August 3, 2006

Mr. J. Conway
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT
NRC COMPONENT DESIGN BASES INSPECTION (CDBI)
INSPECTION REPORT 05000263/2006009(DRS)

Dear Mr. Conway:

On June 23, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Monticello Nuclear Generating Plant. The enclosed report documents the inspection findings which were discussed on June 23, 2006, with you and other members of your staff.

The inspection examined activities conducted under your license, as they relate to safety, and to 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. Specifically, this inspection focused on the design of components that are risk significant, and have low design margin.

Based on the results of this inspection, one NRC-identified finding of very low safety significance, which involved a violation of NRC requirements was identified. In addition, one issue was reviewed under the NRC traditional enforcement process and determined to be a Severity Level IV violation of NRC requirements. However, because these violations were of very low safety significance, and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, a licensee identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of a Non-Cited Violation, 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 Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), 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 05000263/2006009(DRS)
w/Attachment: Supplemental Information

cc w/encl: M. Sellman, Chief Executive Officer
and Chief Nuclear Officer
Manager, Nuclear Safety Assessment
J. Rogoff, Vice President, Counsel, and Secretary
Nuclear Asset Manager, Xcel Energy, Inc.
Commissioner, Minnesota Department of Health
R. Nelson, President
Minnesota Environmental Control Citizens
Association (MECCA)
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer,
Wright County Government Center
Commissioner, Minnesota Department of Commerce
Manager - Environmental Protection Division
Minnesota Attorney General’s Office

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	05000263/2006009(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant (MNGP)
Location:	Monticello, Minnesota
Dates:	May 8, 2006, through June 23, 2006
Inspectors:	A. Dunlop, Senior Reactor Engineer, Lead Inspector M. Bielby, Operations Inspector S. Burgess, Senior Reactor Analyst M. Munir, Reactor Engineer G. O'Dwyer, Reactor Engineer N. Valos, Operations Inspector C. Baron, Mechanical Contractor L. Hajos, Electrical Contractor
Observer:	M. Melnicoff, Engineering Inspector
Approved by:	A. M. Stone, Chief Engineering Branch 2 Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000263/2006009(DRS); 05/08/2006 - 06/23/2006; Monticello Nuclear Generating Plant; Component Design Bases Inspection.

The inspection was a 4-week onsite baseline inspection that focused on the design of components that are risk significant and have low design margin. The inspection was conducted by five regional engineering inspectors and two consultants. One Severity Level IV Non-Cited Violation (NCV), and one Green finding associated with an NCV were identified. The significance of most findings are indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 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 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

C Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving the voltage drop on cables from motor control center (MCC) 134 to various loads. Specifically, the inspectors identified that the licensee failed to verify the calculation assumption that the voltage drop from the MCC to the load was below 2.5 percent. In several cases, this assumption was not met, which resulted in little or no available margin to the safety-related equipment. There was not an operability issue, as the voltage at the loads was determined to be adequate for the equipment to function. The licensee's corrective action included performing an extent of condition to identify additional cables, that may not meet this design assumption, and entered this performance deficiency into their corrective action program for resolution.

  The finding was more than minor because the failure to adequately verify documented assumptions in design calculations could result in failure of the motors to operate properly during starting or running, (i.e., creating design margin capability that would not exist) and could have affected the mitigating systems cornerstone objective of design control. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.1)

- Severity Level IV. The inspectors identified a Non-Cited Violation of 10 CFR 50.59, "Changes, Tests, and Experiments," which had very low safety significance (Green). Specifically, the licensee failed to complete a 50.59 evaluation for an operating procedure change that substituted remote manual operator actions for automatic actions during a station blackout. This procedure change directed the operators to control the reactor vessel water level by manually operating the high pressure core injection pump during a station blackout, bypassing the automatic injection controls. The licensee entered this performance deficiency into their corrective action program for resolution.

The finding was more than minor because the inspectors could not reasonably determine that these procedure changes would not have ultimately required prior approval from the NRC. This finding was categorized as Severity Level IV because the underlying technical issue for the finding was determined to be of very low safety significance using the SDP Phase 1 screening worksheet. (Section 1R21.4.b.1)

B. Licensee-Identified Violations

One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

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 inspectible 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 3.31. In general, the selection was based upon the components and operator actions having a risk achievement worth of greater than 2.0 and/or a risk reduction worth of 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. Since all plant components were not modeled in the licensee's PRA, additional resources were used in the selection process such as the licensee's maintenance rule program, where an expert panel identified additional systems/components that also were considered risk significant.

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 modifications or power uprates, or reductions due to degraded material conditions. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed 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 equipment, system health reports, and the potential margin issues list. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. As practical, the inspectors performed walkdowns of the components to evaluate the as-built design and material condition. A summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Updated Safety Analysis Report (USAR), Technical Specifications (TS), component/system design basis documents, drawings, 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, and the Institute of Electrical and Electronics Engineers (IEEE) Standards, to evaluate acceptability of the systems' design. The review was to verify that the selected components would function as 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 were consistent with the design bases and were appropriate may include installed configuration, system operation, detailed design, system testing, equipment/environmental qualification, equipment protection, component inputs/outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, system health report, and corrective action process documents (CAPs). Walkdowns were conducted for all accessible components to assess material condition and to verify the as-built condition was consistent with the design. Other attributes reviewed are included as part of the scope for each individual component.

The components (17 samples) listed below were reviewed as part of this inspection effort:

- High Pressure Core Injection (HPCI) Pump: The inspectors reviewed various analyses, procedures, and test results associated with operation of the HPCI pumps under transient, accident, and station blackout conditions. The analyses included hydraulic performance, net positive suction head (NPSH), minimum flow, potential water hammer conditions, and transfer of the suction source. The inspectors also evaluated the pump suction trip setpoint to verify that the pump would not inadvertently trip under transient conditions. The control logic and testing of associated valves were reviewed during the inspection. The inspectors also evaluated the impact of the HPCI pump on the capacity of the station batteries during station blackout conditions. In addition, the licensee responses and actions to Bulletin 88-04, "Potential Safety-Related Pump Loss," were reviewed to assess implementation of operating experience.

- HPCI Room Cooler: The inspectors reviewed analyses addressing the maximum potential HPCI room temperatures under accident and station blackout conditions, when room cooling would not be available. This review verified the capability of required HPCI equipment to perform its required function with elevated room temperatures. The inspectors also reviewed operating procedures directing the operators to open the HPCI room doors under station blackout conditions.

- HPCI Injection Valve (MO-2068): The inspectors reviewed the motor-operated valve (MOV) calculations, including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis, to ensure the valve was capable of functioning under design conditions. The inspectors also reviewed the control logic schematic and flow control diagrams to verify the adequacy of valve control logic design. Diagnostic and inservice testing (IST) results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Regulatory Information Summary (RIS) 2001-15, "Performance of DC-Powered Motor-Operated Valve Actuators," was reviewed to ensure it was properly evaluated and implemented as appropriate.

- HPCI Minimum Flow Valve (CV-2065): The inspectors reviewed the capability of the air-operated valve (AOV) to cycle open and closed as required under accident and station blackout conditions, including conditions where the normal supply of instrument air would not be available. The capacity and testing of the associated air accumulator tank was reviewed during the inspection. The inspectors also reviewed the AOV calculation to ensure the valve was capable of functioning under design conditions. The inspectors reviewed control logic schematic and flow control diagrams to verify the adequacy of valve control logic design, including a modification associated with this logic. In addition, the inspectors reviewed the analyses addressing the design and the capacity of the associated nitrogen supplies, including the procedures for leak testing the nitrogen system, to ensure that there was sufficient capacity to operate the AOV.

- Reactor Core Isolation Cooling (RCIC) Pump: The inspectors reviewed various analyses, procedures, and test results associated with operation of the RCIC pumps under transient and station blackout conditions. The analyses included hydraulic performance, NPSH, minimum flow, and transfer of the suction source. The inspectors also evaluated the pump suction trip setpoint to verify that the pump would not inadvertently trip under transient conditions. The control logic and testing of associated valves were reviewed during the inspection. The inspectors also evaluated the impact of using the RCIC pump on the capacity of the station batteries during station blackout conditions. In addition, the licensee responses and actions to Bulletin 88-04 was reviewed to assess implementation of operating experience. Modifications associated with the minimum flow valve control logic and the auto suction switchover were also reviewed to ensure proper implementation.

- RCIC Turbine Steam Supply Valve (MO-2078): The inspectors reviewed MOV calculations, including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis, to ensure the valve was capable of functioning under design conditions. The inspectors also reviewed the control logic schematic and flow control diagrams to verify the adequacy of valve control logic design. Diagnostic and IST test results were reviewed to verify acceptance criteria were met and performance degradation would be identified. The inspectors reviewed RIS 2001-15 to ensure it was properly evaluated and implemented as appropriate.

C RCIC Lube Oil Heat Exchanger: The inspectors reviewed vendor information and test results to verify the capability of this heat exchanger to perform its required function under transient and station blackout conditions. The inspectors verified that the heat exchanger would perform it's function with the most limiting torus water temperatures.

C Essential Service Water (ESW) Pump: The inspectors ensured river levels met vendor requirements for ESW pump suction submergence and NPSH to ensure the pump was capable of performing its safety functions. The inspectors verified that appropriate procedures were in place for potential low river levels. Hydraulic calculations were reviewed to ensure design requirements for flow and pressure were appropriately translated as acceptance criteria for pump IST. Design change history and IST results were reviewed to assess potential component degradation and impact on design margins. The ESW pump replacement test was also reviewed to verify the pump's ability to perform its design safety functions.

C Automatic Depressurization System (ADS) Valve's Alternate Nitrogen System: The inspectors reviewed design calculations for sizing of the alternate nitrogen system, pressurized nitrogen bottles, and accumulator tanks to ensure the ADS valves would function as designed. The inspectors also reviewed the nitrogen leak rate testing procedures and recently completed leak rate tests performed for the alternate nitrogen system to verify that the acceptance criteria were consistent with design requirements and test results were within the defined criteria.

C Residual Heat Removal Division 2 Low Pressure Core Injection (LPCI) Outboard Valve (MO-2013): The inspectors reviewed the MOV calculations, including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis, to ensure the valve was capable of functioning under design conditions. The inspectors reviewed testing and logic diagrams to ensure the valve would function correctly based on LPCI loop selection logic. Local leak rate, diagnostic, and IST test results were reviewed to verify acceptance criteria were met and performance degradation would be identified.

C Emergency Diesel Generator (EDG) No.12: The inspectors reviewed calculations and drawings to determine if the size of the EDG was within equipment ratings. The inspectors reviewed the adequacy and appropriateness of design assumptions and calculations related to EDG protection and relay coordination. The inspectors reviewed design calculations to ensure fuel tank capacities were sufficient to meet required fuel oil consumption rates and to ensure vortexing would not occur in the fuel oil tanks. The inspectors ensured tank capacity tests demonstrated design basis required capacity. To ensure the quality of the fuel oil, the inspectors verified that an appropriate chemical control program for fuel oil was in place, such as moisture and impurity controls. The inspectors performed a review of system normal operating procedures and surveillance test procedures to ensure component operation and alignments were consistent with design and licensing bases assumptions.

C 4160 Volts (V) Bus 16: The inspectors reviewed calculations and drawings to determine if the loading of 4160V Bus 16 was within equipment ratings. The inspectors reviewed the adequacy and appropriateness of design assumptions and calculations related to motor starting and loading voltages to determine if the voltages across motor terminals, under worse case motor starting and loading conditions, would remain above the minimum acceptable values. The inspectors also reviewed load flows for light loading conditions to determine equipment operability at high voltages. The inspectors performed a review of system normal operating procedures and surveillance test procedures and acceptance criteria to verify that Bus 16 was capable of supplying the minimum voltage necessary to ensure proper operation of connected equipment during normal and accident conditions. The inspectors reviewed the adequacy of protective device coordination provided for a selected sample of equipment.

C 480V Motor Control Center (MCC) 144: The inspectors reviewed calculations and drawings to determine if the size of 480V MCC 144 was within equipment ratings. The inspectors reviewed the adequacy and appropriateness of design assumptions and calculations related to in-feed transformer protection and relay coordination. On a sample basis the inspectors reviewed maintenance and test procedures and acceptance criteria to verify that the 480V MCC 144 was capable of supplying power necessary to ensure proper operation of connected equipment during normal and accident conditions.

C 250Vdc [volts direct current] Division II Safety Related Battery No.16, Charger, and Distribution Panel: The inspectors reviewed various electrical documents including battery load and margin, charger sizing calculations, battery float and equalizing voltages, overall battery capacity, performance discharge test, weekly battery surveillance tests, short circuit calculation, breaker interrupting ratings, and electrical coordination. The inspectors also reviewed electrical schematics for selected 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Plants Operating Prior to January 1, 1979," circuits to ensure that coordination existed between the downstream and the upstream fuses. The inspectors reviewed analyses addressing both the minimum and maximum potential battery room temperatures associated with both accident and station blackout conditions. The inspectors also verified that the design conditions for the batteries and associated equipment would be maintained for the duration of the applicable events. The replacement of the battery was also reviewed to verify the design functions of the battery was maintained.

C 125Vdc Division II Safety Related Battery No.13, Charger, and Distribution Panel: The inspectors reviewed various electrical documents including battery load and margin, charger sizing calculations, battery float, and equalizing voltages, overall battery capacity, performance discharge test, short circuit calculation, breaker interrupting ratings and electrical coordination. The inspectors also reviewed electrical schematics for selected 10 CFR Part 50, Appendix R circuits to ensure that coordination existed between the downstream and the upstream fuses.

- Hard Pipe Vent Valves (AO-4539/4540): The inspectors reviewed the AOV calculations, including required thrust, actuator output, maximum differential pressure, setpoint, and margin analyses, to ensure the valves were capable of performing their required functions under design conditions. Diagnostic and IST results, including local leak rate tests, were reviewed to verify acceptance criteria were met and performance degradation would be identified.

- Cross-tie Valve to Low Pressure Coolant Injection (RHRSW-14): The inspectors reviewed the hydraulic calculation to verify that the diesel fire pump and residual heat removal service water (RHRSW) pumps would develop sufficient pressure to inject into the reactor vessel through the RHRSW-14 valve while in the emergency lineup. Since the cross-tie valves were not normally used, the inspectors verified that the cross-tie valve and associated connecting piping were inspected and tested to verify that the system would pass the required flow.

b. Findings

One finding of very low safety significance associated with a Non-Cited Violation (NCV) was identified.

b.1 Voltage Drop from MCC to Motor Terminals

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green) involving the voltage drop on cables from MCC 134 to various loads. Specifically, the inspectors identified that the licensee failed to verify the calculation assumption that the voltage drop from the MCC to each of the loads was below 2.5 percent. In several cases, this assumption was not met, which resulted in little or no available margin to the safety-related equipment.

Description: All motors fed from MCC 134 were rated at 460V. The load flow calculation for degraded grid, CA-93-066, “AC Load Study, Degraded Voltage Setpoint 1R XFRM LOCA Load,” indicated that the minimum allowable voltage at the MCC was 426V (92.6 percent of 460V). The licensee, in lieu of determining each motor’s terminal voltage, made a generic assumption that the voltage drop on all cables between the MCC and the load was less than 2.5 percent, and therefore the motors met the running criteria of having 90 percent voltage at their terminals. The motors were designed to run at a minimum of 90 percent voltage and start at 80 percent voltage.

The inspectors questioned what the motor terminal voltage was under a degraded grid voltage condition (worst case voltage), which the licensee determined to be 427V at MCC 134. Based on this voltage, the licensee calculated that in two cases, the voltage drop assumption was not met. These were for cables 1B3435-A, emergency service water pump, P-111A, and 1B3474-A, EDG No.11 room supply fan, V-SF-10. Although the calculation assumption was not met in these two cases, the motor terminal voltage still met the running criteria of 90 percent of 460V or 414V, such that there was not an operability concern. The inspectors concluded that the voltage margin for operating these components was reduced to little or no available margin due to the failure to verify the calculation assumption. The licensee initiated CAP 01032362 to address this issue.

Analysis: The inspectors determined that the failure to account for the actual voltage drops on the cables feeding the 480V motors could create conditions where the motor would not be able to perform its safety function was a performance deficiency and a finding. The inspectors determined that the finding was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Dispositioning Screening," because it was associated with the attribute of design control, which affected the mitigating systems cornerstone objective of ensuring the availability and reliability of safety-related motors to respond to initiating events to prevent undesirable consequences. Specifically, the failure to establish adequate voltage to the motors could impact their safety function.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, "Operability Determinations, and Functionality Assessments for Resolution of Degraded, or Nonconforming Conditions Adverse to Quality or Safety," did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. The basis for this conclusion was, that despite the higher than assumed voltage drop on these cables, and therefore, loss of design margin in the motor terminal voltage, there was still adequate voltage for the motors to perform their safety function. The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, 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, as of June 28, 2006, the licensee's design control measures failed to verify the adequacy of design, in that the assumption in calculation CA-93-066 that stated the voltage drop on all cables between the MCC and the load was less than 2.5 percent, which was not met in all cases. Specifically, two motors on MCC 144 had voltage drops greater than assumed, which resulted in little or no running voltage margin under a degraded grid voltage condition. The licensee entered the finding into their corrective action program as CAP 01032362 to perform an extent of condition and revise the affected calculation. Because this violation was of very low safety significance, and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000263/2006009-01(DRS).

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed seven operating experience (OE) issues (7 samples) to ensure these issues, either NRC generic concerns or identified at other facilities, had been adequately addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection effort:

- Information Notice 05-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design";
- Station Blackout Rule (10 CFR 50.63, "Loss of All Alternating Current Power");
- RIS 2001-05, "Performance of DC-Powered Motor-Operated Valve Actuators";
- Bulletin 88-04, "Potential Safety-Related Pump Loss";
- OE 021914, "RCIC Governor Resistor Found Failed Open During Post Maintenance Calibration";
- OE 021373, "HPCI Testing Caused a RCIC Trip"; and
- OE 025352, "Engine systems, Inc. Report No. 10CFR21-0089, Revision 0, Woodward Governor "Compensating" EG Series Actuators."

b. Findings

One finding of very low safety significance associated with a Non-Cited Violation was identified.

b.1 Failure to Perform a 50.59 Evaluation for a Operating Procedure Revision

Introduction: The inspectors identified an NCV of 10 CFR 50.59, "Changes, Tests, and Experiments," having very low safety significance involving a procedure change for makeup to the reactor vessel. Specifically, the licensee failed to complete a 10 CFR 50.59 evaluation for a operating procedure change, which substituted remote manual operator actions for automatic actions during a station blackout event. This procedure change directed the operators to control the reactor vessel water level by manually operating the HPCI pump during a station blackout. This action included defeating the automatic start of the HPCI pump, which was based on low-low reactor vessel water level.

Description: The inspectors reviewed abnormal operating procedure C.4-B.09.02.A, "Station Blackout," revision 28, and noted that it directed the operators to manually control reactor vessel water level using either the RCIC or HPCI system in the event of a station blackout. The procedure directed the operators to use RCIC, if available. If RCIC was not available, the procedure directed them to use HPCI to maintain reactor vessel water level. In either case, the procedure directed the operators to place HPCI in “pull-to-lock,” defeating its automatic start function and to manually control the reactor vessel water level between +48 inches and -126 inches. The minimum water level of -126 inches corresponded to the top of active fuel. The automatic start setpoint for the HPCI pumps was -47 inches (low-low level). The inspectors noted that USAR Section 6.2.4 described the automatic initiation of HPCI, but did not include any descriptions of operating HPCI in this manual mode. The inspectors questioned when this operating mode had been first included in the procedures, and if there was a 10 CFR 50.59 evaluation associated with that procedure revision.

In response to these questions, the licensee determined that Revision 7 of operating procedure C.4-B.9.02.A (dated March 19, 1993), was the first procedure that directed the operators to place HPCI in “pull-to-lock,” defeating its automatic start function when HPCI was in service (previous procedures placed HPCI in “pull-to-lock” only when directing the operators to control reactor vessel level using RCIC). In addition, Revision 7 was the first station blackout procedure issued to implement the plants commitment to 10 CFR 50.63, “Loss of All Alternating Current Power,” which was based on using the HPCI system to control level.

With regard to the 10 CFR 50.59, the licensee determined that no formal evaluation had been issued for Revision 7 of C.4-B.9.02.A. Plant records indicated that the Operations Committee had reviewed and approved this procedure revision on March 18, 1993. In accordance with the procedure in effect (4 AWI-04.07.01, “Safety Review Item,” Revision 6), the Operations Committee was responsible to determine if an Unreviewed Safety Question was involved, and if reporting pursuant to 10 CFR 50.59 was required. The plant records did not provide any additional details of the meeting. The licensee also determined that the 10 CFR 50.59 procedure in effect (4 AWI-05.01.09, Revision 1) stated, in part, that a Safety Review Item (50.59 evaluation) shall be prepared if a new procedure or procedure change results in a deviation from steps listed in the USAR or will result in system operation which deviates from the way that system is described in the USAR.

The inspectors were concerned that this change had substituted remote manual operator actions for automatic actions during a station blackout, and that the minimum allowable reactor vessel water level was -126 inches (top of active fuel), as opposed to the automatic setpoint of -47 inches. The inspectors noted that this change would have required a Safety Review Item (50.59 evaluation) in accordance with plant procedure and the 10 CFR 50.59 rule in effect at the time.

In response, the licensee documented this issue in CAPs 01029702 and 01036407 during the inspection. Corrective actions included a revision to the USAR to describe the operation of the HPCI and RCIC systems during a station blackout event, and to perform the required 10 CFR 50.59 evaluations.

<u>Analysis</u>: The team determined that this issue was a performance deficiency since the licensee had failed to have a written evaluation as to why prior approval was not required under 10 CFR 50.59. The inspectors concluded that the violation was reasonably within the licensee’s ability to foresee and correct based on the procedures in effect at the time.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned under the traditional enforcement process instead of the SDP. However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the licensee failed to perform a safety evaluation in accordance with 10 CFR 50.59 for adverse changes made to HPCI operating procedures.

The finding was determined to be more than minor because the inspectors could not reasonably determine that these procedure changes would not have ultimately required prior approval from the NRC.

The inspectors evaluated the underlying technical issue using IMC 0609, Appendix A, Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation. The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: Title 10 CFR 50.59(b)(1) stated, in part, that the licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section, to the extent that these changes constitute changes in the facility as described in the safety analysis report or to the extent that they constitute changes in procedures as described in the safety analysis report.

Contrary to the above, on March 18, 1993, the licensee approved a procedure change that constituted a change to procedures as described in the safety analysis report without a 10 CFR 50.59 evaluation. Specifically, when procedure C.4-B.9.02.A. was revised to defeat the automatic start function of HPCI and to manually control the reactor vessel water level between +48 inches and -126 inches. In accordance with the Enforcement Policy, the violation was classified as a Severity Level IV violation because the underlying technical issue was of very low risk significance. The licensee entered the finding into their corrective action program as CAPs 01029702 and 01036407 to describe the operation of the HPCI and RCIC systems during a station blackout event revision in the USAR, and to perform the required 10 CFR 50.59 evaluations. Because this non-willful violation was non-repetitive and was entered into the licensee’s corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000263/2006009-02(DRS).

.5 Modifications

a. Inspection Scope

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

- Design Change No. 81Z055, "RCIC Auto Suction Switchover";
- Design Change No. 92Q290, "Flood Protection for the Lower 4kV Switchgear Room";
- Design Change No. 92Q615, "HPCI/RCIC Minimum Flow Logic Changes";
- Design Change No. 93Q415, "Emergency Diesel Generator Electrical Upgrades"; and
- Work Order No. 0105477, "Replace 250 Vdc Division II Battery No. 16."

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 five risk significant, time critical operator actions (5 samples). These actions were selected from the licensee's PRA rankings of human action importance based on risk achievement worth and Birnbaum 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 walk through of associated procedures with a plant operator to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required. The following operator actions were reviewed:

- Actions to manually operate the hard pipe vent system;
- Actions to shift control from the control room to the alternate shutdown system panel during a fire in the control room;
- Actions to manually operate the RCIC system after battery depletion;
- Actions to power the Division II 250 Vdc battery chargers from the 13 diesel or the security diesel; and

- Actions during a station blackout to: 1) align the main control board, 2) provide for HPCI and RCIC room cooling, 3) provide for electrical panel cooling, 4) backfeed Bus 13 from 13 diesel generator, and 5) align essential loads to Bus 13.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

.1 Review of Condition Reports

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.

4OA6 Meetings, Including Exits

.1 Exit Meeting

The inspectors presented the inspection results to Mr. J. Conway and other members of licensee management at the conclusion of the inspection on June 23, 2006. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Cornerstone: Barrier Integrity

Criterion III, “Design Control,” of 10 CFR Part 50, Appendix B, 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. The licensee did not adequately translate applicable regulatory requirements into the RCIC system design. Specifically, the control logic of the RCIC pump suction valves MO-2100 and MO-2101 did not meet the licensee's regulatory commitments to NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.K.3.22. The licensee stated, in a 1982 letter to the NRC, that the remote manual RCIC containment isolation had been retained. However, the as-installed design would not allow the operators to close the isolation valves from the control room in the event that the RCIC pump suction had automatically transferred from the condensate storage tank to the torus due to a low water level in the condensate storage tank. The licensee identified this issue based on an NRC finding at another facility with a similar design. The licensee documented this condition in CAP 01029334 during the inspection. The licensee also issued a revised RCIC operating procedure during the inspection. The revised procedure (B.02.03-05, Revision 17) provided directions to block specific relay contacts, allowing these RCIC valves to be closed if required.

The inspectors determined that the finding was more than minor because it was associated with the attribute of design control, which affected the barrier integrity cornerstone objective of ensuring the functionality of primary containment isolation. The finding was of very low safety significance because it did not represent an actual loss of system safety function, and screened as Green using the SDP Phase 1 screening worksheet.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
J. Conway, Site Vice President
D. Alstad, Electrical and Instrumentation Engineering Supervisor
R. Baumer, Licensing
S. Brown, Program Engineering Manager
F. Domue, Electrical Design Supervisor
N. French, Plant Engineering Supervisor
J. Grubb, Engineering Director
B. Guldemond, Nuclear Safety Assurance Manager
S. Hammer, Operations Principal Engineer
N. Haskell, Engineering Design Manager
T. Hurrle, Configuration Management Supervisor
R. Jacobs, Site Director for Operations
B. Mackissock, Operations Manager
A. Myrabo, System Engineering Manager
S. Porter, Equipment Reliability Supervisor
B. Sawatzke, Plant Manager
T. Wellumson, PRA Engineer
P. Young, Turbine Engineering Supervisor

Nuclear Regulatory Commission
S. West, Deputy Director, Division of Reactor Projects
A. M. Stone, Chief, Engineering Branch 2, DRS
S. Thomas, Senior Resident Inspector
R. Orlikowski, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000263/2006009-01	NCV	Voltage Drop Assumption in Calculation Was Not Met (Section 1R21.3.b.1)
05000263/2006009-02	NCV	Failure to Perform a 50.59 Evaluation (Section 1R21.4.b.1)

LIST OF DOCUMENTS REVIEWED

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

1R21 Component Design Bases Inspection

Calculations

Number	**Title**	**Revision**
CA-69-014	Diesel Generator Protective Relays Settings and Coordination	Revision 0
CA-89-048	EA-89-R070-01 Modification 892100 Relay Settings	Revision 0
CA-90-023, add. 1	Minimum Allowable Fuel Oil Storage Tank Level	Revision 0
CA-90-038	Control Room Heat-Up During a SBO Event	Revision 3
CA-90-071	No.11 Emergency Diesel Generator and No.12 Emergency Diesel Generator Neutral Grounding System	Revision 0
CA-91-001	125V DC Fault Current	Revision 0
CA-91-009	250V DC Fault Current	Revision 0
CA-91-069	AC Load Study, 1R XFRM, LOCA, 2 Core Spray Pumps Starting	Revision 7
CA-91-071	AC Load Study 1R XFRM Minimum loading	Revision 7
CA-91-072	AC Load Study, 1R XFRM, Full Plant Loading, RFP Start	Revision 6
CA-91-078	AC Load Study 2R XFRM, Reactor Bypassed, Full Load, No.11 RFP Start	Revision 7
CA-91-091	Plant Fault Study, 2R XFMR, 2RS Reactor By-Passed, Primary Tap - 5 percent (Boost) for Low Voltage Breakers.	Revision 8
CA-91-092, add. 2	Plant Fault Study, 2R XFMR, 2RS Reactor In-line, 2R Primary Tap - 5 percent (Boost) for High Voltage Breakers.	Revision 7
CA-92-65	HPCI System Motor Operated Valve Functional Analysis	Revision 6
CA-92-137	RCIC Room Heat-Up	Revision 0
CA-92-214	RHR System Motor Operated Valve Functional Analysis	Revision 11
CA-92-220	Degraded Grid Relay Set point calculation	Revision 1

Calculations

Number	Title	Revision
CA-92-224	Emergency Diesel Generator Loading	Revision 4
CA-92-295	Protective Settings for New LC103 480V Switchyard Lineup	Revision 1
CA-92-299	Stem Thrust Assessment 16" A/D Globe Valves: MO-2012 and MO-2013	Revision 1
CA-93-14	RCIC System Motor Operated Valve Functional Analysis	Revision 5
CA-93-066	AC Load Study, Degraded Voltage Setpoint 1R XFRM LOCA Load	Revision 5
CA-94-017	Calculation of Alternate Nitrogen System Operability Leakage Criteria	Revision 5
CA-95-109	Stem thrust Assessment 12" A/D Gate Valves: MO-2067 and MO-2068	Revision 0
CA-96-020	HPCI Room Transient Temperature	Revision 4
CA-96-068	AC Electrical Load Study Validation	Revision 1
CA-96-091	Effects of RHR/CS Pump Motor Starting Transients on MOV Performance	Revision 0
CA-96-169	HPCI/RCIC NPSH Evaluation	Revision 3
CA-96-200	Stem Thrust Assessment 3" A/D Globe Valve: MO-2078	Revision 3
CA-97-089	AC Voltage Study 2R to 1R transformer Auto Transfer with LOCA loading	Revision 1
CA-97-090	AC Voltage Study, Minimum 480 V Voltage Determination During Diesel LOOP/LOCA ECCS CS Start Test Condition	Revision 2
CA-97-232	Suction Line Submergence for Vortex Concern	Revision 1
CA-97-235	HPCI/RCIC Suction Transfer from CST Setpoint Calc.	Revision 2
CA-00-137	EDG Capability to accept Service Water Pump Loading	Revision 0
CA-01-032	Operability Evaluation of Torus Cooling	Revision 3
CA-01-045	Determination of Instrument Scaling for LT-2-3-112A/B, Fuel Zone Level Transmitters	Revision 0
CA-01-154	Allowable Leakage Rate for the HPCI Minimum Flow Control Valve Accumulator System	Revision 2
CA-02-148	Evaluation of MO-2078 Using BWROG DC Motor Performance Method	Revision 2
CA-02-150	Evaluation of MO-2068 Using BWROG DC Motor Performance Method	Revision 1

Calculations

Number	Title	Revision
CA-02-192	Monticello Battery 125 Volt Div II Calculation	Revision 0
CA-03-007	AOV Component Calculation, CV-2065, HPCI Pump Minimum Flow Valve	Revision 3
CA-03-008	AOV Component Calculation, AO-4539 and AO-4540	Revision 1
CA-03-012	AOV Component Calculation, CV-2791	Revision 1
CA-03-015	AOV System Calculation, HPCI, CV-2065	Revision 0
CA-03-097	HPCI/RCIC Suction Head Height Difference	Revision 0
CA-04-047	Monticello 250VDC Division II Battery Calculation	Revision 0
CA-04-048	Monticello 250VDC Division I Battery Calculation	Revision 0
CA-04-166, add. 0	12 EDG-ESW Heat Exchanger Performance Test - 2005	Revision 1
CA-04-183	Determination of EDG Heat Load	Revision 0
CA-04-184	Determination of EDG Heat Load	Revision 0
CA-04-230	HPCI/RCIC Low Pump Suction Pressure Switch Setpoint Spurious Trip Avoidance	Revision 0,1
CA-05-128	No. 13 and No. 16 Battery Charger Sizing	Revision 0
CA-05-124	Hydrogen Generation of No. 13 and No. 16 Battery Rooms	Revision 0
II.SPA.95.001	MAAP Simulations of Monticello Station Blackouts ar Various Initial Power Levels	Revision 0
MN06-995-13-100	Determination of Administrative Building Battery Room Temperature Profiles	Revision 0
PRA-CALC-04-039	RCIC Fault Tree	Revision 0
PRA-CALC-04-040	Aux AC Power Fault Tree Revision	Revision 0
PRA-CALC-04-042	FPS Fault Tree Revision	Revision 0
PRA-CALC-04-044	Containment Vent Fault Tree	Revision 0
	Coordination Study FBS-0603-1, FBS-0604-1, FBS-0605-1, FBS-0606-1, FBS-0607-1, FBS-0608-1, and FBS-0609-1	Revision 0

Corrective Action Documents Generated Due to the Inspection

Number	Title	Date
CAP 01029145	P-105 Drain Line Corroded and Insulation Cracked	5/10/06
CAP 01029281	Underfill of EDG CLR WTR Piping Welds	5/10/06
CAP 01029293	Battery Re-torque Values Potentially 5 percent Less Than OEM Spec	5/10/06
CAP 01029334	RCIC Torus Suction Valve May Not Meet 0737 Commitment	5/10/06
CAP 01029367	0255-06-ID-3 Basis References Incorrect Value from CA-01-154	5/10/06
CAP 01029386	Lack of Full Thread Engagement on 11 EDG Ventilation Fan Grounding Strap	5/10/06
CAP 01029680	Drawing Error on NF-36298-1	5/12/06
CAP 01029702	No 50.59 for Bypass of HPCI/RCIC Auto Initiation in SBO	5/12/06
CAP 01031592	Doc. Lacking for Basis of HPCI Min. Flow Accumulator Sizing	5/22/06
CAP 01032035	Short Circuit with EDG in Parallel with Grid	5/24/06
CAP 01032057	EDG Loading Correction for Higher Frequency Operation	5/24/06
CAP 01032096	Cable Temp. in AC Short Circuit Calculations None Conservative	5/24/06
CAP 01032296	Untimely Entry of PM Schedule Change Forms	5/25/06
CAP 01032362	Several Loads Do Not Meet the Criteria for Cable Voltage Drop to Motor Terminal (less than 2.5 percent)	5/25/06
CAP 01032416	Typo Discovered in CA-05-128, Revision 0	5/26/06
CAP 01032442	Excessive Time to Complete Reviews for Champs WO 030652	5/26/06
CAP 01032514	Discrepancy Between CA–96-200 and Re-rate Analysis	5/26/06
CAP 01032548	Molded Case Ckt Brkrs in C-91/92 Do not Receive Formal PMs	5/26/06
CAP 01033872	Margin Issue with 250 VDC Breaker Interrupting Rating	6/5/06
CAP 01034522	Incorrect Use of NPSH in CA-04-230 for HPCI/RCIC Suct. PS	6/7/06
CAP 01034529	Discrepancies in Tech Manual NX-9216-7	6/7/06
CAP 01034531	CA-90-023 Uses a Slightly Non-conservative Assumption	6/7/06
CAP 01034616	C.4-B.09.02.A Procedure Enhancements	6/8/06
CAP 01034763	Page in EDG Tech Manual Is in Wrong Spot	6/9/06
CAP 01034836	B.04.01-05, Procedure H.2, Rev. 22 Procedure Enhancements	6/9/06

Corrective Action Documents Generated Due to the Inspection

Number	Title	Date
CAP 01034849	Operation of RCIC w/o Electrical Power Procedure Enhancements	6/9/06
CAP 01036344	IST Form May Mislead Operator Evaluation of Operability	6/20/06
CAP 01036387	Tracking Items Not Created to Update Calculation Assumptions	6/20/06
CAP 01036407	Inadequate USAR Update for SBO Strategy Implementation	6/21/06
CAP 01036471	NRC Commitment for Alternate N2 System May Not Be Met	6/21/06
CAP 01036494	Could Fire Pump Runout Occur for RPV Injection	6/21/06
CAP 01036823	No JPM for Procedure 8900, Operation of RCIC w/o Elec Power	6/23/06
GAR 01036173	Issues with C.4-C, Rev. 27, Shutdown Outside Control Room	6/20/06
PCR 01035873	Update Procedure 8153 (Alternate Power for 250 V Bat Charger)	6/16/06
PCR 01036054	Reactor Vessel Pressure/Level Control, Procedure Enhancements	6/19/06
PCR 01036201	Revise C.5-3502, Rev. 11, Containment Spray	6/20/06
PCR 01036214	Revise A.3-003, Rev. 8, Operation of Fire Fighting Equipment	6/20/06
PCR 01036267	Revise B.08.04.03-05, Procedure Enhancements	6/20/06
WO 7831	Provide cover pass for 11 EDG welds	5/10/06
WO 7835	Provide cover pass for 12 EDG welds	5/10/06
WR 7797	Repair P-105 drain line corrosion and cracked insulation	5/10/06

Corrective Action Documents Reviewed During the Inspection

Number	Title	Date
AR 01016010	SRV Accum Calc Not Revised to Reflect Change in Inputs	2/23/06
AR 01016146	Calculation CA-94-017 Has Questionable Results	2/24/06
AR 01016191	Review Operator Action Times for Training Needs	2/24/06
CAP 0002976	HPCI-32 Closed Safety Related Function Not Tested	3/5/01
CAP 0004283	Documented Bases for HPCI Minimum Flow Setpoint	9/6/01
CAP 0004358	Basis for RCIC Cooling Water Flow Requirement	9/11/01
CAP 0004385	Air Supply to RCIC Minimum Flow Valve	9/13/01
CAP 0004397	No Basis for RCIC Low Suction Pressure Trip	9/13/01

Corrective Action Documents Reviewed During the Inspection

Number	Title	Date
CAP 0004473	Basis for Sizing RCIC Minimum Flow Valve Accumulator	9/11/01
CAP 0004474	The Basis for RCIC Pressure Losses in Supply	9/12/01
CAP 0005602	CV-2104, RCIC Min Flow Valve, Actuator Air Supply Pressure	11/7/01
CAP 0007871	Applying TS CST Transfer Level to Local Max Tank Bottom Elev. Results in Non-conservative Value in 0420/CML	3/6/02
CAP 0009064	Design Pressure for RCIC/HPCI Cooling Supply Lines	5/23/02
CAP 0026326	CST Level Setpoint Drawdown Calc	3/31/03
CAP 0034945	Calculations Do Not Reference Peak Div II Battery Room Temp	9/22/04
CAP 0034971	Possible Mis-Coordination for D312 250 VDC MCC	9/23/04
CAP 0034937	Accident Room Temp. 250 V Batteries/Chargers not Determined	9/27/04
CAP 0035245	Elevated Particle Counts and Iron in HPCI Booster Pump Oil Sample	10/13/04
CAP 0035279	PS-13-67 Inst Tubing Config has Water Column Above Switch with no Correction	10/15/04
CAP 0035380	Disch Line Void with HPCI Suction from Torus not Addressed	10/22/04
CAP 0038580	CV-2065 Failed PMT Stroke Time Test	4/8/05
CAP 0038737	Elevated Particle Counts in RCIC Main Reservoir Oil Sample	4/18/05
CAP 0038800	0255-06-IA-1 IST Requirements for Pump Testing Not Meet	4/25/05
CAP 0038886	250 VDC Battery Charger Exceeded Recommended Service Time	5/3/05
CAP 0038969	Unplanned LCO for RCIC during the Performance of 0255-08-IA-1	5/9/05
CAP 0039140	RCIC Power Supply ES-13-92 Has Exceeded Recommended Life	5/20/05
CAP 0039612	ESW Pump Operation in Parallel with SW Creates Potential to Degrade ESW Pump	6/22/05
CAP 01003462	Station Screening Team Review - External OE 11/18/05	11/11/05
CAP 01015143	Site Relay Setting Calculations Not Maintained up to Date	2/17/06
CAP 01015656	CA-92-220 Rev.1 Utilizes Unverified Assumptions	2/22/06
CAP 01016010	SRV Accum Calc Not Revised to Reflect Change in Inputs	2/23/06
CAP 01028115	EDG Fuel Oil Spec Does Not Control Heating Value of Fuel	5/4/06

Corrective Action Documents Reviewed During the Inspection

Number	Title	Date
CR 01000985	Ten Minute Torus Cooling Assumption for Design Basis Containment Analysis Not Analyzed for Operator Actions	2/19/01
OE 652965	Evaluation of RIS 2001-15	8/3/01
OE 021373	HPCI Testing Caused a RCIC Trip	6/16/04
OE 021914	RCIC Governor Resistor Found Failed Open During PM Cal.	8/25/04
OE 022463	HPCI Testing Caused a RCIC Trip (Update)	10/8/04
OE 025352	Engine Systems, Inc. Report No. 10CFR21-0089, Rev 0, Woodward Governor “Compensating” EG Series Actuators	6/24/05
OTH 022684	250 V Battery Room Temperature Evaluation	12/13/05

Design Changes/Modifications

Number	Title	Revision/Date
81-Z-055	RCIC Auto Suction Switchover	4/6/83
91-Z-094	Remove Service Water Support SW-13	4/21/92
92-Q-290	Flood Protection for the Lower 4KV Switchgear Room	Rev. 0
92-Q-615	HPCI/RCIC Minimum Flow Logic Changes	2/4/93
93Q415	Emergency Diesel Generator Electrical Upgrades	6/19/97

Drawings

Number	Title	Revision
NE-36347-16	No.144-480V MCC	Revision J
NE-36399-4B	2R Transformer Sec ACB 152-301 Control	Revision H
NE-36399-6	1R Transformer Sec ACB 152-302 Control	Revision L
NE-36399-9B	Essential Bus Transfer Circuits - Div. II	Revision B
NE-36402-2A	No.104 - Load Center Primary ACB No.152-609	Revision L
NE-36402-3A	No. 104 Load Center Trans. Secondary ACB 52-401 Control Scheme B401	Revision J
NE-36402-3F	LC-B4 Schematic Diagram B404 and B408 - Feeder MCC’s 143A and 144	Revision A
NE-36403-2A	Standby Diesel Generator ACB 152-602 Control	Revision J

Drawings

Number	Title	Revision
NE-36442-2	Generator Lockout Relay and Auto Transfer	Revision U
NE-36839-11	Alternate SRV N2 Supply Schematic Diagram	Revision D
NF-36178	Single Line Meter and Relay Diagram 480 V System	Revision 076
NF-36298-1	Electrical Load Flow One Line Diagram	Revision T
NF-36298-2	DC Electrical Load Distribution One Line Diagram	Revision F
NF-36397	4160 V System Buses No.11, No.12, No.13, No.14, No.15,No.16 Schematic - Meter and Relay Diagram	Revision Y
NF-120106-1	Connection Diagram Hard Pipe Vent System	Revision A
NH-116629	Hard Pipe Vent System	Revision F
NH-36049-10	Alternate Nitrogen Supply System	Revision P
NH-36049-12	Instrument Air Reactor Building and Drywell	Revision T
NH-36246	Residual Heat Removal System	Revision BN
NH-36247	Residual Heat Removal System	Revision BR
NH-36249	High Pressure Coolant Injection System (Steam Side)	Revision 076
NH-36249-1	HPCI Hydraulic Control and Lubrication System	Revision E
NH-36250	High Pressure Coolant Injection System (Water Side)	Revision AF
NH-36251	RCIC (Steam Side)	Revision AS
NH-36252	RCIC (Water Side)	Revision AF
NH-36258	Primary Containment and Atmospheric Control System	Revision BB
NH-36664	RHR Service Water and Emergency Service Water Systems	Revision BR
NX-7822-22-2	RCIC System	Revision AE
NX-7822-22-4	Elementary Diagram - RCIC System	Revision S
NX-7822-22-5B	RCIC Steam Supply Line Isolation MO-2076 Scheme D31104	Revision D
NX-7822-22-6	RCIC System	Revision U
NX-8292-12-1	HPCI System Shutdown	Revision Z
NX-8292-12-2	HPCI System Shutdown	Revision U
NX-8292-12-5	Elementary Diagram HPCI System	Revision T

Drawings

Number	Title	Revision
NX-8292-12-6	Shutdown HPCI System	Revision Y
NX-55883-2	HPCI Pump Head Curves	Revision A
NX-55883-3	HPCI Pump Head Curves	Revision A

Miscellaneous Documents

Number	Title	Revision/Date
Bingham Pumps Letter	IEB 88-04 - Minimum Pump Flow Concern	11/8/88
BW/IP Letter	HPCI Pumps, Minimum Flow Evaluation	6/9/93
E-87CT01	Bus Transfer Study	01/06/98
FP-E-CAP-01	Electrolytic Capacitor Aging Management	4/14/05
GE Letter	Monticello Air Failure Study Recommendations	6/14/71
GE Letter	HPCI System Operation at Elevated Suppression Pool Temperatures	8/7/96
GE Letter	Amendment to Flowserve Report Entitled "Overspeed Evaluation on HPCI Pumps" dated November 30, 2000	12/13/00
GE-NE-L12-00832-1	10CFR50 Appendix R Compliance for Fuel Cladding, Reactor Vessel, and Containment Integrity	9/96
II.SMR.02.008	Human Error Probabilities	Revision 2
JPM No. C.4-C-001	Shutdown Outside Control Room	Revision 7
JPM No. C.4-C-002	Transfer Plant Control to ASDS Panel	Revision 1
JPM No. E.4-01-001	Backfeed Bus 13 from 13 Diesel Generator	Revision 3
JPM No. E.4-03-001	Restore Essential Load Centers from 13 Diesel Generator	Revision 2
Letter: NRC to MNGP	Emergency Procedures and Training for Station Blackout Events	2/25/81
Letter: MNGP to NRC	Response to Generic Letter 81-04 - Assessment of Procedures and Training Programs for Station Blackout	6/8/81
Letter: MNGP to NRC	Responding to NRC Letter dated 2/25/81 - Emergency Procedures and Training for Station Blackout Events	6/8/81
Letter: NRC to MNGP	Procedures and Training for Station Blackout	8/20/81

Miscellaneous Documents

Number	Title	Revision/Date
Letter: NRC to MNGP	NRC Acceptance of NSP Implementation Schedule for Station Blackout Events	12/11/81
Letter: MNGP to NRC	Information Related to NUREG-0737, Items II.K.3.22, II.K.3.24, and II.K.3.28	12/31/81
Letter: NRC to MNGP	NUREG-0737, Item II.K.3.22, "Automatic Switchover of Reactor Core Isolation Cooling System Suction"	8/5/82
Letter: MNGP to NRC	Automatic Suction Switchover of Reactor Core Isolation Cooling System	9/8/82
Letter: MNGP to NRC	Loss of All Alternating Current Power Information Required by 10 CFR Part 50, Section 50.63 (c)(1)	4/17/89
Letter: NRC to MNGP	Safety Evaluation Station Blackout Evaluation	8/22/91
Letter: MNGP to NRC	Response to NRC Recommendations Contained in Monticello Station Blackout Evaluation	11/22/91
Letter: NRC to MNGP	Supplemental Safety Evaluation - Station Blackout Rule	8/5/92
MPS 49	Monticello Fuel Oil Specification	4/15/86
NX-16953	Vendor Manual for D100 250V DC Distribution Panel	12/8/87
OC Meeting Minutes No. 1888	Operations Committee Meeting Minutes 1888 and Cycle 16 Start-up	3/18/93
SA 01004826	High Risk - Low Margin Component Assessment	4/14/06
Terry Letter	Bearing Lube Oil Temperatures	10/24/72
USAR Change Form 3473	USAR Section 10.2.5.1 and 6.2.4.1 Change, HPCI not RCIC an SBO Mitigator	5/12/00
	MOV Database: MO-2078, MO-2013, and MO-2068	5/6/06
	IST Test Results: MO-2013, MO-2068, and MO-2078	

Procedures

Number	Title	Revision
0036-02	ECCS Automatic Initiation Test, Including Loss of Auxiliary Power	Revision 28
0114	RCIC Pump Flow and Valve Tests with Reactor Pressure # 165 PSIG	Revision 43

Procedures

Number	Title	Revision
0137	Master Local Leak Rate Test	Revision 27
0137-29	LPCI Loop “B” Injection Valves Local Leak Rate Test	Revision 6
0187-02	12 Emergency Diesel Generator Start and Load Test	Revision 56
0188-02	12 Emergency Generator Starting Air Compressor Check	Revision 56
0190 -02	Emergency Diesel Generator Fuel Oil Service and Transfer Pump Check	Revision 56
0255-04-IA-1-2	RHR Loop B Quarterly Pump and Valve Test	Revision 68
0255-06-ID-3	HPCI CV-2065 Air Accumulator Check Valve (AI-611) Leak Rate Test	Revision 11
0255-08-IA-1	RCIC Quarterly Pump and Valve Tests	Revision 63
0302	Safeguard Bus Degraded Voltage Protection-Relay Calibration	Revision 20
1047-03	Operations Reactor Side Checklist Weekly Procedure	Revision 48
2010	Turbine Building East	Revision 42
2010-01	Alternate Nitrogen System Data	Revision 1
2030-A	Hourly Control Room Log	Revision 12
4510-PM	Maintenance of On-Site Batteries and Battery Chargers at Monticello Nuclear Plant	Revision 19
8153	Powering Division II 250 VDC Battery Chargers from No.13 Diesel, Security Diesel or Portable Generator	Revision 2
8285	Non-Identical Fuse Replacement	Revision 5
8900	Operation of RCIC Without Electric Power	Revision 1
4 AWI-02.02.03	Work Procedure Content	Revision 4
4 AWI-04.07.01	Operations Committee	Revision 6
4 AWI-05.01.09	Safety Review Item	Revision 1
4 AWI-04.05.12	Replacement of Failed Fuses	Revision 3
A.3-003	Operation of Fire Fighting Equipment	Revision 8
B.01.01-06	Core Flooding Instruments (Figure 29)	Revision 7
B.02.03-05	Reactor Core Isolation Cooling	Revision 17

Procedures

Number	Title	Revision
B.04.01-05	Alternate N2 Supply for Operating AO-4539 and AO-4540	Revision 22
B.08.04.03-01	Alternate Nitrogen System	Revision 2
B.08.04.03-05	Alternate Nitrogen System	Revision 9
B.08.11-05	Diesel Oil System	Revision 15
B.09.07-02	480 Volt Station Auxiliary	Revision 8
B.09.08-05	Emergency Diesel Generator - System Operation	Revision 24
C.4-B.09.02.A	Abnormal Procedures - Station Blackout	Revision 7, 28
C.4-C	Shutdown Outside Control Room	Revision 27
C.5-1100	RPV Control	Revision 11
C.5.1-1100	RPV Control	Revision 6
C.5-1200	Primary Containment Control	Revision 16
C.5-2007	Failure to Scram	Revision 14
C.5-3203	Use of Alternate Injection Systems for RPV Makeup	Revision 10
C.5-3502	Containment Spray	Revision 10
C.5-3505	Venting Primary Containment	Revision 10
C.6-003-A-46	N2 Lo Pressure SRVs, INBD MSIVs, OTBD T-Rings, HPV	Revision 6
C.6-003-A-48	N2 Lo Pressure SRVs, INBD T-Rings	Revision 6
C.6-008-B-28	Window 8-B-28 for 11 EDG	Revision 2
C.6-9-94-A-13	12 EDG High Temperature Alarm	Revision 3
C.6-9-93-A-13	11 EDG High Temperature Alarm	Revision 3
E.4-01	Backfeed Bus 13 from 13 DG	Revision 2
E.4-03	Restore Essential Load Centers from 13 DG	Revision 0
EWI-08.15.02	Motor Operated Valve Program Engineering Standards	Revision 6
Form 3107	Inservice Test Deviation From Criteria Control Room Supervisor’s Immediate Action	Revision 25
MWI-3-M-2.06	Fuse/Breaker Coordination Study and Electrical Coordination	Revision 5

Surveillances (completed)

Number	Title	Date performed
0108	HPCI Pump and Valve Tests with Reactor Pressure # 165 PSIG	6/2/03, 4/22/05
0137-24	Primary Containment Vent, Hard Pipe Vent, “B” CGCS Discharge Isolation Valve LLRT	3/17/05, 3/28/05
0137-30	Alternate N2 Supply Pressure and Local Leak Rate Test	11/21/01, 3/28/02, 3/12/05, 3/25/05
0187-02	12 Emergency Service Water Pump Quarterly Test	6/14/05
0192	Diesel Fuel Quality Check	3/15/06
0193-02	No. 16 250 VDC Battery Operability Check (Div. II) - Weekly Test	12/8/05, 1/13/06, 4/28/06, 5/09/06
0255-06-IA-1	HPCI Quarterly Pump and Valve Tests	3/15/04, 12/14/04, 4/15/05, 6/14/05, 9/13/05, 3/14/06
0255-08-III-1	RCIC Comprehensive Pump and Valve Tests	2/7/05, 2/8/05, 4/11/05, 4/16/05, 11/9/05, 11/10/05, 11/11/05,5/15/06
0255-17-IA-5	Alternate Nitrogen System Train A Valve Test	4/6/06
0255-17-IA-8A	Alternate Nitrogen System Cold Shutdown Valve Test	3/28/05
0255-17-IA-11	AI-713 and AI-714 Operability Test	4/6/06
0255-17-ID-1	Master Alternate Nitrogen System Tests	4/7/06
0255-17-ID-7	Train A Alternate Nitrogen System AI-705 and AI-706 Leak Test	3/13/05
0255-17-ID-8	Train A Alternate Nitrogen System AI-729 Exercise Test and Regulator PCV-4903 and PCV-4904 Check	3/14/05
0255-17-ID-9	Train A Alternate Nitrogen System Test Between AI-717, AI-719, and AI-731	4/2/05
0255-17-ID-10	Train A Alternate Nitrogen System Test Downstream of AI-731	3/25/05
0255-17-ID-11	Train B Alternate Nitrogen System AI-714 and AI-713 Leak Test	3/16/05
0255-17-ID-12	Train B Alternate Nitrogen System AI-730 Exercise Test and Regulator PCV-4905 and PCV-4906	3/26/05

Surveillances (completed)

Number	Title	Date performed
0255-17-ID-13	Train B Alternate Nitrogen System Between AI-721, AI-723, and AI-732	4/1/05
0255-17-ID-14	Train B Alternate Nitrogen System Test Downstream of AI-732	4/7/05
0255-17-ID-15	SRV RV-2-71G Pneumatic Supply Leakage Test	3/30/05
0255-11-III-6	12 ESW Comprehensive Pump Test	12/19/03
0255-10-IIB-3	Primary Containment Vent, Hard Pipe Vent, “B” CGCS Discharge Pressure Test	3/17/05, 3/28/05
4858-PM	4kV, AMH Magneblast Air Circuit Breaker Maintenance	1/11/00, 4/24/03, 5/1/03
4865-PM	Klockner-Moeller Molded Case Circuit Breaker Maintenance and Test Procedure	8/2/99
8095	Fill Diesel Oil Receiving Tank from Truck	3/24/06
8096	Fuel Transfer from Diesel Oil Receiving Tank to Diesel Oil Storage Tank	4/6/06
WO 0306502	12 ESW Pump Pre-service Test	12/19/03
	MO-2013 Diagnostic Test Results	5/5/03
	MO-2078 Diagnostic Test Results	2/9/04
	MO-2068 Diagnostic Test Results	3/22/05

Work Orders

Number	Title	Date
WO 9600774	Modify Anti-Rotation Device on MO-2078	5/9/96
WO 0105477	Change-out Battery No. 16	5/11/03
WO 0124662	RHRSW-14 Opened and Closed OK	4/25/03
WO 0306502	Replace 12 ESW Pump, P-111B.	9/22/04
WO 1522996 01	0192 Diesel Fuel Oil Quality Check	3/15/06

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
AOV	Air-Operated Valve
ASME	American Society of Mechanical Engineers
CDBI	Component Design Bases Inspection
CAP	Corrective Action Process
CFR	Code of Federal Regulations
DC	Direct Current
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
ESW	Essential Service Water
GE	General Electric
HPCI	High Pressure Coolant Injection
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IST	Inservice Testing
JPM	Job Performance Measure
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MNGP	Monticello Nuclear Generating Plant
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OA	Other Activities
PARS	Publicly Available Records
PRA	Probabilistic Risk Assessment
RCIC	Reactor Core Isolation Cooling
RHRSW	Residual Heat Removal Service Water
RIS	Regulatory Information Summary
RPV	Reactor Pressure Vessel
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
TS	Technical Specifications
USAR	Updated Safety Analysis Report
V	Volt
Vdc	Volt Direct Current