Effects of the HPCI Steam Isolation Outboard Valve Closure Time Increase)

(2) Failure to Ensure Required Design Basis Analysis was Maintained

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to verify that required analysis was in-place prior to superseding calculation CA-00-082, "HPCI Line Break Mass Release Analysis for the Elimination of the 150,000 lbm/hr Time Delayed Isolation." Specifically, the licensee failed to recognize that the calculation contained required analysis that was not verified to be in other current calculations prior to superseding CA-00-082.

Description: As described in this inspection report's Section 1R17.2.b(1), the inspectors reviewed evaluation SCR-12-0559, "HPCI Logic Change to Provide Margin to MO-2035 and No. 16 Battery." Attachment 1 of the evaluation referenced the applicable SER sections related to Amendment No. 117, which stated the following:

- *"In support of the proposed changes to the HPCI high steam flow isolation, the licensee performed a HPCI line break mass release analysis, Monticello Calculation CA-00-082. The calculation supports that the proposed HPCI isolation and valve closure times are such that the core will not be uncovered. The staff has reviewed the licensee's submittal and supporting calculation. The results of the calculation show that all applicable acceptance criteria are met with the proposed HPCI high steam flow setpoint and time delay."*
- *"Since the results of the calculations show that all applicable acceptance criteria are met with the proposed HPCI high steam flow setpoint and time delay, the resulting doses are within the acceptance criteria given in 10 CFR Part 100 and GDC-19, and the MSLB remains bounding, the proposed changes to TS Table 3.2.1 for the HPCI high steam flow isolation instrumentation are acceptable."*

The purpose of CA-00-082 was to develop the mass release from a HPCI steam line break for various scenarios to support license amendment activities. The analysis contained in CA-00-082 was superseded on November 19, 2012. As a result, the inspectors were concerned because the analysis and acceptance criteria established in CA-00-082 were not transferred to, and addressed, in other calculations.

The licensee initially stated that EC 18264, "HELB-Related Setpoint Calculation Changes," provided the justification for superseding CA-00-082 based on the revisions to CA-95-014, "Determination of HPCI High Steam Flow Instrument Setpoints (DPIS-23-76 A/B)," CA-97-146, "GOTHIC HELB Project Verification," and CA-11-143, "Radiological Evaluation of Reactor Building HELBS." However, after a subsequent review of these calculations, the licensee determined that the analysis for justifying License Amendment 117 was not included in those or any other current calculations. As a result, the licensee entered this issue into their Corrective Action Program as AR01405518, AR01406283, AR01406284, and AR01407041, where the licensee is reviewing the impact of increasing the MO-2035 valve closure time on HELB calculations and to ensure that the analysis will be restored consistent with the provisions of CA-00-082 and License Amendment 117. A preliminary analysis concluded the HPCI HELB was still bounded by the main steam line break analysis.

Analysis: The inspectors determined that failure to verify that required analysis was in-place prior to superseding CA-00-082 was a performance deficiency. Specifically, the

licensee failed to recognize that the required analysis in the superseded calculation were not transferred to other current calculations.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone's attribute of equipment performance and affected the cornerstone's objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to recognize that the superseded calculation contained required analysis that was not verified to be in other current calculations.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors used Exhibit 2 – "Mitigating Systems Screening Questions" for mitigating SSCs and functionality. The finding was a design deficiency that did not result in a loss of operability or functionality. Specifically, a preliminary analysis concluded the HPCI HELB was still bounded by the main steam line break analysis. Therefore, the inspectors answered "no" to all the Mitigating Systems Screening Questions in Exhibit 2 and screened the finding as having very low safety significance (Green).

This finding has a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee did not use human error prevention techniques, such as self and peer checking, to ensure that work activities were performed safely. Specifically, the licensee failed to recognize that the superseded calculation contained required analysis that was not transferred to other current calculations. [H.4(a)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that the licensee provide for verifying or checking the adequacy of 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.

Contrary to the above, from November 19, 2012, to November 22, 2013, the licensee failed to verify or check the adequacy of design, such as by the performance of design reviews to ensure that appropriate required analysis was in-place prior to superseding CA-00-082. Specifically, the licensee failed to transfer required analysis from the superseded calculation to other current calculations.

Because this violation was of very low safety significance and it was entered into the licensee's Corrective Action Program as AR01405518, AR01406283, AR01406284, and AR01407041, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The licensee performed a preliminary analysis that verified the HPCI HELB was still bounded by the main steam line break analysis and is reviewing the impact of increasing the allowed closure time for MO-2035 on HELB calculations and will revise the associated documentation as required. (NCV 05000263/2013007-02, Failure to Ensure Required Design Basis Analysis was Maintained)

(3) Failure to Maintain Qualification of Undervoltage (UV) Relay 27-43A

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to maintain the seismic qualification of safety-related UV relay 27-43A.

Specifically, the licensee did not ensure there was proper test analysis and documentation in-place within the commercial grade dedication (CGD) to specify the requirements to allow replacement of the UV relay's coil (i.e., a subcomponent of the UV relay) while maintaining its seismic qualification.

Description: Undervoltage relay 27-43A is one of two safety-related relays that perform redundant functions for the low pressure coolant injection (LPCI) swing bus transfer logic. The LPCI swing bus consists of MCC-133B and MCC-143B. The UV relays check for loss of voltage on the swing bus and allows closure of swing bus breaker B4300. The transfer logic is comprised of UV relays 27-43A and 27-43B, which are installed to provide a permissive to close B4300 (i.e., alternate supply to LPCI swing bus) when B3300 (i.e., normal supply to LPCI swing bus) is open. The UV relays are necessary to prevent spurious closure of B4300 with B3300 closed in the event of a fire that causes a hot short in the control cable between the two breakers. Failure of this UV relay and the UV relay 27-43B could prevent the swing bus transfer from occurring, or allow a spurious transfer in the event of a hot short, resulting in both B3300 and B4300 being closed and both divisions of the alternating current distribution system being cross tied.

The inspectors reviewed CGD-2011-011, "Coil for GE Undervoltage Relay," which was to document the commercial grade dedication of a relay coil. The relay coil is a subcomponent of the GE type NGV15 UV relays. The relay coil was to replace a defective coil that failed initial bench testing in safety-related UV relay 27-43A, which was being installed during the 2011 refueling outage as part of the EC 17436, "Appendix 'R' Hot Short Modification Project." The replacement process was followed because the licensee was unable to procure a new safety-related relay. Since only the relay's coil was defective, the licensee purchased a non-safety-related relay, removed the relay's coil, and dedicated the coil for use in safety-related UV relay 27-43A.

The inspectors noted that CGD-2011-011 stated there were applications where the relay coil was used in seismic applications. The licensee stated UV relay 27-43A needed to be seismically qualified based on ESM-00, "Monticello Nuclear Generating Plant (MNGP) Engineering Standards Manual," Section 7.1, "Seismic Qualification," which stated the following:

*All new safety related electrical equipment **SHALL** be qualified in accordance with IEEE 344-1975, "Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." This standard provides recommended practices for establishing procedures that will yield data, which verify that the Class 1E equipment can meet its performance requirements during and following one safe shutdown earthquake (SSE) preceded by a number of operating basis earthquakes (OBE).*

However, CGD-2011-011 stated that the relay coil did not require seismic qualification since the relay coil was seismically insensitive. The justification documented in CGD-2011-011 for the relay coil being seismically insensitive was based on Material Requirement Evaluation (MRE) 557, "Seismically Insensitive Items List," which stated the following:

- *This MRE documents items that have been classified as seismically insensitive. Seismically insensitive items are defined as an item or class of items whose*

performance is not affected by earthquake loads. Seismic insensitive replacement items are items which have no seismic specific attributes.

- *Coils (relay, solenoid, contactor- NOTE that only the coils are insensitive, not the host item itself)*

As part of the dedication process, the coil was installed, calibrated, and functionally tested prior to returning the relay to service on June 25, 2011, as part of EC 17436.

Based on the above described activities, the inspectors were concerned the seismic qualification was not maintained for the repaired UV relay. This was based on the lack of proper analysis, documentation, and installation instructions that were not in-place to identify those components that were authorized to be replaced without invalidating the seismic qualification analysis (i.e., the test report's Institute of Electrical and Electronics Engineers (IEEE) 344-1975 seismic qualification).

The inspectors reviewed work order (WO) 00420652, which installed the replacement coil. Task 11, page 2, identified that in addition to the relay's coil being replaced in UV relay 27-43A, the WO stated:

Removed and replaced the "... rheostat, bridge and SCR ..." to have "... the old relay working IAW manufacturers instruction. Contact I&C [Instrumentation and Control] for soldering support for removal and installation of parts on the relay."

The inspectors noted a lack of recorded and verifiable documentation to accomplish the above described steps. For example, the type of solder used to re-solder the connections, the applicable procedure, and the specific I&C individuals performing the work were not identified in the WO. The licensee could not determine who the I&C technicians were or definitively determine the type of solder used due to the fact that no documentation could be found to validate that safety-related solder (Resin Core Wire, 63 percent tin, 37 percent lead) was used. As a result, the licensee issued AR01407385, which indicated that documentation requirements in MWI-8-M-4.12, "Soldering of Electrical Connections," were not followed.

The inspectors questioned if there was any documentation that discussed maintaining the IEEE 344-1975 seismic qualification of the relay, when certain subcomponents (i.e., specifically the replacement of the coil per CGD-2011-011) were replaced. The licensee stated that the vendor was contacted and concerns were identified that would affect maintaining the relay's seismic qualification. As a result, the licensee issued AR01409551 to address this issue with an apparent cause evaluation.

The licensee stated that the relay had been removed from the plant on May 10, 2013, when the pickup and dropout setpoints were found out-of-specification and could not be calibrated to within acceptance criteria limits. The licensee stated, although, UV relay 27-43A was unable to be calibrated within the acceptance criteria limits, it would still have performed its function based on a Maintenance Rule Functional Failure Evaluation. The licensee replaced the out-of-specification relay with a like-for-like safety-related spare. The replacement relay was satisfactorily calibrated and tested prior to its return to service.

Analysis: The inspectors determined that failure to maintain seismic qualification of safety-related UV relay 27-43A was a performance deficiency. Specifically, the licensee did not ensure there was proper test analysis and vendor documentation in-place that specified the requirements to allow replacement of the UV relay's coil to maintain its seismic qualification.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems' cornerstone's attribute of design control and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to recognize that to maintain seismic qualification, proper analysis, and documentation must be in-place to identify those components that are authorized to be replaced without invalidating the seismic qualification analysis.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors used Exhibit 2 – "Mitigating Systems Screening Questions" for mitigating SSCs and functionality. The performance deficiency affected the design or qualification of a mitigating SSC; however, the SSC maintained its operability or functionality. Specifically, failure of relay 27-43A due to its lack of seismic qualification would not prevent functioning of the LPCI loop select logic. Therefore, the inspectors answered "yes" to the Mitigating Systems' Screening Question A.1 in Exhibit 2 and screened the finding as having very low safety significance (Green).

This finding has a cross-cutting aspect in the area of Human Performance, Decision-Making because the licensee did not make safety-significant decisions using a systematic process, especially when faced with unexpected plant conditions, to ensure safety is maintained. Specifically, the licensee failed to recognize that to maintain seismic qualification, proper analysis and documentation must be in-place to identify those components that are authorized to be replaced without invalidating the seismic qualification analysis. [H.1(a)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Contrary to the above, from June 25, 2011 to May 10, 2013, the licensee failed to review for suitability of application of parts and processes essential to the safety-related functions of UV relay 27-43A, where the relay's coil was replaced without proper analysis and documentation. Specifically, the licensee failed to maintain the relay's seismic qualification by ensuring the proper analysis and documentation was in-place to identify those components that are authorized to be replaced without invalidating the seismic qualification analysis.

Because this violation was of very low safety significance and it was entered into the licensee's Corrective Action Program as AR01407385 and AR01409551, this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. Corrective actions include addressing the cause that lead to this issue. In addition, the relay had previously been replaced with a qualified component prior to this inspection. (NCV 05000263/2013007-03, Failure to Maintain Qualification of UV Relay 27-43A).

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

a. Inspection Scope

During the 2009 Component Design Bases Inspection, as documented in Inspection Report 05000263/2009007, the inspectors identified Unresolved Item (URI) 05000263/2009007-05 regarding the emergency diesel generator (EDG) fuel oil supply system. Specifically, the EDG fuel oil supply system design did not meet the single failure criteria. However, at the time of the inspection it was not clear whether the as-built design was consistent with the current or historical licensing basis. The inspectors initiated Task Interface Agreement (TIA) 2012-03, "Monticello Nuclear Generating Plant Request for Technical Assistance – Design and Licensing Basis on Diesel Fuel Oil Supply of the Emergency Diesel Generators at Monticello Nuclear Generating Plant," to have the issue reviewed by the Office of Nuclear Reactor Regulation. The TIA response, dated August 20, 2013, concluded the EDG fuel oil system design did not meet the single failure criteria with respect to having two safety-related pumps that are physically separated and provided with independent piping and safety-related power source. The inspectors reviewed the TIA response and subsequent licensee actions with respect to the EDG fuel oil supply system.

b. Findings

(1) EDG Fuel Oil Supply System Design does Not Meet the Single Failure Criteria

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the licensee's failure of EDG fuel oil system original design to meet the single failure criteria with respect to having two safety-related pumps that were physically separated and provided with independent piping and safety-related power source.

Description: The EDG fuel oil supply system consisted of two pumps, the fuel oil transfer pump (FOTP) and the fuel oil service pump (FOSP). In addition, a portable gasoline driven fuel oil pump was kept onsite if both installed pumps were unavailable. However, only the FOTP was safety-related. The FOTP was provided emergency power from EDG-12; however, the non-safety-related FOSP could only be provided power from EDG-11 through manual operations that were not part of the licensing basis for the plant as concluded in the TIA response. In addition, the piping configuration was not redundant in that both pumps discharged into common piping. As such, a failure of EDG-12 to supply power to the FOTP or failure of the common discharge piping would prevent the EDG fuel oil system from supplying either EDG with fuel oil to power the plant in a loss-of-offsite-power event.

As stated in the Monticello Updated Safety Analysis Report (USAR), Appendix E, the plant was designed to be in compliance with the 1970 Atomic Energy Commission (AEC) proposed General Design Criteria (GDC). Atomic Energy Commission Criterion 39 delineated the requirements for onsite and offsite power systems. The original Monticello plant USAR, dated October 15, 1969, EDG description (Section 4.0, Plant Standby Diesel Generator Systems) stated:

Two independent diesel generators provide redundant standby power sources. Each diesel generator is capable of providing sufficient power to safely shut down the reactor upon the loss of all outside power simultaneous with the design basis accident.

The diesel generator sets shall be complete package units with all auxiliaries necessary to make them self-sufficient power sources capable of automatic start at any time and capable of continued operation at rated full load and frequency until either manually or automatically shutdown.

Other auxiliaries required to ensure continuous [EDG] operation shall be supplied from the essential buses or control power transformers associated with the engine generator.

The AEC documented and accepted the EDG system in its SER of Monticello on March 18, 1970. The SER described the EDGs as follows:

The diesel generators are separate and independent with respect to physical location, cooling systems, air start systems, control and sequential loading circuits and fuel supplies.

The March 18, 1970, SER concluded that:

...the onsite emergency electrical power system is acceptable since no single failure should prevent power from being supplied to the engineered safety features from onsite sources.

The conclusion of the Office of Nuclear Reactor Regulation staff in response to the TIA determined that the as-built configuration for the diesel fuel oil transfer system at Monticello was not consistent with current and historical licensing and design basis documents, and/or applicable design requirements. The use of manual actions to align the non-safety-related FOSP to EDG-11 to establish the fuel oil transfer function during a loss-of-offsite-power (LOOP) event or use of a portable gasoline driven fuel oil pump was not an implied or stated action in the licensing basis and, therefore, is not part of the design basis. Therefore, the licensee was noncompliant with the NRC-approved plant design. Based on the conclusion of the TIA, the licensee would not be able to meet the single failure criteria specified in the proposed GDC, which the AEC used as the basis for concluding the onsite emergency electrical power system was acceptable.

The licensee initiated AR01394150 to address this issue. As part of the corrective actions put in place to address the concern, the licensee used the SQUG [Seismic Qualification User Group] methodology as a basis to seismically qualify the non-safety-related FOSP, closed the discharge cross-tie valves between the two fuel oil pumps to resolve a single system vulnerability due to a piping failure, installed power from a safety-related source to the FOSP, and installed switches that provide auto restart logic to both fuel oil pumps. The regulatory adequacy of the temporary modification supplying power to the FOSP will be reviewed during a future inspection. In addition, the licensee was in the design phase to install independent safety-related fuel oil transfer systems for each EDG. Based on the actions taken, the EDG fuel oil supply system was considered operable but nonconforming until installation of modifications planned for the spring 2015 outage.

Analysis: The inspectors determined the failure to meet the single failure criteria was a performance deficiency. Specifically, the EDG fuel oil system did not have two safety-related pumps that were physically separated and provided with independent piping and safety-related power source.

The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone's attribute of equipment performance and affected the cornerstone's objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to ensure the capability to supply fuel oil to the EDG-11 fuel oil day tank (FODT) with the failure of EDG-12 or the failure of the FOTP (P-11) during a LOOP event.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors used Exhibit 2 – "Mitigating Systems Screening Questions" for mitigating SSCs and functionality. The response to question 1 was No based on the finding being a deficiency affecting the design of the EDG fuel oil system where operability was not maintained. The response to Question 2 was Yes based on the finding represented a loss of the EDG fuel oil system, such that a Detailed Risk Evaluation was required.

The Senior Reactor Analysts evaluated the finding using the Monticello Standardized Plant Analysis Risk (SPAR) model, version 8.20 and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 8.0.9.0 software. The Monticello SPAR model was modified using post-processing rule changes to account for only 8 hours of EDG-11 run time following a LOOP if EDG-12 or the FOTP was failed. Two fuel oil recovery actions were also added to a new fault tree in the Monticello SPAR model (1) to allow EDG-11 to crosstie power to the non-essential bus that supplies power to the FOSP (P-77) and then start the FOSP, and (2) to allow using a portable gasoline driven fuel oil pump (P-229) to allow transferring fuel oil from the diesel oil storage tank (T-44) to the EDG-11 day tank (T-45A) in an event where the FOTP and the FOSP are not available. With no fuel oil recovery actions credited, the delta core damage frequency (Δ CDF) for the finding was determined to be 6.35E-6/year.

Two Human Error Probabilities (HEPs) were developed associated with the fuel oil recovery actions:

- (1) The HEP that the operators would crosstie EDG-11 to supply power to the non-essential bus that supplies power to the FOSP and then start the FOSP was determined using NUREG/CR-6883, "SPAR-H Human Reliability Analysis Method." Using SPAR-H, for Diagnosis, the Performance Shaping Factor (PSF) for "Stress" was determined to be "High," the PSF for "Complexity" was determined to be "Moderately Complex," with the other PSFs at a nominal value. For Action, the PSF for "Stress" was determined to be "High," with the other PSFs at a nominal value. This resulted in an HEP to manually crosstie EDG-11 to supply power to the non-essential bus that supplies power to the FOSP of 4.2E-2.
- (2) The HEP that the operators would use a portable gasoline driven fuel oil pump (P-229) to allow transferring fuel oil from the diesel oil storage tank (T-44) to the EDG-11 day tank (T-45A) in an event where the FOTP and the FOSP are not

available was also determined using the SPAR-H method. Using SPAR-H, for Diagnosis, the PSF for "Stress" was determined to be "High," the PSF for "Complexity" was determined to be "Moderately Complex," with the other PSFs at a nominal value. For Action, the PSF for "Stress" was determined to be "High," the PSF for "Complexity" was determined to be "Moderately Complex," the PSF for "Experience/Training" was determined to be "Low," the PSF for "Ergonomics" was determined to be "POOR," with the other PSFs at a nominal value. This resulted in an HEP for use of the portable gasoline-driven fuel oil pump to allow transferring fuel oil of $1.6E-1$.

To determine the Δ CDF for the finding with credit for the fuel oil recovery actions, the Deficient Case was determined using the modified Monticello SPAR model with the changes described above and solving the LOOP event trees. The Base Case was determined using the SPAR model without the changes described above (since without the SPAR modifications, the SPAR model assumed that EDG-11 had a 24-hour supply of fuel available) and again solving the LOOP event trees. The difference (i.e., Deficient Case minus Base Case) was the Δ CDF. The exposure time for the finding was the maximum of 1 year.

Using the modified Monticello SPAR model, the result for the core damage frequency (CDF) for the Deficient Case was $1.34E-6$ /year. Using the Monticello SPAR model without the SPAR modifications, the result for the CDF for the Base Case was $1.30E-6$ /year. The difference between these two values resulted in a Δ CDF of $4E-8$ /year with credit for fuel oil recovery actions. The dominant sequences are associated with a station blackout (SBO).

Based on the Detailed Risk Evaluation, the Senior Reactor Analysts determined that the finding was of very low safety significance (Green).

The inspectors determined there was no cross-cutting aspect associated with this finding because this was a legacy design issue and therefore was not reflective of current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, since initial plant licensing, the licensee failed to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, AEC Criterion 39 as discussed in the Monticello USAR, Appendix E, states "alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system, and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system." The safety-related EDG fuel oil system original design failed to meet the single failure criteria with respect to having two safety-related pumps that were physically separated and provided with independent piping and safety-related power source.

Because this violation was of very low safety significance and it was entered into the licensee's Corrective Action Program as AR01394150, this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. Corrective actions implemented included separating the fuel oil system into individual trains for each EDG, providing each pump with safety related power, and tracking for completion of final resolution. (NCV 05000263/2013007-04, EDG Fuel Oil Supply System Design does Not Meet the Single Failure Criteria)

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

.1 Routine Review of Condition Reports

a. Inspection Scope

The inspectors reviewed 17 corrective action process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations of changes, tests, or experiments. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

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

.1 (Closed) Licensee Event Report (LER) 05000263/2013-06-00: Unanalyzed Condition for Emergency Diesel Generator Fuel Oil Pumps Train

The LER stated that the root cause for the issue was still under investigation and a supplement to the LER will be submitted following completion of the investigation and would provide corrective actions to address the root cause. This issue was also the subject of the unresolved item that is described in Section 1R21 and was resolved in parallel to an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control."

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000263/2009007-05: EDG Fuel Oil Supply System Design does Not Meet the Single Failure Criteria

This URI issue is described in Section 1R21 and was resolved to an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control."

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

The following Permanent Plant Modification baseline inspection samples were also reviewed as part of the inspection of activities associated with the licensee's Extended Power Uprate licensee amendment.

- EC 11444, 2R and 1R Transformer Upgrades; and
- EC 11445, New 13.8KV Bus11 and 12 Switchgear Upgrades.

No concerns were identified.

4OA6 Meetings

.1 Exit Meeting Summary

On November 22, 2013, the inspectors presented the inspection results to Ms. K. Fili, and other members of the licensee staff. The licensee personnel acknowledged the inspection results presented and did not identify any proprietary content.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Fili, Site Vice President
J. Grubb, Plant Manager
D. Alstad, Projects Manager
J. Gausman, Design Engineering Team Lead
N. Haskell, Site Engineering Director
P. Kissinger, Regulatory Affairs Manager
D. Mattioli, Mechanical Design Engineer
M. Murphy, Regulatory Affairs Director
S. O'Connor, Regulatory Affairs
R. Stadlander, Operations Support Manager
J. Strasser, Senior Electrical Design Engineer
R. Zyduck, Engineering Design Manager

Nuclear Regulatory Commission

P. Zurawski, Senior Resident Inspector
P. Voss, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened/Closed

05000263/2013007-01	NCV	Failure to Evaluate the Effects of the HPCI Steam Isolation Outboard Valve Closure Time Increase (Section 1R17.2.b(1))
05000263/2013007-02	NCV	Failure to Ensure Required Design Basis Analysis was Maintained (Section 1R17.2.b(2))
05000263/2013007-03	NCV	Failure to Maintain Qualification of UV Relay 27-43A (Section 1R17.2.b(3))
05000263/2013007-04	NCV	EDG Fuel Oil Supply System Design does Not Meet the Single Failure Criteria (Section 1R21.1.b(1))

Closed

05000263/2009007-05	URI	EDG Fuel Oil Supply System Design does Not Meet the Single Failure Criteria (Section 4OA5.1)
05000263/2013-06-00	LER	Unanalyzed Condition for Emergency Diesel Generator Fuel Oil Pumps Train (Section 4OA3.1)

LIST OF DOCUMENTS REVIEWED

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

10 CFR 50.59 EVALUATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Revision</u>
SCR-10-0377	Specification for use of Concrete Expansion Bolts (P-503)	0
SCR-11-0080	Condensate Demineralizer Digital Controls & Larger Drain Valve	0, 1
SCR-11-0314	2R and 1R Transformer Upgrades and Associated ADL	0
SCR-11-0297	EPU-New 13.8KV Bus11 and 12 Switchgear Upgrades and Associated ADL	0
SCR-12-0178	Revision to B.09.06-05 Rev 36 for OPR in CAP01332373	0
SCR-12-0559	HPCI Logic Change to Provide Margin to M0-2035 and #16 Battery	0

10 CFR 50.59 SCREENINGS

<u>Number</u>	<u>Description or Title</u>	<u>Revision</u>
SCR-10-0079	Revision 12 of Specification MPS-0407	0
SCR-10-0168	B4319 Starter Coil Control Circuit Voltage Drop Determination	0
SCR-10-0288	Add Filter Delay to FLR-6-96 Rx Water Level Low Alarm and Associated Plant Computer Point RPVI 19	1
SCR-11-0003	B.03.02-05 HPCI Ops Manual Updates	0
SCR-11-0216	Recirc and LPCI Instrument Tubing Not Installed Using 3D Bends	0
SCR-11-0244	Revise MSIV Close Stroke Time Acceptance Band	0
SCR-12-0123	MSIV Stroke Time Settings and RHRSW System Minimum Pressure Requirements	0
SCR-11-031	EPU- 13.8 KV Condition Monitoring.	0
SCR-12-0250	Diesel Oil Transfer Pump, P-11, Suction Margin Calculation	0, 1
SCR-12-0292	Modification to Torque Seat MO-2076	0
SCR-12-0298	Update to HPCI Tech Spec Basis	0, 1
SCR-12-0570	Update to HELB Crack and Break Drawings	0
SCR-12-0595	Low Pressure Alarm Point RWM Setpoint	0
SCR-13-0251	Revision to Interconnection Resistance Acceptance Criteria	0

CALCULATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Revision</u>
CA-94-094	MCC Starter Coil Pick-up Voltages and Maximum Cable Lengths	1
CA-99-122	MO-2075 and MO-2076 Performance Analysis for Proposed Installation of Parallel Double Disc Gate Valve	0
CA-00-082	HPCI Line Break Mass Release Analysis for the Elimination of the 150,000 lbm/hr Time Delay Isolation (Superseded)	7
CA-06-104	480V MCC to Motor Terminal Voltage Drop	3
CA-08-029	Mass and Energy Release for RCIC and HPCI HELB and Critical Cracks	0
CA-10-219	HPCI HELB Mass & Energy Release Rate for Transient Reactor Pressure	0, 0A

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

<u>Number</u>	<u>Description or Title</u>	<u>Date</u>
AR01094761	Loss of Motor Cooling Flow to Div 1 RHRSW Pumps	06/01/07
AR01178713	Basket Strainers Poor/Indeterminate Quality Failed Hydro	04/30/09
AR01180558	Non Safety Related Pressure Gauges Installed for EC 14065	05/02/09
AR01202466	Adverse Trend in Double Disc Gate Valve LLRT Performance	10/14/09
AR01209851	MPS-407 Provides Guidance Inconsistent with Vendor Data	12/08/09
AR01257284	HPCI HELB Doesn't Have Correct Sequence for MOV Close	11/04/10
AR01263913	MO-2035, Incorrect Voltage used in MOV Calculation	11/22/10
AR01284234	Lower Wedge of MO-2076 Found Damaged	05/05/11
AR01304614	FSA-CDBI 2011 -250 VDC Alarm Panel Lids Not Model in Battery Calculations	09/20/11
AR01306191	Justification in 50.59 Screening does Not Support Conclusion	09/29/11
AR01363943	GAP to Industry Standard with Respect to FMEA Evaluations	12/19/12
AR01366007	USAR 05.02, Table 5.2-03B (MO-2035, EC 20039)	01/09/13
AR01377905	11 Battery Replacement Intercell Resistance Above A/C	04/11/13
AR01382342	27-43A Undervoltage Relay Found Out of Spec	05/09/13
AR01386518	Breaker 152-101 Fault Resulting in Loss of Offsite Power	06/13/13
AR01387539	2R Xfmr Lock Out during No. 11 Recirc MG Set Start ACE	07/25/13
AR01394150	NRC TIA 2012-03 Final Response EDG Fuel Oil Supply	08/21/13

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

AR01405367	Issues with Calculation 10-219	11/07/13
AR01405462	Obsv of Operators Ability to Answer some 13.8 Questions	11/07/13
AR01405465	Mod Incomplete Information Request	11/07/13
AR01405518	What Superseded Calc 00-082, 150,000 HPCI Trip	11/08/13
AR01406283	Improperly Identified Inputs Impact HELB Calcs	11/07/13
AR01406284	Information Provided is Incorrect	11/13/13
AR01407041	HPCI Flow Time Delay Calc Doesn't Support HELB	11/18/13
AR01407235	PORC Review Not Performed	11/19/13
AR01407352	OPR Procedure Conflicts with OPR Form	11/20/13
AR01407385	MWI-8-M-4.12 Requirements Not Met on WO4206	11/20/13
AR01407527	50.59 Screening 12-0595 Summary Error	11/21/13
AR01409551	Failure to Maintain IEEE Qualification of Relay	12/06/13

DRAWINGS

<u>Number</u>	<u>Description or Title</u>	<u>Revision</u>
ND-178628-19	Feedwater System HELB Break & Critical Crack Locations Feedwater Heater E-15B	77
ND-178628-20	Feedwater System HELB Break and Critical Crack Locations Feedwater Recirculation Line	77
NE-36394-18A	Emergency Service Water Pump P-111B and Scheme B4319	F
NF-239905-1	345kv-115kv Equipment Layout and Sections	BC
NF-36176	Gen. Aux Transformer and 13,800 Volt System Buses 11 and 121	82
NF-36177	Single Line Meter and Relay Diagram 4160 Volts System Buses Nos. 13, 14, 15 and 16	83
NF-36298-1	Electrical Load Flow One Line Diagram	107
NH-36051	P&ID Diesel Oil System	81
NH-36249	P&ID (Steam Side) High Pressure Coolant Injection System	79
NH-36250	P&ID (Water Side) High Pressure Coolant Injection System	82
NH-36665	P&ID Service Water System and Intake Structure	84
NX-236745-10-01	13.8 KV Bus 11 and 2R Aux Transformer Lockout Relays	1
NX-236745-20-01	13.8 KV Bus 11 Key Diagram	1
NX-236745-30-15	Connection Diagram 13.8KV Bus 11 2R Aux Transformer SEC ACB 152-101 Sheet 3 of 3	6
NX-236745-40-01	13.8 KV Bus 11 Front Elevation and Base Plan View	3

DRAWINGS

<u>Number</u>	<u>Description or Title</u>	<u>Revision</u>
NX-236746-10-01	13.8 KV Bus 12 and 1R RES Transformer Lockout Relay	2
NX-236746-40-01	13.8 KV Bus 12 Front Elevation and Base Plan View	3
NX-7822-22-5B	RCIC Steam Supply Line Isolation MO-2076 Scheme	77

MODIFICATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Revision</u>
CGD-2011-011	Coil for GE Undervoltage Relay	0
CGD-2012-007	Cooling Coil, V-AC-8A, AAF	0
CGD-2013-010	Buchanan Series 200 Terminal Block (ID 6573270 & 6573619)	0
EC 14065	RHRSW Motor Cooler Strainers	1
EC 11444	2R and 1R Transformer Upgrades	0
EC 11445	New 13.8KV Bus11 and 12 Switchgear Upgrades	0
EC 15569	Modification to Torque Seat MO-2076	0
EC 20039	Logic Change to Provide Margin to MO-2035 & #16 Battery	0
IEE-2010-016	Soak Back Lube Oil 1" Check Valve	0

OTHER DOCUMENTS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
GEF-4376	Renewal Parts NGV Undervoltage Relay	B
GEI-90805	Instructions NGV Undervoltage Relay	J
MRE 557	Seismically Insensitive Items List	5
MRE 180	EDG Air Line Filter and Parts	0
PO 00037079	Xcel Energy Purchase Order	04/06/11
PO M001079	Xcel Energy Purchase Order	04/09/11
PO PC4417MQ	Purchase Order for UV Relay	06/04/91
Xcel Internal Correspondence	Letter from J. Strasser To D. Jensen Concerning MNGP DC Load Study Report	07/20/12
4AWI-04.05.04	Conduct of Maintenance, Equivalencies and Design Changes	34

PROCEDURES

<u>Number</u>	<u>Description or Title</u>	<u>Revision</u>
B.09.06-05	4.16 KV Station Auxiliary, System Operation	36
ESM-00	Monticello Nuclear Generating Plant (MNGP) Engineering Standards Manual	19
FP-SC-PE-01	Dedication of Commercial Grade Items and Services	5
MS-0407	Specification used for Concrete Expansion Bolts (P-503)	12
MWI-8-M-4.12	Soldering of Electrical Connections	4
2154-53	1R Transformer Prestart Valve Checklist	0
2154-54	2R Transformer Prestart Valve Checklist	0
2270	Critical Safety System Checklist	8
4851-30-PM	LPCI Swing Bus Relay Calibration	14

WORK ORDERS (WOs)

<u>Number</u>	<u>Description or Title</u>	<u>Date</u>
00381797	EC-14065 Install RHRSW Motor Cooler Strainers	05/01/09
00394603	MO-2076, Perform 4900-01-PM	04/21/11
00394604	MO-2076, Disassembly and Inspection	05/11/11
00398696	Clean/Inspect P-109A Motor Cooling Supply Strainer	09/08/10
00420652	EC 17436 Eliminate Hot Short Issues at MCC-143B Appendix R	04/20/11
00452952	Perform PM 4851-30 (LPCI Swing Bus Relay Calibration)	05/07/13
00458287	Clean/Inspect P-109C Motor Cooling Supply Strainer	09/03/13
00479970	Replace 27-43A MCC-143B Undervoltage Relay	05/10/13
00481994	Remove 152-101 BRKR from Cube and Inspect Bus/BRKR	06/18/13

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AEC	Atomic Energy Commission
AR	Action Request
CAP	Corrective Action Program
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CGD	Commercial Grade Dedication
Δ CDF	Delta Core Damage Frequency
dc	Direct Current
EC	Engineering Change
EDG	Emergency Diesel Generator
FODT	Fuel Oil Day Tank
FOSP	Fuel Oil Service Pump
FOTP	Fuel Oil Transfer Pump
GDC	General Design Criteria
GE	General Electric
HELB	High Energy Line Break
HEP	Human Error Probability
HPCI	High Pressure Coolant Injection
I&C	Instrument and Control
IEEE	Institute of Electrical & Electronic Engineers
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
kV	Kilovolt
lbm/hr	Pounds per hour
LER	Licensee Event Report
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
MNGP	Monticello Nuclear Generating Plant
MRE	Material Requirement Evaluation
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
PARS	Publicly Available Records System
PSF	Performance Shaping Factor
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SDP	Significance Determination Process
SER	Safety Evaluation Report
SPAR	Standardized Plant Analysis Risk
SQUG	Seismic Qualification Utility Group
SSC	Structure, System and Component
SSE	Safe Shutdown Earthquake
TIA	Task Interface Agreement
URI	Unresolved Item
USAR	Updated Safety Analysis Report
Vdc	Volts Direct Current
WO	Work Order

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at <http://www.nrc.gov/reading-rm/adams.html> (the Public Electronic Reading Room).

Sincerely,

/RA by R. A. Langstaff for/

Robert C. Daley, Chief
Engineering Branch 3
Division of Reactor Safety

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

Enclosure: Inspection Report 05000263/2013007;
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ™

DISTRIBUTION:

Brett Rini
RidsNrrDorLpl3-1 Resource
RidsNrrPMMonticello
RidsNrrDirIrib Resource
Cynthia Pederson
Anne Boland
Steven Orth
Allan Barker
Carole Ariano
Linda Linn
DRPIII
DRSIII
Patricia Buckley
Tammy Tomczak
ROPreports.Resource@nrc.gov

DOCUMENT NAME: G: Monticello 5059Mod InspReport 2013007.docx

Publicly Available Non-Publicly Available Sensitive Non-Sensitive

To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII		RIII				
NAME	RLangstaff for ADunlop:ls		RLangstaff for RDaley				
DATE	12/30/13		12/30/13				

OFFICIAL RECORD COPY