UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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January 6, 2010

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

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT NRC COMPONENT DESIGN BASES INSPECTION (CDBI) INSPECTION REPORT 05000263/2009007(DRS)

Dear Mr. O'Connor:

On December 4, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed a component design bases inspection at your Monticello Nuclear Generating Plant. The enclosed report documents the inspection findings, which were discussed on October 2, 2009, and on December 4, 2009, with you and other members of your staff.

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

Based on the results of this inspection, five NRC-identified findings of very low safety-significance were identified. The findings involved a violation of NRC requirements. However, because of their very low safety-significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Monticello Nuclear Generating Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Monticello Nuclear Generating Plant. The information that you provide will be considered in accordance with Inspection Manual Chapter 0305.

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/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report 05000263/2009007
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No: License No:	50-263 DPR-22
Report No:	05000263/2009007(DRS)
Licensee:	Northern States Power Company, Minnesota
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello, MN
Dates:	August 31 through December 4, 2009
Inspectors:	A. Dunlop, Senior Reactor Engineer, Lead N. Féliz Adorno, Reactor Engineer, Mechanical C. Brown, Reactor Engineer, Operations J. Bozga, Reactor Engineer, Mechanical M. Munir, Reactor Engineer, Electrical S. Kobylarz, Electrical Contractor H. Campbell, Mechanical Contractor
Observers:	L. Jones, Reactor Engineer S. Edmonds, Reactor Engineer
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000263/2009007; 08/31/2009 – 12/04/2009; Monticello Nuclear Generating Plant; Component Design Bases Inspection (CDBI) and Power Uprate

The inspection was a 3-week onsite baseline inspection that focused on the design of components that are risk-significant and have low design margin; and power uprate. The inspection was conducted by regional engineering inspectors and two consultants. Five Green findings were identified by the inspectors. The findings were considered Non-Cited Violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Barrier Integrity

- Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety-significance for the failure to incorporate the actual physical configuration of the inboard main steam isolation valves (MSIVs) and the correct pneumatic system pressure drop into the pneumatic pressure requirement calculation for the inboard MSIVs. Specifically, the licensee failed to adjust the actuator moving part weight to reflect that the actuator was offset by 45 degrees instead of being mounted vertically and to correctly compute the system pressure drop. This finding was entered into the licensee's corrective action program and a preliminary calculation performed by the licensee concluded that the valves were operable.

  The finding was more than minor because it was associated with the Barrier Integrity cornerstone attribute of structures, systems, components, and barrier performance, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. This finding is of very low safety-significance (Green) because there was no actual barrier degradation. The inspectors did not identify a cross-cutting aspect associated with this finding because this was a legacy design issue and therefore was not reflective of current performance. (Section 1R21.3.b.(1))

Cornerstone: Mitigating Systems

- Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety-significance for the failure to restore the emergency service water (ESW) piping supports to their design specifications. Specifically, although the licensee identified the existence of gaps between the ESW piping supports and the baseplates, the licensee failed to recognize that this condition did not meet seismic Category 1 design basis requirements. As a result, corrective actions were not implemented. The licensee entered this issue into its corrective action program and restored the supports to their design specifications.

The finding was more than minor because it was associated with the Mitigating Systems cornerstone attribute of protection against external events and affected the cornerstone objective of ensuring the availability of the ESW system, and ultimately the emergency diesel generators (EDGs), to respond to initiating events to prevent undesirable consequences. This finding is of very low safety-significance (Green) because the design deficiency was confirmed not to result in loss of operability or functionality. This finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not properly prioritize and evaluate an identified problem. [P.1(c)]. (Section 1R21.3.b.(2))

- Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety-significance for the failure to adequately evaluate circuit loads in determining design limits in electrical calculations. Specifically, three examples were identified where the licensee: (1) failed to perform a calculation for safety-related motor starters that included all control circuit loads in determining the minimum voltage available at 120Vac starter coils, which was used to establish the coil voltage test acceptance criteria; (2) failed to include thermal overload heater and starter contact resistance when calculating the minimum voltage at 480Vac motor terminals; and (3) failed to assure that the minimum voltage at the 120Vac solenoid operated control valves was in conformance with vendor requirements. These issues were entered into the licensee's corrective action program to re-evaluate the voltage available, and to test coils, as required, to verify the pick-up voltage.

  The finding was more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of safety-related equipment to respond to initiating events to prevent undesirable consequences. This finding is of very low safety-significance (Green) because the design deficiency was confirmed, with the exception of ESW pump P-111B, not to result in loss of operability or functionality. Specifically, the failure to assure adequate voltage was available at the solenoid valves coils; and to perform periodic testing to assure the minimum voltage remained acceptable as the components aged, did not result in an impact on current operability. With respect to the ESW pump, it was determined that the pump would not have started under degraded voltage condition as required such that the ESW pump was considered inoperable. Based on a Phase III analysis, the failure of the pump to start under degraded voltage conditions was determined to be very low safety-significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding because this was a legacy design issue, therefore was not reflective of current performance. (Section 1R21.3.b.(3))

- Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety-significance for the failure to have adequate testing for safety-related equipment to monitor component degradation. Specifically, the licensee failed to verify that the motor control center contactors would continue to pick-up under degraded voltage conditions with less than the vendors' required minimum voltage. These issues were entered into the licensee's corrective action program to test the 13 contactors as soon as practicable and to revise the maintenance procedures to incorporate the requirements for periodic testing of contactors.

The finding was more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of safety-related equipment to respond to initiating events to prevent undesirable consequences. This finding is of very low safety-significance (Green) because the testing deficiency was confirmed, with the exception of ESW pump P-111B, not to result in loss of operability or functionality. Specifically, subsequent testing confirmed for nine contactors that the safety-related starter coils would still function at the calculated degraded voltage values. Although three of the contactors have not been tested, they were of a different size than the failed contactor and there appeared to be reasonable assurance based on the successful tests that these contactors also remained operable. With respect to the ESW pump, the failed test confirmed that the motor starter contactor would not pickup under degraded voltage conditions due to mechanical binding of the contactor arm such that the ESW pump was considered inoperable. Based on a Phase III analysis, the failure of the pump to start under degraded voltage conditions was determined to be very low safety-significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding because this was a legacy design issue and therefore was not reflective of current performance. (Section 1R21.3.b.(4))

Cornerstone: Mitigating Systems and Barrier Integrity

- Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety-significance for the failure of two pipe supports to meet their design requirements. Specifically, the calculation for pipe support SR-526 failed to use the minimum yield strength in determination of the allowable bending stress of the pipe support baseplate as required in the American Institute of Steel Construction code. In addition, the calculation for pipe support PS-16 failed to use the design basis concrete compressive strength in determination of the anchor bolt allowable as required in the licensee's design specification. This finding was entered into the licensee's corrective action program and a preliminary analysis performed by the licensee concluded that the pipe supports were operable but nonconforming.

  The performance deficiency for pipe support SR-526 example was more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the safety-related residual heat removal and core spray pumps. This finding is of very low safety-significance (Green) because the design deficiency was confirmed not to result in loss of operability or functionality. The performance deficiency for pipe support PS-16 example was more than minor because it was associated with the Barrier Integrity cornerstone attribute of design control and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. This finding is of very low safety-significance (Green) because there was no actual barrier degradation. The inspectors did not identify a cross-cutting aspect associated with this finding because this was a legacy design issue and therefore was not reflective of current performance. (Section 4OA5.1.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

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

The objective of the component design bases inspection is to verify that design bases have been correctly implemented for the selected risk-significant components and that operating procedures and operator actions are consistent with design and licensing bases. As plants age, their design bases may be difficult to determine and an important design feature may be altered or disabled during a modification. The Probabilistic Risk Assessment (PRA) model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones for which there are no indicators to measure performance.

Specific documents reviewed during the inspection are listed in the attachment to the report.

.2 Inspection Sample Selection Process

The inspectors selected risk-significant components and operator actions for review using information contained in the licensee's PRA and the Monticello Standardized Plant Analysis Risk (SPAR) Model, Revision 3P. In general, the selection was based upon the components and operator actions having a risk achievement worth of greater than 1.3 and/or a risk reduction worth greater than 1.005. The operator actions selected for review included actions taken by operators both inside and outside of the control room during postulated accident scenarios. In addition, the inspectors selected operating experience issues associated with the selected components.

The inspectors performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design reductions caused by design modification, or power uprates, or reductions due to degraded material condition. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as performance test results, significant corrective action, repeated maintenance activities, Maintenance Rule (a)(1) status, components requiring an operability evaluation, NRC resident inspector input of problem areas/equipment, and system health reports. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. A summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

This inspection constituted 30 samples as defined in Inspection Procedure 71111.21-05.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Updated Safety Analysis Report (USAR), Technical Specifications (TS), design basis documents, drawings, calculations, and other available design basis information, to determine the performance requirements of the selected components. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code, Institute of Electrical and Electronics Engineers (IEEE) Standards and the National Electric Code, to evaluate acceptability of the systems' design. The NRC also evaluated licensee actions, if any, taken in response to NRC issued operating experience, such as Bulletins, Generic Letters (GLs), Regulatory Issue Summaries (RISs), and Information Notices (INs). The review was to verify that the selected components would function as designed when required and support proper operation of the associated systems. The attributes that were needed for a component to perform its required function included process medium, energy sources, control systems, operator actions, and heat removal. The attributes to verify that the component condition and tested capability was consistent with the design bases and was appropriate may include installed configuration, system operation, detailed design, system testing, equipment and environmental qualification, equipment protection, component inputs and outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, system health reports, operating experience-related information, and licensee corrective action program documents. Field walkdowns were conducted for all accessible components to assess material condition and to verify that the as-built condition was consistent with the design. Other attributes reviewed are included as part of the scope for each individual component.

The following 20 components were reviewed:

- Emergency Diesel Generator (EDG) (DG 11): The inspectors reviewed the EDG loading calculation and vendor ratings for conformance with design basis load requirements. The inspectors also reviewed EDG vendor de-rating requirements for potential impact on design basis loading and operating procedures to determine whether de-rating requirements were incorporated appropriately. In addition, the inspectors reviewed surveillance testing to determine whether design basis load requirements were demonstrated during periodic load testing to satisfy TS.

- Diesel Fuel Oil Transfer and Service Pumps (P-11 and P-77): The inspectors reviewed the system hydraulic calculations including net positive suction head (NPSH) and vortexing to ensure that the pumps were capable of providing sufficient flow such that the day tanks remained filled during diesel operation. The inspection also included a review of operating procedures related to these functions. The inspectors reviewed vendor specifications and pump curves to make sure that these parameters had been correctly translated into calculations, as required. In addition, design change history, corrective actions, surveillance results, and trending data were reviewed to assess potential component degradation and impact on design margins.

- EDG 11 Room Louvers/Fan (V-SF-10): The inspectors reviewed calculations and operating procedures to ensure that the ventilation system was capable of maintaining room temperatures within design limits. The review included the control logic associated with the ventilation system to ensure it would function as designed. Seismic Category I qualification of ventilation ductwork supports in EDG building were also reviewed. The inspectors reviewed vendor specifications to verify these parameters had been correctly translated into calculations, as required. In addition, design change history, corrective actions, surveillance results, and trending data were reviewed to assess potential component degradation and impact on design margins.

- Emergency Service Water (ESW) Pump (P-111A): The inspectors reviewed the system hydraulic calculations including NPSH, vortexing, and waterhammer to ensure the pump was capable of providing sufficient flow under accident conditions. The review also included the control logic and bay water level setpoint associated with the pump to ensure it would function as designed. Operating procedures related to the pump were also reviewed. The inspectors reviewed vendor specifications and pump curves to verify these parameters had been correctly translated into calculations, as required. In addition, design change history, corrective actions, surveillance results, and trending data were reviewed to assess potential component degradation and impact on design margins. The inspectors reviewed the operability evaluation for the piping and pipe support analysis for ESW 1B and ESW 2B to verify there was a reasonable assurance of operability. The motor sizing and pump brake horsepower requirements and vendor ratings were reviewed for conformance with design basis load conditions. The inspectors also reviewed design calculations to determine the adequacy of voltage at motor terminals and motor starter contactors during degraded voltage conditions and the adequacy of motor feeder cable sizing. The motor and feeder cable coordination calculation was reviewed to determine the adequacy of protection and coordination.

- Main Steam Isolation Valve (MSIV) (AO-2-86D): The inspectors reviewed thrust calculations associated with actuator thrust and pneumatic supply, and environmental qualifications of the valve actuator to ensure the valve was capable of functioning under design conditions. In addition, the inspectors reviewed completed surveillances to ensure actual performance was acceptable and vendor specifications to ensure parameters had been correctly translated into calculations. Procedures related to the MSIVs were also reviewed. Design change history, corrective actions, surveillance results, and trending data were reviewed to assess potential component degradation, impact on design margin.

- Residual Heat Removal (RHR) Pumps (P-202A/C): The inspectors reviewed the pumps capability to meet design basis assumptions with respect to pump flow and pressure. Calculations, drawings, procedures, tests, and other analyses were reviewed to verify selected calculation inputs, requirements, and methodologies were accurate and justified, and were consistently applied. The inspectors reviewed completed surveillance tests to confirm the acceptance criteria and test results demonstrated the capability of the pump to provide required flow rates. Inservice Test (IST) results were reviewed to assess potential component degradation and impact on design margins. In addition, the licensee responses and actions to NRC Bulletin 88-04, “Potential Safety-Related

Pump Loss," were reviewed to assess implementation of operating experience. The calculation which developed the extent of containment overpressure required to satisfy NPSH was also reviewed, in addition to emergency operating procedures that directly incorporated this information. The inspectors reviewed motor sizing and pump brake horsepower requirements and vendor ratings for conformance with design basis load conditions. The inspectors also reviewed design calculations to determine the adequacy of voltage at motor terminals during degraded voltage conditions and the adequacy of feeder cable sizing. The motor and feeder cable coordination calculation was reviewed to determine the adequacy of protection and coordination.

- RHR Pump Room Cooler (V-AC-4): The inspectors reviewed design calculations associated with room heat loads and cooling to ensure the pump and associated components remained within their temperature limits. Surveillance test results, including verification of adequate air and water flows were also reviewed to assess the capability of the room cooler to maintain temperature within prescribed limits. In addition, two operability evaluations concerning room cooler water flow test results and associated flow instrumentation were reviewed, including the recently installed high accuracy flow instrumentation.

- RHR Pump Miniflow Valve (CV-1994): Setpoint and scaling calculations were reviewed to ensure that the vendor requirements per NRC Bulletin 88-04 for minimum flow conditions were established. Maximum expected differential pressure calculations, coupled with valve actuator performance requirements were also reviewed to ensure the valve was capable of functioning under design conditions. The inspectors reviewed the electrical schematic diagrams and investigated the adequacy of voltage at the solenoids under design basis accident conditions. The inspectors verified the adequacy of the feeder circuits including the circuit breakers and its settings and cables. Finally, surveillance test results also were reviewed to assess potential component degradation and impact on design margins.

- RHR "B" Loop Discharge to Torus Isolation Valve (MO-2007): The inspectors reviewed motor-operated valve (MOV) calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, pressure locking, seismic Category I qualification of pipe support SR-852, and valve weak link analysis. The inspectors reviewed the electrical schematic diagrams and the degraded voltage calculations for both the power and control circuits of the MOV. The inspectors also verified the adequacy of the feeder circuits including the circuit breakers and its settings, the power cables, and the thermal overload relays. Diagnostic testing and IST surveillance results, including stroke time testing, were reviewed to verify acceptance criteria were met and performance degradation could be identified.

- RHR "B" Loop Low Pressure Coolant Injection Outboard Injection Valve (MO-2013): The inspectors reviewed MOV calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, seismic Category I qualification of piping and pipe supports, and valve weak link analysis. The inspectors reviewed the electrical schematic diagrams and the degraded voltage calculations for both the power and control circuits of the MOV. The inspectors also verified the adequacy of the feeder circuits including the circuit breakers and its settings, the power cables, and the thermal overload relays. Diagnostic testing and IST surveillance results, including stroke time and leak rate testing, were reviewed to verify acceptance criteria were met and performance degradation could be identified.

- RHR Containment Spray Outboard Isolation Valve (MO-2020): The inspectors reviewed MOV calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, seismic Category I qualification of pipe support SR-680, and valve weak link analysis. The inspectors reviewed the electrical schematic diagrams and the degraded voltage calculations for both the power and control circuits of the MOV. The inspectors also verified the adequacy of the feeder circuits including the circuit breakers and its settings, the power cables, and the thermal overload relays. Diagnostic testing and IST surveillance results, including stroke time and leak rate testing, were reviewed to verify acceptance criteria were met and performance degradation could be identified. In addition, a modification to provide operations the ability to throttle the valve was reviewed.

- Residual Heat Removal Service Water (RHRSW) Pumps (P-109A/C): The inspectors reviewed system hydraulic calculations including those addressing NPSH and vortex concerns related to the intake structure and associated required water levels. Further, calculations and the adequacy of the differential pressure setpoint across the RHR heat exchanger were reviewed to ensure the service water side was at a higher pressure than the RHR side. The inspectors reviewed pump installation and manufacturing details, vendor manuals, specifications, and pump curves to make sure that these parameters had been correctly translated into calculations, as required. Motor sizing and pump brake horsepower requirements and vendor ratings were reviewed for conformance with design basis load conditions. The inspectors also reviewed design calculations to determine the adequacy of voltage at motor terminals during degraded voltage conditions and the adequacy of feeder cable sizing. The motor and feeder cable coordination calculation was reviewed to determine the adequacy of protection and coordination. In addition, the inspectors reviewed completed pump surveillances to ensure that actual performance was acceptable and no significant degradation was taking place.

- RHRSW Flow Control Valve (CV-1729): The inspectors reviewed maximum expected differential pressures calculations, coupled with valve actuator performance requirements to ensure the valve was capable of functioning under design conditions. The calculation addressing the control feature of this valve, to ensure a positive differential pressure of service water to RHR was also reviewed. The inspectors reviewed the electrical schematic diagrams and investigated the adequacy of voltage at the solenoids under design basis accident conditions. The inspectors also verified the adequacy of the feeder circuits including the circuit breakers and its settings and cables. Surveillance and diagnostic test results were reviewed to ensure that actual performance was acceptable and no significant degradation was taking place.

- Reserve Auxiliary Transformer (1AR): The inspectors reviewed one-line diagrams and vendor test results for impedance data to confirm that correct transformer impedances were utilized in design analyses. The inspectors confirmed the adequacy of the overcurrent relay setting calculation for design basis loading and protective relay setting requirements. Surveillance testing for the overcurrent relays was reviewed for issues that affect reliability and for conformance with relay setting cards. The inspectors reviewed the modification that installed the transformer for potential impact on the design basis.

- 345-4.16kV Low Voltage Auxiliary Transformer (2R): The inspectors reviewed the design basis descriptions, equipment specifications, drawings, equipment nameplate data, voltage drop calculations, and short circuit and load flow calculations to evaluate the capability of transformer 2R to supply the voltage and current requirements to station safeguard loads. Protective relay trip setting calculations were reviewed to verify whether adequate protection coordination margins were provided. The relay settings review included the transformer overall differential currents and ground overcurrent relays. The inspectors reviewed the results of completed transformer preventive maintenance and relay calibrations to verify that the test results were satisfactory.

- 4.16 kV Switchgear (Bus 15): The inspectors reviewed load flow and short circuit current calculations to determine the design basis for maximum load, interrupting duty and bus bracing requirements, and the switchgear equipment vendor ratings for conformance with design basis. The coordination/protection calculation for the incoming line and feeder breaker relay settings was reviewed for design basis loading and protective relay setting requirements. The inspectors reviewed surveillance testing for the overcurrent relays for issues that affect reliability and for conformance with relay setting cards. The inspectors also reviewed selected breaker preventive maintenance for any recurring issues that affect reliability.

- 480V Load Center 103 (LC 103): The inspectors reviewed load flow and short circuit current calculations to determine the design basis for maximum load, interrupting duty and bus bracing requirements and the load center equipment vendor ratings for conformance with design basis. The coordination/protection calculation for the incoming line and feeder breaker relay settings was reviewed for design basis loading and protective relay setting requirements. The inspectors reviewed surveillance and preventive maintenance testing for the breakers for issues that affect reliability and for conformance with protective device trip requirements.

- 480V Motor Control Center (MCC 103A): The inspectors reviewed load flow and short circuit current calculations to determine the design basis for maximum load, interrupting duty and bus bracing requirements and the motor control center equipment vendor ratings for conformance with design basis. The coordination/protection calculation for the incoming line and feeder breaker relay settings was reviewed for design basis loading and protective relay setting requirements. The inspectors reviewed bus and motor starter surveillance and preventive maintenance testing for issues that affect reliability.

- <u>125Vdc Station Battery/Battery Charger (D11/D10)</u>: The inspectors reviewed calculations and analyses relating to battery sizing and capacity, hydrogen generation, station blackout (SBO) coping, and battery room transient temperature. The review was performed to ascertain the adequacy and appropriateness of design assumptions, and to verify that the battery was adequately sized to support the design basis required voltage requirements of the 125Vdc safety-related loads under both design basis accident and SBO conditions. The inspectors reviewed calculations relating to sizing and current limit setting to ascertain the adequacy and appropriateness of design assumptions, and to verify that the charger was adequately sized to support the design basis duty cycle requirements of the 125Vdc safety-related loads and the associated battery under both normal and design basis accident conditions. The inspectors also reviewed a sampling of completed surveillance tests, service tests, performance discharge tests, and modified performance tests. The review of various discharge tests was to verify that the battery capacity was adequate to support the design basis duty cycle requirements and to verify that the battery capacity meets TS requirements. In addition, the test procedures were reviewed to determine whether maintenance and testing activities for the battery charger were in accordance with vendor's recommendations. The inspectors also witnessed a weekly battery surveillance test. Seismic Category I qualification of the battery charger D10 anchorage was also reviewed.

- <u>125Vdc Distribution Panel (D11, D111)</u>: The inspectors reviewed 125Vdc short circuit calculations and verified that the interrupting ratings of the fuses and the molded-case circuit breakers were well above the calculated short circuit currents. The 125Vdc voltage calculations were reviewed to determine if adequate voltage would be available for the breaker open and close coils and spring charging motors. The inspectors reviewed the motor control logic diagrams and the 125Vdc voltage drop calculation to ensure adequate voltage would be available for the control circuit components under all design basis conditions. The inspectors also reviewed the 125Vdc short circuit and coordination calculations to assure coordination between the motor feed breaker open and close control circuit fuses and 125Vdc supply breakers and to verify the interrupting ratings of the control circuit fuses and the 125Vdc control power feed breaker.

b. <u>Findings</u>

(1) <u>Calculation Errors Associated With the Pneumatic Pressure Requirements for the Inboard Main Steam Isolation Valves (MSIVs)</u>

<u>Introduction</u>: A finding of very low safety-significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure to incorporate the actual physical configuration of the inboard MSIVs and the correct pneumatic system pressure drop into the pneumatic pressure requirement calculation for the inboard MSIVs.

<u>Description</u>: On September 2, 2009, the inspectors identified that the licensee failed to adjust the weight of the actuator moving parts for the inboard MSIVs to reflect that the actuator was offset by 45 degrees instead of being mounted vertically as assumed in calculation CA-94-037, "Calculation of Inboard MSIV Post-LOCA Closing Forces and

Pneumatic Pressure Requirements.” This error affected the calculated minimum alternate nitrogen system operating pressure required to close the inboard MSIVs.

The MSIVs were part of the primary containment with safety functions that included: (1) providing primary containment isolation; (2) preventing core damage by limiting the loss of coolant from the reactor vessel following a main steam line break outside the primary containment; and (3) preventing excessive release of radioactivity to the environs under assumed conditions of a primary steam line break outside of the primary containment. The inboard MSIVs were wye pattern globe valves. The main disc was attached to the lower end of the stem and moves in the valve guides at a 45 degree angle from the inlet pipe. The valves were provided with springs that maintain the valves in the closed position once they were shut by the valves’ air operator.

The normal pilot and safety grade pneumatic supplies for the inboard MSIVs was provided by the alternate nitrogen system. The manifold and system pressures of the alternate nitrogen system were monitored by pressure switches that give control room annunciation on low pressure. The nominal setpoint for the pressure switches was 91 psig. However, the pressure switch has an instrument uncertainty of 3.7 psig, such that the alarm potentially would not actuate until actual pressure was 87.3 psig.

The calculated minimum pneumatic pressure required to close the inboard MSIVs was 87 psig assuming that the valves were installed vertically. However, since the valves were at a 45 degree angle, the closing force provided by the weight of the actuator moving parts would be less than that assumed in the calculation. In addition, the alternate nitrogen system pressure drop was incorrectly determined because of a computational error. When the calculation incorporated the actual configuration of the valves and the corrected pressure drop, the minimum required pneumatic pressure increased to 88.84 psig, which was greater than the potential minimum pressure of 87.3 psig allowed by plant procedures.

The licensee evaluated the pneumatic pressure requirements for the inboard MSIVs and determined that the valves were operable by removing conservatisms in the affected calculation. The inspectors performed an independent review of the evaluation and had no further concerns. The licensee initiated action requests (ARs) 01196394 and 01198495 to revise the affected calculation.

Analysis: The inspectors determined that the failure to incorporate the actual physical configuration of the inboard MSIVs and the correct pneumatic system pressure drop into the pneumatic pressure requirement calculation of the inboard MSIVs was a performance deficiency.

The performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity cornerstone attribute of structures, systems, components and barrier performance, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the engineering calculation error was significant enough to require the calculation to be re-performed to assure that the minimum operating pressure of the alternate nitrogen supply was adequate to support the MSIVs pneumatic requirements.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 3b for the Barrier Integrity cornerstone. The finding screened as of very low safety-significance (Green) because it was a design deficiency of the physical integrity of the reactor containment that: (1) did not affect the barrier function of the control room against smoke or a toxic atmosphere; (2) did not represent an actual open pathway in the physical integrity of reactor containment; and (3) did not involve an actual reduction in function of hydrogen igniters in the reactor containment. Specifically, the last alternate nitrogen system surveillance demonstrated that the pressure was higher than the corrected minimum required pressure. In addition, the licensee showed that the calculation had sufficient conservatisms to account for the calculation errors. The inspectors performed an independent review of this evaluation and had no further concerns.

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 shall 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, as of September 4, 2009, the licensee failed to correctly translate applicable design basis into specifications. Specifically, calculation CA-94-037 failed to adjust the closing force provided by the weight of the actuator moving parts to reflect that the actuator was at a 45 degree angle and to correctly compute the pneumatic system pressure drop. Because this violation was of very low safety-significance and it was entered into the licensee's corrective action program (ARs 01196394 and 01198495), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000263/2009007-01).

(2) Emergency Service Water (ESW) Piping Supports Did Not Meet Seismic Category 1 Design Basis Requirements

Introduction: A finding of very low safety-significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified by the inspectors for the failure to restore the ESW piping supports to their design specifications.

Description: On September 9, 2009, the inspectors identified that the licensee failed to recognize that the ESW discharge pipe supports did not meet their seismic Category 1 design basis requirements and that, as a result, the issue was not properly evaluated such that corrective actions were not implemented.

During a plant walkdown, the inspectors noted that ESW piping stanchion supports SWH-42 and SWH-47 were not in contact with their baseplates. The supports were located on the discharge side of their respective ESW pumps near the boundary of safety and non-safety-related piping. The gaps between the supports and the floor baseplates were about ¼ inch. These supports were classified seismic Category 1 because their function was to protect the safety-related piping that provides the required cooling water to the EDGs. The EDGs were relied upon to safely shut down the reactor upon the loss of all outside power simultaneous with a design basis accident or event.

Upon further review, it was discovered that the licensee had previously identified this condition on at least three separate occasions.

- On February 3, 2003, AR 00633248 documented the discovery of gaps at supports SWH-42 and SWH-47. The piping was evaluated and found to be operable assuming that the supports were not functional. In addition, the licensee repaired the supports by removing the gaps to restore the piping and associated supports to within ASME Code allowables. The AR was classified Level B, which meant that the AR documented a condition that typically results in moderate impact to the plant.

- On April 28, 2008, ARs 01135806 and 01135808 documented the discovery of the gaps at the same supports. On this occasion, the licensee referred to the evaluation conducted in 2003 and determined that no corrective actions were necessary because the piping had been determined to be operable. Both ARs were classified Level D, which meant that the ARs were considered to be associated with a condition not adverse to quality.

- On February 26, 2009, AR 01170994 documented the rediscovery of the gaps and noted that the condition was previously evaluated in 2003 and 2008. No corrective actions were implemented. The AR was classified Level C, which meant that the AR documented a condition that typically results in minor impact to the plant.

It was also noted that in 2006, two check valves were removed and an air operated valve was replaced with a manual valve on each ESW lines under Engineering Change (EC) 768, “Service Water Check Valves Replacement,” resulting in a weight reduction on the pipe. This could have potentially resulted in the condition identified in 2008; however, this change was not evaluated as part of the 2008 or 2009 ARs.

The licensee initiated AR 01196345 to address the condition of the supports and AR 01196433 to document that the condition adverse to quality was not corrected when it was identified in 2008 and 2009. The piping was evaluated by the licensee and found to be operable assuming that the supports were not functional. The inspectors performed an independent review of this evaluation and had no further concerns. The licensee repaired the supports by installing shims to fill in the gaps through work requests 49141 and 49142.

Analysis: The inspectors determined that failure to restore the ESW piping supports to their design specifications was a performance deficiency.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of protection against external events and affected the cornerstone objective of ensuring the availability of the ESW system, and ultimately the EDGs, to respond to initiating events to prevent undesirable consequences. Specifically, the gaps between the ESW piping supports and the baseplates created a condition were the supports did not meet their seismic Category 1 design basis requirements. Furthermore, the finding impacted the availability of the EDGs because they cannot operate for a long period of time without cooling provided by ESW.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of findings,” Table 3b for the Mitigating System cornerstone. The finding screened as very low safety-significance (Green) because the finding was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee performed an operability evaluation that concluded the ESW piping remained operable assuming the affected supports were not functional. The inspectors performed an independent review of the operability evaluation and did not have further concerns.

This finding has a cross-cutting aspect in the area of problem identification and resolution, because the licensee did not thoroughly evaluate the issue such that the corrective actions were inadequate. Specifically, the licensee failed to implement corrective actions during the subsequent occasions that the condition was identified because the problem was not properly prioritized and evaluated. The ARs initiated subsequent to the condition being identified in 2003 were closed without corrective actions under the premise that the condition was determined to be acceptable in 2003. However, while the evaluation conducted in 2003 determined that the supports were operable, it also determined that the supports needed to be restored to the original design specifications. [P.1(c)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected.

Contrary to the above, in April 28, 2008 and February 26, 2009, the licensee failed to promptly correct a condition adverse to quality regarding the ESW piping supports. Specifically, although the licensee identified the gaps between the ESW piping supports and the baseplates, it failed to recognize that this condition did not meet the seismic Category 1 design basis requirements. Because the licensee failed to recognize the qualification requirement, the condition was evaluated incorrectly and determined to be acceptable such that corrective actions were not implemented. Because this violation was of very low safety-significance and it was entered into the licensee’s corrective action program (ARs 01196345 and 01196433), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000263/2009007-02).

(3) Failure to Adequately Evaluate Minimum Voltage Available at Safety-Related Electrical Components

Introduction: A finding of very low safety-significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the failure to adequately evaluate circuit loads in determining design limits in electrical calculations. Specifically, the inspectors identified three examples associated with the following: (1) the licensee’s failure to perform a calculation for safety-related motor starters that included all control circuit loads in determining the minimum voltage available at 120Vac starter coils, which was used to establish the coil voltage test acceptance criteria; (2) the licensee’s failure to include thermal overload (TOL) heater and starter contact resistance when calculating the minimum voltage at 480Vac motor terminals; and (3) the licensee’s failure to assure that the minimum voltage at the 120Vac solenoid operated control valves was in conformance with vendor requirements.

Description: The inspectors identified three examples where the licensee failed to adequately evaluate circuit loads in determining design limits in electrical calculations. These examples are as follows:

- Failure to Evaluate Circuit Loads in Determining the Minimum Voltage Available At Safety-Related Motor Starter Contactors

  The inspectors reviewed the licensee evaluation on IN 2007-09, "Equipment Operability Under Degraded Voltage Conditions," and found that the licensee had not adequately reviewed the notice for the issue on the adequacy of voltage at motor starter coils during degraded voltage conditions. The licensee initiated AR 01196447 to perform the required review. The inspectors reviewed calculations CA-92-250, "Project 92Q600 Cable Sizing Calculation," and CA-94-094, "MCC Starter Coil Pickup Voltages and Maximum Cable Lengths," for the motor starter coil voltage for ESW pumps P-111A and P-111B respectively. The inspectors found that calculation CA-94-094 determined 96.32 volts were available at the P-111B motor starter contactor coil and subsequently used 96 volts as the coil test acceptance criteria on drawing NE-36347-1A, "480V MCC Schedules." A starter contactor coil test was used in lieu of meeting the vendor criteria for minimum pickup voltage of 102 volts, which was 85 percent of rated voltage. However, the inspectors found that the calculation failed to consider the load on the control power transformer (CPT) due to timing relay burden, indicating light load, and contact resistance in determining the limiting voltage available at the motor starter.

  The licensee performed a hand calculation during the inspection, which determined that less than 96 volts was available when all the circuit resistance elements were included in the voltage analysis and established 94 volts as the new acceptance criteria for testing the coil. The contactor coil was previously tested in 1994 to pickup at 84 volts, which provided the basis for immediate operability on AR 01200723. As additional corrective actions, the licensee planned to test all the affected coils on drawing NE-36347-1A. On October 15, 2009, the test for the starter contactor coil for P-111B determined that the voltage needed for the coil to pickup was 104 volts, not the 94 volts required by the revised calculation. The licensee subsequently replaced the starter with a spare unit that was functionally tested at 80 volts. The licensee sent the failed starter to a test laboratory to determine the cause of failure and to initiate an evaluation of extent of condition.

  The test laboratory determined that the cause of the failure was mechanical binding of the starter right side contact arm and plunger binding due to abrasion and gouging, which then required a higher voltage for the coil to function. The test lab was not able to determine the root cause of the abrasion and binding that lead to the failure. Since this pump was operated periodically, the normal available voltage to the starter contactor coil was sufficient to overcome the binding effect of the contact arm such that the pump started as required. As part of the extent of condition for this issue, the licensee established a motor starter action plan to test 12 additional safety-related starters. In addition, motor starter testing methodology and preventive maintenance tasks were under review such that this issue could be identified prior to failure.

- Failure to Fully Evaluate Power Circuit Resistance in Determining Minimum Voltage Available at Safety-Related Motor Terminals

  The inspectors reviewed calculation CA-06-104, “480V MCC to Motor Terminal Voltage Drop,” and identified that the calculation did not consider TOL heater resistance or starter contact voltage drop in determining the voltage available at the motor terminals. For ESW Pump P-111A, this reduced the available voltage 0.2 volts, which reduced the voltage margin available to the minimum motor voltage (414 volts) by 12 percent. The inspectors found that the voltage available was approximately 0.45 percent more than the minimum voltage required by the motor based on the motor rating and the pump load requirement. The licensee initiated AR 01197431, which determined that although the available margin had been reduced, the operability of the pump was not affected by the calculation error.

- Failure to Assure Adequate Minimum Voltage at Safety-Related 120Vac Solenoid Operated Control Valves

  Section 8.10 of the USAR stated that the 120Vac instrument buses were maintained between 108Vac and 132Vac (120 ± 10 percent). Per MWI-3-M-2.01, “AC Electrical Load Study,” 120Vac rated loads, including the solenoid operated control valves, would have adequate voltage if the AC instrument buses were maintained within this voltage range. The vendor specified a voltage range of 102V - 132V for the solenoid operated control valves to operate properly. The published minimum operating voltage, as well as the minimum voltage used for equipment qualification testing for the solenoid valves (SV) was 102Vac. This value was identified in the “ASCO Nuclear Catalog-Nuclear Products Qualified to IEEE Specifications,” in “ASCO Solenoid Catalog No. 31,” and was used as the minimum test voltage for the ASCO Test Report AQS-21678/TR.

  The inspectors requested the voltage drop analysis for the SVs, however, the licensee was unable to locate an analysis for the SVs. The licensee performed an informal analysis for SV1729 and SV1994, which were in the inspectors’ scope, and as part of an extent of condition, SV1728, SV1995, and SV1996. Based on this analysis, the licensee determined that the voltages at SV1728 and SV1995 were below the vendor specified minimum rating. As a result of the negative margin identified for SV1728 and SV1995, the licensee tested three ASCO SVs, two of which were the same models as the existing solenoids, to determine their minimum pull-in and dropout voltages at various differential pressure values across the valve. These tests concluded that the SVs were capable of energizing at voltages lower than the specified 102Vac.

  The licensee entered this issue in their corrective action program as AR 01199936 to complete a formal voltage drop calculation for 120Vac supplies to safety-related equipment and to restore voltage margin as a long-term solution. The licensee documented the details of the testing for the sampled SVs in operability recommendation OPR 199936-1. Based on the results of the tests, the licensee determined that there was reasonable assurance that these valves would perform their safety functions as designed and therefore the OPR classified the solenoids as operable but nonconforming.

Analysis: The inspectors determined that the licensee's failure to ensure adequate voltage was available to energize the starter coils for 480Vac safety-related equipment and the 120Vac solenoid valve coils was a performance deficiency because the operability of safety-related equipment could not be assured and could have resulted in a loss of function during a design basis accident concurrent with a degraded voltage condition.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability and capability of safety-related equipment to respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure adequate voltage was available to energize safety-related starter coils to supply 480Vac power to motors and 120Vac to power SV coils would have affected the availability of their respective equipment to respond to initiating events.

The inspectors evaluated the finding, with the exception of ESW pump P-111B, using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," SDP Phase 1 screening. The finding screened as very low safety-significance (Green) because the finding was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the failure to assure adequate voltage was available at the starter coils and SV coils; and to perform periodic testing to assure the minimum voltage remained acceptable as the components aged did not result in an impact on current operability.

The inspectors evaluated the finding of ESW pump P-111B using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," SDP Phase 1 screening. In accordance with Table 3b, "SDP Phase 1 Screening Worksheet for Initiating Events, Mitigating Systems, and Barriers Cornerstones," the finding affected the safety of an operating reactor, specifically, the Mitigating Systems Cornerstone. In accordance with Table 4a, "Characterization Worksheet for Initiating Events, Mitigating Systems, and Barriers Cornerstones," the finding represented a potential loss of a safety function of a single train of emergency AC power for greater than its TS allowed outage time. The inspectors contacted a Region III Senior Reactor Analyst (SRA) to perform an SDP review of the finding.

In accordance with IMC 0609, Appendix A, the SRA determined that the Phase 2 SDP pre-solved tables showed that having one EDG service water pump inoperable for an entire year was a Yellow finding. The SRA determined this result to be conservative since P-111B was not completely failed, but shown to be nonfunctional only during certain degraded voltage conditions. Further, the voltage of interest at the P-111B starter coil was relevant when 12 EDG was supplying its associated safety-related 4160 Vac bus (Bus 16) and also during a loss of coolant accident (LOCA) with offsite power available (with EDG 12 operating but not supplying Bus 16) and degraded voltage conditions prior to a loss of off-site power LOOP occurring. The SRA performed an SDP Phase 3 analysis to further characterize the significance of the finding.

The SRA performed a Phase 3 analysis using the NRC Risk Assessment Standardization Project Handbook and the SPAR Model for Monticello, Revision 3P, Level 1, Change 3.45, dated August 2008. Assuming failure to start of P-111B for a one year exposure, the delta-CDF was less than 1E-7. The dominant cut-sets involved station blackout events with failure of emergency power sources and failure to recover either offsite or emergency power. The SRA contacted the licensee to review their risk analysis.

The licensee reviewed recent emergency core cooling system (ECCS) test records, which showed that P-111B started successfully under nearly identical voltage conditions to those that would be expected during a loss of offsite power transient with an ECCS initiation signal present. The licensee also performed a quantitative analysis assuming a failure to start probability for P-111B of ten times its nominal value. The resultant delta-CDF was about 3.6 E-7/yr. Nearly all of the increased risk was associated with internal flooding scenarios where the flood initiates at or propagates to the lower switchgear, which in turn causes a loss of offsite power event.

Given the low numerical risk results for the core damage sequences, the SRA concluded that the risk associated with this performance deficiency was very low safety-significance (Green).

Based on the results of the SDP Phase I and Phase III screenings, these three examples for this finding screened as very low safety-significance (Green). The inspectors determined there was no cross-cutting aspect associated with this finding because the three examples were legacy design issues and therefore was not reflective of current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that design control measures 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 suitable testing program.

Contrary to this requirement, prior to September 30, 2009, for P-111B and September 28, 2009, for the solenoid valve coils, the licensee's design control measures failed to verify the adequacy of minimum design voltage for the safety-related coils. Specifically, the licensee failed to adequately establish the degraded voltage value for safety-related starters and solenoid valve coils. In addition, periodic testing at degraded voltage, which was lower than the vendor's rated minimum voltage, was not performed to verify that the coils remained acceptable as the components aged. However, because this violation was of very low safety-significance and because the issue was entered into the licensee's corrective action program (AR 01200723, AR 01197431, and AR 01199936), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000263/2009007-03)

(4) Inadequate Testing for Motor Control Center (MCC) Contactors

Introduction: A finding of very low safety-significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," was identified by the inspectors for the failure to have adequate testing for safety-related equipment to monitor component degradation. Specifically, the licensee failed to verify that the MCC contactors would continue to pick-up under degraded voltage conditions with less than the vendors' required minimum voltage.

Description: The licensee's AC load study identified that the lowest voltage at the MCCs under degraded voltage conditions was 426V. In order for MOVs and other components to operate, the MCC contactors needed to be verified to pick-up at the MCC degraded voltage. Calculation CA-94-094, "MCC Starter Coil Pick-Up Voltages and Maximum Cable Lengths," determined the maximum cable length for each size of control cable to ensure reliable circuit operation of contactors under degraded voltage conditions. This calculation was performed to ascertain the maximum cable lengths with both 75 percent and 85 percent contactor pick-up voltage for various combinations of National Electrical Manufacturers Association (NEMA) starter sizes and CPT sizes at degraded voltage. The calculation determined there were 13 contactors that had less than 85 percent pick-up voltage, although only MO-2013 was included in the inspectors' scope. MO-2013 contactor was shown to have 84.52 percent pick-up voltage. The calculation, however, did not account for valves MO-2007 and MO-2020, which were also in the inspectors' scope.

The licensee performed an informal evaluation for MO-2007 and MO-2020 and determined that the cable lengths for both MOVs contactors were bounded by the maximum cable lengths determined for a NEMA size 1 starter with a 75 volt amp CPT. The inspectors reviewed this evaluation and determined that the assessment for these two MOVs was acceptable. This evaluation also identified that there were interposing relays installed in the MO-2013 control circuit to facilitate the pick-up of the contactor, which were not accounted for in the calculation CA-94-094.

The licensee provided calculation CA-92-223, "Analysis for Mod 92Q520 Control Circuit Voltage Drop," to address the interposing relay issue. This calculation determined that the starter coil for MO-2013 would have 83 percent of the rated voltage with the interposing relays in the circuit, which was less than the 84.52 percent identified in calculation CA-94-094. This calculation also stated a coil pick-up voltage acceptance criterion of 75 percent instead of 85 percent as indicated in calculation CA-94-094. The inspectors were concerned about the discrepant acceptance criteria used for the contactor pick-up voltage. In response, the licensee stated that during the evaluation of IN 94-050, "Failure of GE Contactors to Pull in at the Required Voltage," it was confirmed by General Electric (GE) that the contactor coils were rated for 85 percent and not the 75 percent pick-up voltage indicated in calculation CA-92-223. The licensee initiated AR 01200487 to supersede calculation CA-92-223 with calculation CA-94-094.

The inspectors expressed concern with the 13 GE contactors (associated with an MOV, fan, or pump) having less than 85 percent pick-up voltage and prompted the licensee for justifying operability under degraded voltage conditions. The licensee stated tests were conducted in 1976, 1977, 1994, and 2000, on a few contactor samples. The testing indicated that the contactors were able to pick up at less than 85 percent voltage. However, the licensee did not have a periodic testing program in place to ensure that age related degradation did not have an adverse impact on the contactors. The licensee provided Procedure 4847-PM, "GE 7700 Line Motor Control Center Maintenance Procedure," which indicated testing would only be performed on a starter when it was replaced. As such, the licensee was relying on the old test data to justify the continued operability of these contactors.

The licensee entered this issue into their corrective action program and initiated ARs 01200539 and 01200540 to revise the maintenance procedures to incorporate the requirements for periodic testing of contactors. In addition, the licensee initiated work orders to test the 13 contactors as soon as practicable. As of October 15, 2009, the licensee had tested 10 of the contactors, which verified for 9 out the 10 that the contactors would still pick-up at the calculated degraded voltage condition. However, as discussed in Section 1R21.3.b.(3) of this report, the starter motor contactor associated with ESW Pump P-111B failed during this testing. Additional corrective action and extent of condition for this issue was discussed in Section 1R21.3.b.(3) of this report.

Analysis: The inspectors determined that the failure to have an adequate testing program for the MCC contactors was a performance deficiency, because the failure of the contactors to pick up could have resulted in a loss of function during design basis accident conditions. Specifically, the licensee had determined from calculations that the voltage at the 13 contactors would be less than the vendor required 85 percent pick-up voltage and had not implemented any periodic testing to ensure the contactors would not degrade over time. Failure of the contactors to pick-up would prevent the associate MOV, fan, or pump to operate as required under a degraded voltage condition.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability and capability of safety-related equipment to respond to initiating events to prevent undesirable consequences. Specifically, the failure to verify that safety-related starter coils were not degrading over time thorough periodic testing such that they would continue to function at a lower calculated voltage. The failure of the starter coil to energize would prevent the associate MOV, fan, or pump to operate as required under a degraded voltage condition.

The inspectors determined the finding, with the exception of ESW pump P-111B, could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of findings,” Table 3b for the Mitigating System cornerstone. The finding screened as very low safety-significance (Green) because the finding was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee confirmed for 9 of the 13 contactors that they would function at a lower voltage than required by the vendor. The inspectors reviewed the results of the tests and verified that the tested contactors were able to pick up at less than 85 percent voltage. The inspectors determined that the schedule submitted by the licensee for testing the remaining contactors was reasonable. Based on these test results, no apparent common cause to the failed starter motor, and initial contactor testing performed in 1976, 1977, 1994, and 2000, there was reasonable assurance of operability for the remaining untested contactors.

The inspectors Phase III screening of the ESW pump P-111B finding was discussed in Section 1R21.3.b.(3) of this report, which concluded that the risk associated with this performance deficiency was very low safety-significance (Green).

Based on the results of the SDP Phase I and Phase III screenings, this finding screened as 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 XI, “Test Control,” requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design documents.

Contrary to the above, as of October 2, 2009, the licensee failed to establish a periodic testing program to ensure that the MCC contactors would be energized to perform its safety-function under design basis accident conditions. Specifically, the licensee failed to require periodic testing of contactors to ensure that age related degradation did not have any adverse impact on contactors' pick-up voltage. However, because this violation was of very low safety-significance and because the issue was entered into the licensee's corrective action program (AR 01200539 and 01200540), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000263/2009007-04)

(5) EDG Fuel Oil Supply System Design Does Not Meet Single Failure Criteria

Introduction: The inspectors identified an unresolved item (URI) regarding the EDG fuel oil supply system. Specifically, the diesel fuel oil supply system design does not meet the single failure criteria.

Description: The diesel fuel oil supply system consisted of two pumps, the fuel oil transfer pump and the fuel oil service pump. However, only the fuel oil transfer pump was considered safety-related. Nonetheless, TS Basis, Section 3.8.1, stated that both pumps must be operable in order for both EDGs to be operable.

The original Monticello Safety Evaluation Report (SER), dated March 20, 1970, stated that the EDGs were separate and independent with respect to fuel supplies. It also concluded that the onsite emergency electric power system was acceptable since no single failure should prevent power from being supplied to the engineering safety features from onsite sources. In addition, the original Safety Analysis Report stated that auxiliaries required to ensure continuous operation of the EDGs shall be supplied from the essential buses or control power transformers associated with the EDGs. However, the fuel oil service pump was powered by a non-essential motor control center that would be load shed on an essential bus transfer load shed signal. In fact, EDG 11 does not supply either pump without manual operator action. The fuel oil transfer pump was normally powered by EDG 12.

In addition, the USAR, Section 10.3.1.5, stated that the safe-shutdown analysis performed to comply with 10 CFR Part 50, Appendix R, Section G, revealed three fire areas that required repair of equipment or use of alternate fuel oil pumping methods and procedures for diesel oil pumping capability to the EDG day tanks to achieve cold shutdown.

Further, the TS Basis, Section 3.8.1, described the diesel fuel oil supply piping as redundant. However, the inspectors confirmed that the actual configuration was not redundant in that both pumps discharge into common piping.

The inspectors also questioned if the service pump was tested to supply an adequate flow. Specifically, TS Surveillance Requirement 3.8.1.5 verified, in part, that the fuel oil transfer system operates to transfer fuel oil from the T-44 storage tank to the day tanks. The TS Basis, Section 3.8.1, stated that this surveillance provides assurance that the required fuel oil transfer pumps are operable. Operable was defined in TS 1.1, which stated, in part, when the component is capable of performing its specified safety function. The TS Basis, Section 3.8.1, also stated that the fuel oil transfer pump and the fuel oil service pump are individually capable of maintaining the level in the day tank when both EDGs are operating at full load. However, the inspectors noted that the licensee only ensured that the fuel oil service pump transferred fuel and maintained level in just one EDG day tank.

This issue is unresolved pending further NRC review of the licensing basis for the diesel fuel oil transfer system and determination of NRC courses of action for resolution of the issues. (URI 05000263/2009007-05).

(6) Inadequate Tornado Missile Protection for the EDG System Components

Introduction: The inspectors identified an unresolved item (URI) regarding the design and licensing basis for the standby diesel generator (EDG) building ventilation system and whether the ventilating system had to be protected from the effects of a design basis tornado.

Description: During a walkdown of the EDG building, the inspectors noted that temperature control dampers in the ventilation system were mounted flush with the outside walls of the building with only a metal grating serving as a barrier for tornado missiles. The inspectors questioned the adequacy of the design and received the licensee position that the EDG building was a Class 1 building designed to protect the EDG and the fuel oil day tank from tornado missiles. However, the licensee stated that the temperature control dampers and ventilation fans (V-SF-9 and - 10) were not designed to be protected from a tornado missile. The licensee stated that the designer and the licensee intended on protecting Class 1 equipment required to assure safe-shutdown of the reactor from a missile event based on the credibility of the missile. The licensee agreed that the ventilation fans were necessary for the EDGs to be operable and that the EDGs could only run for a short period of time without the fans running. However, the licensee maintained that the EDGs could perform their safe-shutdown function without the ventilation fans. The inspectors disagreed as the EDGs could not run long enough to shut the reactor down, remove latent and decay heat, and maintain the reactor in a cold shutdown condition following the loss of the EDG ventilation systems. The licensee maintained that a single missile could not take out both fans as a full height reinforced concrete wall separated the two ventilation systems.

The inspectors reviewed the original design submittal (NSP-1 dated October 17, 1969), and found that the Class 1 equipment included the "Standby Diesel Generator System," and the "Emergency Buses and other electrical gear to and including power equipment required for safe-shutdown." The ventilation systems were not included in the list of Class 1 equipment or in the list of Class 2 equipment. However, the licensee believed the ventilation systems were considered Class 2 based on the last entry in the Class 2 list: "All Other Piping and Equipment not listed under Class 1." After further research, the licensee stated that there was no specific statement regarding the EDG building ventilation in any licensing document.

The inspectors identified the following statements in various licensing documents.

- Section 2.2.4, "Standby Diesel Generating Building," of the USAR stated, "The principal function of this building is to provide a safe enclosure and protection for the standby diesel generators and portions of the power distribution systems enclosed therein."

- Section 2.2.4.1, "Structure Description," stated, in part, "A north-south [sic] interior wall of reinforced concrete extends the full height of the structure providing physical separation of the diesel generator systems."

- The original Monticello SER, Section 3.1.2, "Meteorology," stated, in part, that "the facility structures and systems, which are necessary for a safe-shutdown of the reactor are designed to withstand the effects of wind loadings and potential missiles resulting from a tornado."

- In the current Revision 25 of the USAR the inspectors noted the following:

  a) Section 8.4.1.1, "Design Basis," stated that two independent EDGs provide redundant standby power sources.

  b) Section 8.4.1.1.b stated, "The EDG 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 voltage and frequency until either manually or automatically stopped."

  c) Section 8.4.1.1.d stated, "The EDGs shall be located in Class 1 structures."

  d) Section I.4.3.14, "HVAC Systems," stated: "The only HVAC Equipment required for safe-shutdown are the ECCS Room Coolers, V-AC-5 (Division I) and V-AC-4 (Division II), located in respective Reactor Building corner rooms on the 920-foot elevation, and the EDG supply fan, V-SF-9 for EDG No. 12, and V-SF-10 for EDG No.11."

  e) USAR Table J.4.5-1, "Appendix R Safe-shutdown Equipment List," identified fans V-SF-9 and -10 as safe-shutdown equipment.

The inspectors concluded that the EDG ventilation fans were an auxiliary necessary for the EDG system and that the term EDGs in the USAR included all of the auxiliaries for the EDGs; therefore, the ventilation fans were necessary for safe-shutdown of the reactor (to achieve and maintain cold shutdown).

Based on no actual licensing document specifically mentioning the EDG ventilation, the necessity of the fans for EDG operability, the USAR references to the fans being safe-shutdown equipment, and the NSP-1 statement that the full-height wall separated the EDG systems, the inspectors concluded that the fans and temperature control dampers should be considered Class 1 equipment and should have been protected from tornado missiles, as well as the effects of tornadoes on the ventilation ducts as described in RIS 2006-023, "Post Tornado Operability of Ventilating and Air Conditioning Systems Housed in Emergency Diesel Generator Rooms." However, because this issue was not clearly defined in the original licensing documents, this will be an unresolved item pending consultation with NRC headquarters for clarification of whether the EDG building ventilation was or was not required to be a Class 1 system.
(URI 05000263/2009007-006)

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed six operating experience issues to ensure that NRC generic concerns had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection:

- IN 2006-22, "New Ultra-Low-Sulfur Diesel Fuel Oil Could Adversely Impact Diesel Engine Performance;"
- IN 2007-09, "Equipment Operability Under Degraded Voltage Conditions;"
- IN 2008-02, "Findings Identified During Component Design Bases Inspections;"
- IN 2009-02, "Biodiesel in Fuel Oil Could Adversely Impact Diesel Engine Performance;"
- RIS 2006-023, "Post Tornado Operability of Ventilating and Air Conditioning Systems Housed in Emergency Diesel Generator Rooms;" and
- SC06-01, "Worst Single Failure for Suppression Pool Temperature Analysis."

b. Findings

No findings of significance were identified.

.5 Modifications

a. Inspection Scope

The inspectors reviewed three permanent plant modifications related to selected risk-significant components to verify that the design bases, licensing bases, and performance capability of the components had not been degraded through modifications. The modifications listed below were reviewed as part of this inspection effort:

- EC 12361, "EPU-Provide Operations the Ability to Throttle MO-2020 and MO-2021";
- EC-11734, "1AR Transformer Replacement, Monticello Plant EPU Project"; and
- SRI-91-010, "USAR Update Concerning Diesel Oil Day Tank."

b. Findings

No findings of significance were identified.

.6 Risk-Significant Operator Actions

a. Inspection Scope

The inspectors performed a margin assessment and detailed review of four risk-significant, time critical operator actions. These actions were selected from the licensee's PRA rankings of human action importance based on risk achievement worth values. Where possible, margins were determined by the review of the assumed design basis and USAR response times and performance times documented by job performance measures results. For the selected operator actions, the inspectors performed a detailed review and walk through of associated procedures, including observing some actions in the plant with an appropriate plant operator to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required.

The following operator actions were reviewed:

- Loss of Instrument Air;
- Operating Suppression Pool Cooling;
- Align Diesel Fire Pump in a Station Blackout; and
- Time Critical Manual Actions Outside the Main Control Room Before Core Damage.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

The inspectors reviewed a sample of the selected component problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action program. 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.

4OA5 Power Uprate (71004)

.1 Plant Modifications (2 samples)

a. Inspection Scope

The inspectors reviewed plant modifications for those implemented for the extended power uprate. This includes seismic qualification of balance of plant piping and pipe supports for extended power uprate and the control logic changes to allow operations the ability to manually set an incremental position for the safety-related outboard containment spray valves.

- EC 12361, "EPU-Provide Operations the Ability to Throttle MO-2020 and MO-2021"; and
- EC 11126, "EPU-MOD 11-Balance of Plant Piping Support Modifications."

b. Findings

(1) Pipe Support Design Deficiencies

Introduction: A finding of very low safety-significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure to demonstrate pipe supports met their design requirements. Specifically, the inspectors identified that the calculation for pipe support SR-526 failed to use the minimum yield strength in determination of the allowable bending stress of the pipe support SR-526 baseplate as required in the American Institute of Steel Construction (AISC) code. In addition, the inspectors identified that the calculation for pipe support PS-16 failed to use the design basis concrete compressive strength in determination of the anchor bolt allowable as required in the licensee's design specification.

Description: The inspectors identified two examples where pipe supports would not meet their design requirements. These examples are as follows:

- Pipe Support SR-526

  The design function of pipe support SR-526 was to hold and maintain SW6-24"-JF discharge line in position during a seismic Category I design basis event to meet internal flooding requirements. The failure of this pipe support may create a pipe break in the SW6-24"-JF piping line, which could cause internal flooding and subsequent impact on the RHR pumps P-202A/C and core spray pump P-208A. Also, the failure of pipe support SR-526 due to a seismic Category I design basis event could impact the RHR heat exchanger E-200A. The inspectors identified a non-conservative technical error in pipe support SR-526 calculation CA-04-112, "Evaluation of Support SR-526 for Supplementary Load Combination." The calculation evaluated acceptability of the baseplate connection based on an allowable bending stress of 0.75 times the actual baseplate material yield stress. The acceptance criterion for allowable bending stress of a pipe support baseplate was established in Specification No. MPS-1100, "Specification for the Analysis of Piping and Pipe Support System," Revision 8. Section 8.2.2 of MPS-1100 required the seismic Category I pipe support stress limits to be in accordance with the AISC. The requirement in the AISC was that the allowable bending stress was 0.75 times the specified minimum yield stress of the material. The licensee determined that the calculation CA-04-112 specified the actual material yield stress instead of the minimum yield stress. The use of actual material yield stress for the evaluation of the baseplate did not meet AISC code requirements.

The licensee agreed that the use of the Certified Material Test Report or the actual material yield stress to evaluate the pipe support baseplate was outside the allowable bending stress acceptance criteria as described in AISC. This issue was entered in the licensee's corrective action program as ARs 01198415 and 01199540. The licensee performed an analysis and determined the pipe support was operable but nonconforming.

- <u>Pipe Support PS-16</u>

  The design basis function of pipe support PS-16 was to hold and maintain PS4-18"-ED main steam line in position during a seismic Category I design basis event. The outboard MSIV AO-2-86D was located between the primary containment penetration X-7D and pipe support PS-16. The inspectors were concerned that, if the pipe support did not meet its seismic Category 1 design basis requirements, the line could be damaged during a design basis event at a location between the primary containment penetration and the MSIV. This condition could adversely affect the capability of the outboard MSIV to perform its safety-related function of providing primary containment isolation.

  The inspectors identified a non-conservative technical error in pipe support PS-16 calculation CA-96-147, "Evaluation of PS1-18"ED, PS2-18"ED, PS3-18"ED and PS4-18"ED Outside Containment for Revised Turbine Stop Valve Closure Loads." The calculation evaluated the acceptability of the anchor bolts based on allowables that use age hardened concrete compressive strength of 6000 pounds per square inch. The acceptance criteria for anchor bolt allowables were determined from Specification No. MPS-1100. Section 8.2.7 of MPS-1100 stated "the concrete design strength for determining allowable loads for the anchor bolts shall be 3000 pounds per square inch or 4000 pounds per square inch." The licensee determined that calculation CA-96-147 specified that use of anchor bolt allowables based on age hardened concrete compressive strength of 6000 pounds per square inch instead of 3000 pounds per square inch or 4000 pounds per square inch as required in Specification No. MPS-1100.

  The licensee agreed that the use of age hardened concrete compressive strength to evaluate pipe support PS-16 anchor bolts was outside the anchor bolt acceptance criteria established in Specification No. MPS-1100. This issue was entered in the licensee's corrective action program AR 01200215. The licensee performed an analysis and determined the pipe support was operable but nonconforming.

<u>Analysis</u>: The inspectors determined that the failure to use minimum specified yield stress to analyze the applied bending stress on the baseplate for pipe support SR-526 as described in the AISC code and failure to use the design basis concrete compressive

strength in determination of the anchor bolt allowable for pipe support PS-16 as required in MPS-1100 was a performance deficiency.

The performance deficiency for the pipe support SR-526 example was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the RHR and core spray pumps. Specifically, compliance with seismic Category I design basis requirements (AISC code requirements) was to ensure the pipe support SR-526 would function as required during a seismic Category I design basis event and not cause internal flooding and subsequent impact on the RHR pumps P-202A/C, and core spray pump P-208A.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Mitigating Systems cornerstone. The finding screened as of very low safety-significance (Green) because it was a design deficiency that did not represent an actual loss of safety function. The inspectors agreed with the licensee's position that the pipe support SR-526 was operable but nonconforming.

The performance deficiency for the pipe support PS-16 example was determined to be more than minor because it was associated with the Barrier Integrity cornerstone attribute of design control and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, compliance with seismic Category I design basis requirements was to ensure the pipe support PS-16 would function as required during a seismic Category I design basis event and not adversely affect the outboard MSIV AO-2-86D.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 3b for the Barrier Integrity cornerstone. The finding screened as of very low safety-significance (Green) because it was a design deficiency of the physical integrity of the reactor containment that: (1) did not affect the barrier function of the control room against smoke or a toxic atmosphere; (2) did not represent an actual open pathway in the physical integrity of reactor containment; and (3) did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors agreed with the licensee's position that the pipe support PS-16 was operable but nonconforming.

Based on the results of the two SDP Phase I screenings, this finding screened as very low safety-significance (Green). The inspectors determined there was no cross-cutting aspect associated with this finding because the two examples were legacy design issues, therefore was not reflective of current performance.

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 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, on September 18, 2009 and September 29, 2009, the licensee failed to demonstrate design adequacy of pipe support SR-526 and pipe support PS-16 was consistent with seismic Category I requirements. Specifically the performance of design reviews for pipe support SR-526 and pipe support PS-16 were inadequate, in that design calculation CA-04-112 did not demonstrate that the pipe support SR-526 baseplate would meet AISC code requirements and the design calculation CA-96-147 did not demonstrate the pipe support PS-16 would meet design basis anchor bolt requirements. Because this violation was of very low safety-significance and it was entered into the licensee's corrective action program (ARs 01198415, 01199540 and 01200215), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000263/2009007-07)

4OA6 Meeting(s)

.1 Exit Meeting Summary

On October 2, 2009, the inspectors presented the inspection results to Mr. T. O'Connor, and other members of the licensee staff. On December 4, 2009, the inspectors conducted a re-exit of the inspection results to Mr. T. O'Connor, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Several documents reviewed by the inspectors were considered proprietary information and were either returned to the licensee or handled in accordance with NRC policy on proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
T. O'Connor, Site Vice President
J. Grubb, Plant Manager
P. Albares, Operations Shift Manager
R. Anderson, System Engineering Supervisor
R. Baumer, Regulatory Affairs Compliance Engineer
V. Bhardwaj, System Engineering Manager
N. Haskell, Engineering Director
B. Halvorson, Configuration Management Supervisor
K. Jepson, Business Support Manager
S. Oswald, Regulatory Affairs Analyst
D. Pennington, Mechanical Design Engineer
G. Salamon, Corporate Regulatory Affairs Manager
R. Siepel, Electrical Design Engineer
E. Watzel, Electrical Design Engineering Supervisor
P. Young, Regulatory Programs Supervisor

Nuclear Regulatory Commission
S. Thomas, Senior Resident Inspector, DRP
L. Haeg, Resident Inspector, DRP

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000263/2009007-01	NCV	Calculation Errors Associated With the Pneumatic Pressure Requirements for the Inboard MSIVs
05000263/2009007-02	NCV	ESW Piping Supports Did Not Meet Seismic Category 1 Design Basis Requirements
05000263/2009007-03	NCV	Failure to Adequately Evaluate Minimum Voltage Available at Safety-Related Electrical Components
05000263/2009007-04	NCV	Inadequate Testing for MCC Contactors
05000263/2009007-05	URI	EDG Fuel Oil Supply System Design does Not Meet the Single Failure Criteria
05000263/2009007-06	URI	Inadequate Tornado Missile Protection for the EDG System Components
05000263/2009007-07	NCV	Pipe Support Design Deficiencies

Closed

05000263/2009007-01	NCV	Calculation Errors Associated With the Pneumatic Pressure Requirements for the Inboard MSIVs
05000263/2009007-02	NCV	ESW Piping Supports Did Not Meet Seismic Category 1 Design Basis Requirements
05000263/2009007-03	NCV	Failure to Adequately Evaluate Minimum Voltage Available at Safety-Related Electrical Components
05000263/2009007-04	NCV	Inadequate Testing for MCC Contactors
05000263/2009007-07	NCV	Pipe Support Design Deficiencies

Discussion

None

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.

CALCULATIONS

Number	Description or Title	Revision
CA-00-104	Intake Structure Minimum Water Level	0
CA-01-036	Inservice Testing (IST) Pump and Valve Acceptance Criteria Rounding Evaluation for 4lh Ten-Year Code Interval	28
CA-01-062	Instrument Uncertainty Calculation	1
CA-01-113	Setpoint for RHR Minimum Flow Switches FS-10-121A, B, C, and D	13
CA-01-174	Minimum RHRSW Pump Differential Pressure	2
CA-01-177	Determination of Containment Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps Updated for Suction Strainer Debris Loading	8
CA-02-179	125 Volt Div 1 Battery Calculation	1
CA-03-111	EDG Jacket Cooler Maximum Allowed Tubes Plugged	10
CA-03-145	Evaluation of SR-680	1
CA-03-146	Evaluation of New Guides on The High Point Vent	1
CA-03-200	Internal Flooding Evaluation Due To A Postulated Break In 2.5” Fire Line	11
CA-04-010	EOP Calculation Inputs	2
CA-04-103	Emergency Diesel Generator Room Heat-up Air-Cooler Fan-Flow Analysis	12
CA-04-112	Evaluation of Support SR-526 for Supplementary Load Combination	0
CA-04-113	Evaluation of Support SR-531 for Supplementary Load Combination	0
CA-04-114	Evaluation of Support SWH-48A for Supplementary Load Combination	0
CA-04-127	Determination of Maximum Allowable Outside Air Temperature	13
CA-04-133	AOV Component Calculation, AO-2-86A, B, C, D	15
CA-04-145	EDG Ventilation: Cooling Load and Air Flow Determination	12
CA-04-146	Design of Supports for HVAC Ductwork in DGB (Diesel Generator Building)	1
CA-06-054	Evaluation of SRP Module J.3 due to Replacement of MO-2009	1
CA-06-081	Evaluation of New Support SR-852 Next to TWH-87	0
CA-06-104	480V MCC to Motor Terminal Voltage Drop	1
CA-07-047	Residual Heat Removal System MOV Performance Analysis	0
CA-08-047	D10 Battery Charger Mounting Evaluation	0

CALCULATIONS

Number	**Description or Title**	**Revision**
CA-08-082	Evaluation of SR-537	0
CA-08-083	Evaluation of SWH-150	0
CA-08-157	Combined AC Model Database	0
CA-09-120	Evaluation of SRP Module F.2 for 120% EPU	0
CA-09-148	Cycle 25 Monticello Specific EOP Calculations, Appendix B.15	0
CA-09-177	Sizing of the Orifice in the SW lines to the RHR Pump Motor Coolers	0
CA-88-003	RHR Pump Minimum Flow Calculation for IE Bulletin #88-04 Response	0
CA-90-018	Determination of Acceptance Criteria for RHR Pump Surveillance Testing and Verification of Adequate LPCI Flow Under Four and Two Pump Operation for SRI No. 90-002 and GE Report NEDC-31786P, Respectively	01/09/91
CA-90-023	Minimum Allowable Fuel Oil Storage Tank Level	2
CA-91-001	125V DC Fault Current	1
CA-91-006	125V DC Battery Charger Sizing	1
CA-91-012	125V DC SBO Load Profile Study	2
CA-91-063	Determination of Uncertainty in RHR Pump D/P Measurement During Performance of Test 0255-04-111	2
CA-91-086	AC Load Study 1AR No.10 XFMR, LOCA Load 2 Core Spray Pumps Starting	2
CA-91-092	Plant Fault Study, 2R XFMR, 2RS Reactor In-Line	7
CA-91-093	Plant Fault Study, 1AR Transformer Feed From #10 Transformer	4
CA-91-098	AC Load Study Short Circuit Contribution Database	7
CA-91-100	AC Load Study Transformer Database	7
CA-92-024	125 V DC Battery Sizing Calculation	5
CA-92-036	RHR Pump Room Temperature Analysis	7
CA-92-038	Determination of RHRSW Instrumentation Inaccuracies	4
CA-92-214	RHR System Motor Operated Valve Functional Analysis	11
CA-92-223	Analysis for Mod. 92Q520 Control Circuit Voltage Drop	0
CA-92-224	Emergency Diesel Generator Loading	5
CA-92-250	Project 92Q600 Cable Sizing Calculation	0
CA-92-295	Protection Settings for New LC-103 480V Switchgear Lineup	1
CA-92-299	Stem Thrust Assessment of 16" A/D Globe Valves: MO-2012 and MO-2013	1
CA-93-023	Instrument/Control Setpoint Calculation: Main Air Supply To Outboard MSIV	1
CA-93-066	AC Loads Study, Degraded Voltage Setpoint, 1R XFMR, LOCA Load	6
CA-93-084	Hydrogen Generation of No. 11 and No. 12 Battery Rooms	1

CALCULATIONS

Number	Description or Title	Revision
CA-94-017	Calculation of Alternate Nitrogen System Operability Leakage Criteria	6
CA-94-037	Calculation of Inboard MSIV Post-LOCA Closing Forces and Pneumatic Pressure Requirements	5
CA-94-084	Determination of MSIV Stroke Time Acceptance and Setpoint Bands	7
CA-94-094	MCC Starter Coil Pickup Voltages and Maximum Cable Lengths	0
CA-94-142	Stem Thrust Assessment of 10” A/D Gate Valves: MO-2020, MO-2021, MO-2022, and MO-2023	2
CA-95-076	Calculation for ASCO Pressure Switch Set Points	0
CA-95-117	Stem Thrust Assessment of 12” A/D Gate Valves: MO-2006 and MO-2007	0
CA-96-012	Minimum Thrust Required to Close Inboard MSIVs and to Obtain Leak Tight Seat	2
CA-96-015	MO-2006 and MO-2007 Pressure Locking Interim Calc.	0
CA-96-113	Temperature of RHR Rooms During DBA LOCA	0
CA-97-090	AC Voltage Study, 480V Voltage Determination During Diesel LOOP/LOCA ECCS CS Start Test Conditions	2
CA-97-157	RHR Room Temperature Response to LOCA	2
CA-98-008	Environmental Qualification (50.49) of Automatic Valve Company Air Control Assembly	5
CA-98-032	Environmental Qualification (50.49) of Namco EA740/EA180 Series Limit Switches	11
CA-98-033	Environmental Qualification (50.49) of Namco EC210 Quick Disconnects	15
H20-M-001	Cooling Coil Performance at Minimum Water Flow Rate	2

CORRECTIVE ACTION PROGRAM DOCUMENTS

Number	Description or Title	Date
00633248	Supports SWH-42,47 Stanchion do Not Touch Baseplate	02/23/03
00839220	EDG Fuel Oil Transfer System is Vulnerable to Operator Error	06/01/05
01002620	Packing has Slight Leak on MO-2007	11/02/05
01002810	Potential NRC Violation for Sprinkler Obstruction in EDG room	11/03/05
01046881	MO-2007 12 RHR Discharge to Torus has an Oil Leak	08/28/06
01051775	New Fuel Shipment is Ultra-Low-Sulfur-Diesel	09/22/06
01051793	CST Line Above Torus has Wood Wedge Between Wall and Pipe	09/22/06
01056182	Motor Terminal Voltages Could Drop Below 90% Rated	10/17/06
01061972	Diesel Fuel Oil Pump Disch Pressure Shows Lowering Trend	11/15/06
01103509	Station Transformer Annual Insulating Oil Dielectric Test	07/25/07
01119969	Inconsistencies Between B Manual and TS Bases	12/04/07

CORRECTIVE ACTION PROGRAM DOCUMENTS

Number	Description or Title	Date
01130761	CV-1729 Not Controlling at 7000 gpm	03/12/08
01135806	Support SWH-42 Stanchion Does Not Touch Baseplate	04/28/08
01135808	Support SWH-47 Stanchion Does Not Touch Baseplate	04/28/08
01146078	High Voltage Spurious Trips-New D10 Charger Bench Test	07/31/08
01149737	Potential Safety Hazard from MO-2013 Insulation Lagging	09/05/08
01151194	Packing Leak on MO-2007, RHR “B” Outboard Torus Isolation	09/19/08
01152938	11 and 12 EDG-ESW IST Pump Trends	09/30/08
01155238	Station Evaluation Of Industry And Internal OE	10/14/08
01160133	Implementation of Time Critical Operator Actions Ctrl Process	11/21/08
01164976	Unplanned TS Action Entry due to RCIC Inoperability	01/10/09
01170994	Supports SWH-42,47, Inadequate Documentation of Issue	02/26/09
01171900	EC 12361 MO-2020 (DW Spray)-ECN Error	03/05/09
01173431	AO-2-86D Has Excessive Seat Leakage	03/18/09
01175465	OSP-DOL-0543, Rev 2	03/28/09
01186755	NRC Question Re Core Spray Flow Throttling for ASDS	06/24/09
01187408	Documentation of 1AR Transformer Operability Questions	06/29/09
01190108	Station Transformer Annual Insulating Oil Test results	07/20/09
01193840	Error/Wrong Input in Calc 06-104, 480V Motor Term Voltage	08/17/09
01194290	Error in Voltage Drop Calculation CA-06-104	08/24/09
01195281	Cracks in Concrete Pad of 1AR Transformer	08/26/09
01195865	Minimum Voltage for 2R to 1R Auto Transfer	08/31/09

CORRECTIVE ACTION PROGRAM DOCUMENTS GENERATED AS A RESULT OF THE INSPECTION

Number	Description or Title	Date
01193742	Panel D21 Is Missing Two Bolts	08/15/09
01196091	Minor References List Error in EPU Calc. 08-082	09/01/09
01196110	Bus Bracing Ratings Not Documented 4.16 kV Switchgear and 480V Load Centers	09/02/09
01196224	EPU Calcs Not Listed on ADLs and Stored As Required	09/02/09
01196345	ESW Supports SWH-42 and SWH-47 are Not Functional	09/03/09
01196371	CT Ratio Discrepancy Between Relay Cards and Drawing	09/11/09
01196394	CA-94-037 Calculation Error	09/03/09
01196433	Identified Condition Adverse to Quality Not Corrected	09/03/09
01196448	Discrepancy Between Drawings of MCC 142 for B4208	09/04/09
01196451	EDG Base Tank Volume Calculation CA 90-023	09/03/09
01196495	EDG Fuel Oil XRF Pump Not Included in the EDG Loading	09/04/09
01196513	Stanchion for PI-1982 Only has 4 Bolts Installed	09/04/09
01196531	NE-93194-11 and NE-36438-9 Show Incorrect HP for P-77/MTR	09/04/09

CORRECTIVE ACTION PROGRAM DOCUMENTS GENERATED AS A RESULT OF THE INSPECTION

Number	**Description or Title**	**Date**
01196811	Apparent Deficiency in Documentation of EDG Air Start	09/10/09
01196861	CA-00-104, Intake Structure Min Water Level Is Incorrect	09/08/09
01197333	No Documentation for Fuse in D11-22 (Computer Input)	09/17/09
01197339	Inconsistency in Fuse Sizing Div 1 125 VDC D111-04 Circuit	09/11/09
01197431	Undocumented Assumption in Calculation 06-104	09/12/09
01197447	Evaluation of IN 2007-09 Did Not Address Issue	09/03/09
01197792	CA-91-100 2R Impedances Are For 50 MVA, Not 25 MVA	09/15/09
01197923	Mod Did Not Assess Impact to T-44 Tank	09/15/09
01197979	Typo in Ops Man B-09.06-02 for 2R Transformer	09/16/09
01198021	Errors on MCC Schedule Drawings	09/16/09
01198033	CA-07-047 Not Updated with 2003 Test Data Correctly	10/16/09
01198046	D111 Panel Lower Right Side Wing Panel Has Gap	09/16/09
01198056	Incorrect Pressure Term Used in CA-92-214	10/16/09
01198090	Revise CA-92-224 to Remove 2750 kW and 3050 kW Ratings	09/16/09
01198227	Update CA-03-111 to Incorporate Enhancement	09/17/09
01198266	Weak Link Labeled Incorrectly in MOV Database	09/17/09
01198415	Service Water Line Support May Not Meet Code Requirements	09/18/09
01198495	CA-94-037 Has Error	09/18/09
01198855	50.59 Did Not Evaluate Use of Age Hardened Concrete	09/21/09
01199213	Ref. for EPU CA-08-081 Documented Incorrectly	09/23/09
01199540	Calc Accept Criteria for Suprt May Not Meet Code Reqmts	09/25/09
01199542	New Crack Not Identified in PUSAR	09/25/09
01199606	Error in 50.59 Screening for EPU EC 11126	09/25/09
01199936	Voltage Drop Evaluation for RHR/RHRSW ASCO Solenoid Vlvs	09/28/09
01199999	Minor Error in Line Designation Table Drwg ND-57664-1	09/28/09
01200120	Procedure Question Operation of RHRSW at 6000 gpm	09/29/09
01200170	Pickup Voltage on 480V MCC Contactors Not Periodically Tested	09/29/09
01200215	Incorrect Input was Used in Calculation	09/29/09
01200265	Incorrect Concrete Strength Chosen for 2 EPU Calcs	09/29/09
01200397	Information Lacking on 09-120 to Assess Suitability	09/30/09
01200432	Incorrect Input Used in Calculation No. 09-122	09/30/09
01200487	Apparent Conflict Between Calculations	09/30/09
01200539	Revise Procedure 4847-PM (GE 7700 Line MCC) to Require Contactor Pickup Voltage Test	10/01/09
01200540	Revise Procedure 4027-PM (Klockner-Moeller MCC B34 and B44) to Require Contactor Pickup Voltage Test	10/01/09

CORRECTIVE ACTION PROGRAM DOCUMENTS GENERATED AS A RESULT OF THE INSPECTION

Number	Description or Title	Date
01200546	Relays 10A-K70A and10A-K70-B are Agastat E7012AD004	10/01/09
01200682	Support SR-526 is in Contact with Insulation on RHR HX	10/01/09
01200723	Voltage at P-111B Contactor is Less Than 96 Volts	10/02/09
01200725	Instr. Uncertainty Assumed in OPR 01076631 Not Bounding	10/02/09
01202633	B4319 Contactor Failed Degraded Voltage Testing	10/15/09

DRAWINGS

Number	Description or Title	Revision
DNL32274	1AR Transformer Nameplate	E
FSB-0503-1	Fuse Breaker Study	0
FSB-0504-1	Fuse Breaker Study	0
FSB-0507-1	Fuse Breaker Study	1
FSB-0508-1	Fuse Breaker Study	0
M-112	PandID RHR Service Water and Emergency Service Water Sys	84
M-120	PandID Residual Heat Removal System	76
M-121	PandID Residual Heat Removal System	76
M-133	PandID Diesel Oil System	76
M-134	PandID Fire Protection System Interior Locations	70
M-143	PandID Primary Containment and Atmospheric Control System	77
M-343	Plans and Sections, HandV Standby Diesel Generator Building	1/21/69
M-811	PandID Service Water System and Make-Up Intake Structure	91
NE-100346	Div I and Div II 120V Instrument AC Distribution Panel Schedules	M
NE-36347-10	No. 142-480V MCC B42	AA
NE-36347-13	No. 133 and No. 143-480V MCC B33 and B43	N
NE-36347-15	No. 134-480V MCC B34	N
NE-36347-1A	480V MCC Schedules	77
NE-36347-8	No. 133-480V MCC B33	79
NE-36394-18	Monticello Nuclear Generating Plant Emergency Service Water Pumps	76
NE-36404-18	RHR System Auxiliary Controls	L
NE-36404-18A	Schematic Diagram Reactor Auxiliary Systems	K
NE-36771-3	Instrument AC and Uninterruptible AC Panel Schedules Y10, Y20 andY30	76
NE-93576	Single Line Diagram 480V MCC B34	J

DRAWINGS

Number	**Description or Title**	**Revision**
NF-36176	Generator Aux Transformer and 4160 Volt System Buses 11 and 12	78
NF-36177	Single Line Meter and Relay Diagram 4160 Volt System Buses No. 13, No. 14, No. 15, and No. 16	77
NF-36298-1	Electrical Load Flow One Line Diagram	81
NF-36298-2	DC Electrical Load Distribution One Line Diagram	78
NL-96763-1	Pipe Support SR-526	A
NL-96792-1	Pipe Support SR-147	A
NX-28921-3	Transformer, Class A, Core form, Outdoor, 3 Phase, 60 Hertz KVA-50000-Namplate Drawing	A
NX-7905-46-11	Elementary Diagram Residual Heat Removal System	J
NX-7905-46-14E	11 RHR Containment Spray Outboard Isolation MO-2020, Scheme B3339	76
NX-7905-46-16	Residual Heat Removal System Elementary Diagram	J
NX-7905-46-17	Elementary Diagram Residual Heat Removal System	77
NX-7905-46-18	Elementary Diagram Residual Heat Removal System	R
NX-7905-46-19	Elementary Diagram Residual Heat Removal System	P
NX-8714-2-4	MCC B33(A) Turbine Building Elevation 971'-0"	H
NX-8714-2-5	No. 133-480V MCC B33 (A) Rear Essential Turbine Bldg-East	78
NX-8714-4-2	Motor Control Center B43-B No. 143 (B) 480V	H
NX-8763-23	V-AC-4 Air Cooling Unit	C
NX-8763-23-1	RHR/Core Spray Pump Room Cooling Coil (V-AC-4)	A
NX-9525-1	RHRSW Pumps – Vertical Pump Dimensions/Layout	76
NX-9525-18	RHRSW Pumps – Bowl Assembly	A
USAR Figure 8.7-1	Instrument AC and Uninterruptible AC Distribution System Single Line	22

MISCELLANEOUS

Number	**Description or Title**	**Date or Revision**
	IST Stroke Time Data for MO2007, MO2013, and MO2020	
	14 ESW Quarterly Pump Test Data (05/22/07 – 07/22/09)	
	Phase 2 Risk Evaluation of CAP AR00076349, Loss of Containment Overpressure Due to Spurious Operation of the Torus and Drywell Purge and Vent Valves	07/14/09
4 AWI-04.05.12	Replacement of Failed Fuses	4
5828/6453-E-6	Specification for 480 Volt Motor Control Centers for the Monticello Nuclear Generating Plant – Unit 1	0
8285	Non-Identical Fuse replacement	8
B779R3	60,000 GAL U.G. Oil TK	09/05/67

MISCELLANEOUS

Number	Description or Title	Date or Revision
CML	Component Master List for CV-1729 Loop Components: E/P 1729, FT-10-97B, FI-4105, FI-10-132B, DPIC-10-130B, DPI-10-130B, FI-7188	02/21/05
EC 13790	Outboard MSIV Spring Degradation Evaluation	02/23/09
EC 14932	Evaluation of Inboard MSIV Closure Margin Using Existing Alternate Nitrogen System Minimum Pressure	09/23/09
EWI-08.15.02	Motor Operated Valve Program Engineering Standards	8
FORM 3090	IST Program Pump Data Sheet for No. 11 RHR Pump (202A)	5
Job No. 5828	Civil Structural Design Criteria for the Monticello Nuclear Generating Plant-Unit 1	1
LER 2009-01	Containment Overpressure Not Ensured in the Appendix Analysis	09/18/09
Letter to NRC	IE Bulletin 88-04 Response	07/08/88
Letter to NRC	IE Bulletin 88-04 Final Response	12/13/88
Letter To NSP	Sulzer Bingham Pumps INC., (Vendor Information Responding to Bulletin 88-04 for RHR Pumps)	11/08/88
MPS-1100	Specification for The Analysis of Piping and Pipe Support Systems, NPD-M-038	8
MWI-3-M-2.01	AC Electrical Load Study	12
MWI-3-M-2.06	Fuse/Breaker Coordination Study and Electrical Coordination	9
NX-17496-3	MNGP Protective Relay Cards – 4KV	3
NX-17496-4	MNGP Protective Relay Cards – 480V	1
NX-7905-55	11 RHR Pump Curve Vendor Curve (No. 26653)	A
NX-7905-56	11 RHR Pump Curve Vendor Test Data	A
NX-7905-58	13 RHR Pump Curve Vendor Curve (No. 26680)	A
NX-7905-59	Residual Heat Removal Pump 270430	A
NX-7905-70-1	11 RHRSW Pump Curve	03/31/75
NX-7905-72-1	13 RHRSW Pump Curve	03/31/75
SAR01116710	CDBI Focused Self-Assessment	12/17/08
SAR01166267	CDBI FSA Effectiveness and Inspection Readiness	07/21/09

MODIFICATIONS

Number	Description or Title	Date or Revision
04Q030	EDG Ventilation System Upgrades	0
EC 11126	EPU-MOD 11-Balance of Plant Piping Support Modifications	0
EC 12361	EPU-Provide Operations the Ability to Throttle MO-2020 and MO-2021	0
EC 14085	Security Improvements – 2009- Force on Force Project	0
EC 9419	EDG Rooms Sprinkler Modification	0

MODIFICATIONS

Number	**Description or Title**	**Date or Revision**
EC-11734	Modification No. 5 1AR Transformer Replacement, Monticello Plant EPU Project	4
SRI-91-010	USAR Update Concerning Diesel Oil Day Tank	12/01/92

OPERABILITY EVALUATIONS

Number	**Description or Title**	**Date**
AR 01079705	LAR Required for Use of TORMIS Code Methodology	02/28/07
CA-03-124	ESW Model ESW2B-Operability Evaluation	06/24/03
CA-03-125	ESW Model ESW1B-Operability Evaluation	06/24/03
FA 010709705	Modify EDG Exhaust Silencer Lines to Restore the EDGs Within a Reasonable Time Frame	10/01/09
OPR 1076631	Possibility of Inadequate Flow to Room Cooler	03/08/07
OPR 1101934	Operability Determination on 14 RHR Pump	07/22/07
OPR 1130761	Flow Control Problems with “B” RHRSW Valve CV-1729,	03/20/08
OPR 1169854-1	Outboard MSIVs Have Not Been Tested IAW Testing Requirements	02/23/09
OPR 1199936-1	Terminal Voltage at SV-1995 and SV-1728	10/02/09
OPR1193840-1	Motors Supplied by 480V MCCs With Voltage Less than 90% Rated Voltage	08/21/09

PROCEDURES

Number	**Description or Title**	**Revision**
0137-15-01	Containment Spray Loop “A” Isolation Valve Local Leak Rate Test	10
0137-29	LPCI Loop “B” Injection Valves Local Leak Rate Test	7
0187-01	11 EDG/ESW Quarterly Pump and Valve Tests	70
0192	Diesel Fuel Quality Check	26
0255-04-IA-1-1	RHR Loop A Quarterly Pump and Valve Tests	78
0255-04-IA-1-2	RHR Loop B Quarterly Pump and Valve Tests	80
1052-03	11 Diesel Generator Auxiliary Systems Test	11
1487	Site Housekeeping Quarterly Inspection	5
4066-PM	D10 Battery Charger Preventive Maintenance	0
4106-01-PM	Emergency Diesel 1 Cycle Maintenance	24
4844-PM	GE Thermal Overload Relay Test Procedure	20
4846-PM	GE/W Molded Case Circuit Breaker Maintenance and Test Procedure	18
4847-PM	7700 Line Motor Control Center Maintenance Procedure	19
4858-60-PM	1AR Reserve Transformer Maintenance	12

PROCEDURES

Number	Description or Title	Revision
6-C-3	Diesel Service Oil Pump Tripped	2
6-C-6	Diesel Generator Tank T-45A Level/Flow Low	2
6-C-7	Diesel Generator Tank T-45B Level/Flow Low	3
8196	Temporary Shielding Installation	4
8900	Operation of RCIC without Electric Power	2
8-B-19	ESW Pump 11 Lo Discharge Pressure	4
8-B-20	ESW Pump 11 OL/MAN Override	4
A.6	Acts of Nature	31
B.03.04-06	Residual Heat Removal System - Figures	5
B.08.01.03-01	RHR Service Water System – Function and General Description	10
B.08.01.03-05	RHR Service Water System-System Operation	39
B.08.04.01-05	Instrument and Service Air	10
B.08.05-05	Fire Protection - System Operation	45
B.08.11-05	Diesel Oil System	17
B.09.06-02	4.16 KV Station Auxiliary	10
B.09.13-06	Instrumentation AC and Uninterruptible AC Distribution System - Figures	5
C.4-B.08.01.03-A	Loss of Instrument Air	17
C.4-B.09.02.B	Loss of Normal Offsite Power	11
C.4-C	Shutdown Outside Control Room	32
C.4-H	Restoration Of Plant Loads	13
C.5.1-1000	EOP Introduction	20
C.5.1-1100	RPV Control	8
C.5-3203	Use of Alternate Injection Systems for RPV Makeup	10
C.5-3502	Containment Spray	14
FP-E-MOD-04	QF-0515a Design Input Checklist	10
FP-E-MOD-04	QF-0515b Design Input Checklist	3
FP-E-MOD-06	QF-0525 Design Description Form	2
FP-OP-COO-01	Conduct of Operations	6
FP-OP-OL-01	Operability / Functionality Determination	5
OSP-DOL-0543	Fuel Oil Receiving Quality Check	2
OSP-EDG-0540-11	11 Emergency Diesel Generator 24 Month Test	2
OWI-03.07	Time Critical Operator Actions	0

SURVEILLANCES (COMPLETED)

Number	**Description or Title**	**Date or Revision**
0255-05-IA-1-1	"A" RHRSW Quarterly Pump And Valve Tests	05/15/09 05/27/09 06/11/09
0198-01	125 VDC Battery Capacity Test	03/31/05
4510-PM	Maintenance of Onsite Batteries and Battery Chargers	01/25/07
OSP-EDG-0540-11	11 Emergency Diesel Generator 24 Month Test	04/04/09
0187-01B	11 Emergency Diesel Generator/11ESW/Monthly Pump and Valve Tests	06/14/09 08/09/09
0187-01	11 Emergency Diesel Generator/11ESW/Quarterly Pump and Valve Tests	07/12/09

WORK ORDERS

Number	**Description or Title**	**Date**
00002043	Valcor Solenoid Valve Control Assembly	12/11/01
00134749	Pre-Op Test New 125V DC Division 1 Battery Charger D10	08/15/08
00200755	PM 4900-1, VOTES for MO2013	05/05/03
00200759	MO2020 VOTES, Inspect LS	05/16/03
00285275-01	PM 4106-01-PM, 11 EDG G-3A, Perform Electrical PM	10/09/06
00293778	P-111A, Rebuild Pump and Return to Stock	06/05/09
00306982	EC210 O Rings On Valcor Control Assembly J-box	03/14/05
00307012	11 RHR Pump Minimum Flow, (PM CV-1994 Air Operator)	03/14/05
00307804	Make Repairs to Support SWH-47	05/28/03
00307805	Make Repairs to Support SWH-42	05/17/03
00309248	11 RHR Pump Minimum Flow, Baseline Testing on CV-1994	03/15/05
00311664	D-10, 125VDC Charger for 11 Battery 480V Supply	07/21/04
00311666	MO-1427, RBCCW INL to DW Cooler V-CC-1 480V Supply	03/29/05
00311667	MO-1429, RBCCW INL to DW Cooler V-CC-3 480V Supply	03/29/05
00311670	P-88B, ECCS Area Drain Pump B (480V Supply)	05/26/04
00336541	Monitor ADS Pneumatic Supply	01/22/08
00343696-01	PM 4030-01-PM, 11 Diesel Generator Control Panel C-91/93	03/07/09
00343716	CV-1994, Valve Position Indication Test, (0255-03-IA-2B)	05/27/09
00343738	CV-1728 Diagnostic Test: Post Outage PM work.	04/04/09
00343827-01	PM 4851-12 (52-304) MCC-133A Feeder Breaker Cubicle	03/09/09

SURVEILLANCES (COMPLETED)

Number	**Description or Title**	**Date or Revision**
00343889	Limit Switch (AO-2-86D)	04/23/09
00343994	PM 4058-6 (B RHR RM Air Cooling Unit V-AC-4)	07/24/09
00344785	Monitor ADS Pneumatic Supply	03/18/08
00349292	Monitor ADS Pneumatic Supply	05/14/08
00352042	Perform Coupling Inspection of P-77	07/15/08
00352697	Perform Coupling Inspection of P-11	07/08/08
00354013	D10 125 VDC Charger 24 Month Capacity Test	06/13/08
00354414	MO2007 Rebuild Actuator and Perform 4900-01-PM	04/04/09
00355172	CV-1728, Perform IandCPM 7070 (A)on RSW-1 Instruments	09/15/08
00356497	CV-1729, Comprehensive 11 RHRSW Pump and Valve Tests	09/10/08
00356504	11 EDG Auxiliary Systems Test	03/29/09
00356593-01	PM 4109-01-PM, 11 EDG 12 Year Maintenance	03/07/09
00356627	Battery 11 Modified Performance Test	03/29/09
00359064-11	Perform Acceptance Testing of 1AR Transformer	07/24/09
00362759	0255—04-III-3A Comprehensive 13 RHR Pump and Valve Test	12/09/08
00362760	CV-1729, Comprehensive 11 RHRSW Pump and Valve Tests	12/26/08
00363012	Division 1 LPCI Pump Discharge Flow-Low Bypass Channel Calibration, (ISP-RHR-0547-01)	12/15/08
00363156	0255-04-IA-11 RHR Loop A QRTRLY Pump and Valve Test	07/20/09
00366681	CV-1729, Comprehensive 11 RHRSW Pump and Valve Tests	03/13/09
00374280	11 and 12 125 VDC Battery Weekly Surveillance	06/30/09
00375976	11 EDG/ESW Quarterly Pump and Valve Tests	07/10/09
00377539	11 EDG/ESW Quarterly Pump and Valve Tests	08/09/09
00377780-01	Perform Major PM on Spare Breaker LCB-037	04/04/09
00379154	0255-04-IA-11 RHR Loop A QRTRLY Pump and Valve Test	12/09/08
00385117	Diesel Fuel Oil Receiving Quality Check	05/19/09
00403241	11 RHR Pump Minimum Flow, (CV-1994 failed to Operate)	09/17/04
09906286	Measure Air Flow on V-AC-4 and V-AC-5	07/01/09

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AISC	American Institute of Steel Construction
AR	Action Request
ASME	American Society of Mechanical Engineers
CDBI	Component Design Bases Inspection
CFR	Code of Federal Regulations
CPT	Control Power Transformer
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
GE	General Electric
GL	Generic Letter
IEEE	Institute of Electrical and Electronic Engineers
IN	Information Notice
IMC	Inspection Manual Chapter
IR	Inspection Report
IST	Inservice Test
kV	Kilovolt
LC	Load Center
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-site Power
MCC	Motor Control Center
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NEMA	National Electrical Manufactures Association
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records
PM	Preventative Maintenance
PRA	Probabilistic Risk Assessment
psig	Pounds Per Square Inch Gauge
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIS	Regulatory Information Issue
SBO	Station Blackout
SDP	Significance Determination Process
SER	Safety Evaluation Report
SPAR	Standardized Plant Analysis Risk
SV	Solenoid Valve
TOL	Thermal Overload
TS	Technical Specification
USAR	Updated Safety Analysis Report
URI	Unresolved Item
Vac	Volts Alternating Current
Vdc	Volts Direct Current

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/RA/
Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

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