April 23, 2004

Mr. T. Palmisano
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 INTEGRATED INSPECTION REPORT 05000263/2004002

Dear Mr. Palmisano:

On March 31, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Monticello Nuclear Generating Plant. The enclosed integrated inspection report documents the inspection findings which were discussed on April 2, 2004, with Mr. Jack Purkis 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, there were four NRC-identified findings of very low safety significance, of which three involved a violation of NRC requirements. However, because these violations were of very low safety significance and because the issues were entered into the licensee's corrective action program, the NRC is treating these violations as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

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 and its enclosure 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). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by Geoffrey Wright Acting for/

Bruce L. Burgess, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000263/2004002
w/Attachment: Supplemental Information

cc w/encl: J. Cowan, Executive Vice President
and Chief Nuclear Officer
Manager, Regulatory Affairs
J. Rogoff, Vice President, Counsel, and Secretary
Nuclear Asset Manager, Xcel Energy, Inc.
Commissioner, Minnesota Department of Health
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	05000263/2004002
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	January 1 through March 31, 2004
Inspectors:	S. Burton, Senior Resident Inspector R. Orlikowski, Resident Inspector D. McNeil, Reactor Engineer J. Bond, Regional Inspector D. Chyu, Regional Inspector M. Parker, Regional Inspector
Observers:	None
Approved by:	B. L. Burgess, Chief Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000263/2004002; 01/01/2004 - 03/31/2004; Monticello Nuclear Generating Plant; Fire Protection and Operability Evaluations.

This report covers a 3-month period of baseline resident inspection. The inspections were conducted by Region III reactor inspectors and the resident inspectors. 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 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstones: Initiating Events and Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors for a violation of Technical Specification for failing to follow Fire Protection Program procedures which required that changes made to the Fire Protection Program be evaluated for impacts to safe-shutdown capabilities. The Engineering Department failed to evaluate the replacement of two dry chemical fire extinguishers with two pressurized water extinguishers in the intake structure area. The licensee has instituted corrective actions including a formal root cause evaluation to assess this issue.

  This issue was more than minor because an unsuppressed electrical or oil fire could affect both trains of emergency service water. The issue was of very low safety significance because the 20-foot separation between two trains did not contain any combustibles and because the automatic fire suppression system was not affected by the finding. The issue was a Non-Cited Violation of Technical Specification 6.5.A, which requires written procedures covering the Fire Protection Program. (Section 1R05(1))

- Green. Three (3) examples of a finding of very low safety significance were identified by the inspectors for a violation of 10 CFR 50, Appendix B, Corrective Action requirements for failing to take prompt and adequate corrective actions to correct pre-fire strategies. The licensee has instituted corrective actions including a formal root cause evaluation to assess this issue.

  This issue was more than minor because pre-fire strategies are used by the fire brigade to identify additional equipment needed and to determine the fire hazards in the fire zones. Failure to have updated and accurate pre-fire strategies could impair the fire brigade’s ability to promptly and properly respond in the event of a fire. The issue was determined to be of very low safety significance as a result of an SDP evaluation which provided credit for the robustness of the fire protection methodology and the automatic fire suppression system for the fire zone. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action” was identified for failure of the licensee to take prompt actions to correct conditions adverse to quality. (Section 1R05(2))

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance with no associated violation was identified by the NRC inspectors associated with the non-safeguards 13 diesel generator (DG) output breaker. The finding was associated with the failure of the Electrical Maintenance Department to identify and correct a damaged output breaker, resulting in increased plant risk. During a monthly surveillance test in January 2004 the 13 DG output breaker failed to shut. An investigation was performed and no apparent cause of the breaker's failure to shut was identified prior to returning the 13 DG to service. During the February surveillance test, the 13 DG output breaker again failed to shut for monthly testing. Further investigation identified a bent linkage in the breaker, which was the cause of the breaker's failure to shut. The Electrical Maintenance Department repaired the bent linkage and returned the 13 DG to service.

  Since the 13 DG has a cumulative impact over time on the plant's safety due to its contribution to core damage frequency (CDF), the inspectors concluded that the finding was more than minor because this finding would become a more significant safety concern if left uncorrected. This finding was of very low safety significance because there was no design deficiency, no actual loss of safety function, no single train loss of safety function for greater than the Technical Specification allowed outage time, and no risk due to external events. (Section 1R15(1))

Cornerstones: Mitigating System and Barrier Integrity

- Green. A finding of very low safety significance was identified by the Engineering Department, but because the finding required a Phase 2 significance determination, the finding was treated as an NRC-identified finding. The finding was associated with the failure to maintain the qualification of switchgear when non-safety related alarm modules were installed on the Division I and Division II 250 VDC buses without an appropriate interface. The alarm re-flash units were installed without safety-related fuses as the interface between the safety and non-safety components. The licensee instituted corrective actions to install an appropriate interface and review certain past modifications for similarities.

  The issue was more than minor because it directly impacted the design control attributes for both the Mitigating Systems and Barrier Integrity objectives. The results of the SDP process found the issue to be Green after consideration of the robust design of the modification and because the fuses had in the past blown to protect the source and adequately isolated the non-safety equipment from the bus. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control" was issued for failure to maintain the safety qualification of safety-related switchgear. (Section 1R15(2))

<u>**Licensee-Identified Violations**</u>

None.

REPORT DETAILS

Summary of Plant Status

Monticello operated at full power for the entire assessment period except for brief down-power maneuvers to accomplish rod pattern adjustments and to conduct planned surveillance testing activities.

1. REACTOR SAFETY

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

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of trains of risk-significant mitigating systems equipment. The inspectors reviewed equipment alignment to identify any discrepancies that could impact the function of the system and potentially increase risk. Identified equipment alignment problems were verified by the inspectors to be properly resolved. The inspectors selected redundant or backup systems for inspection during times when equipment was of increased importance due to unavailability of the redundant train or other related equipment. Inspection activities included, but were not limited to, a review of the licensee's procedures, verification of equipment alignment, and an observation of material condition, including operating parameters of equipment in-service. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors selected the following equipment trains to verify operability and proper equipment line-up for a total of two samples:

- hard pipe vent system with Division II residual heat removal (RHR) out-of-service for maintenance, during the week ending January 7, 2004; and
- Division I residual heat removal service water (RHRSW) system with Division II RHR out-of-service for maintenance, during the week ending January 7, 2004.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

The inspectors performed a complete walkdown of equipment for one risk significant mitigating system. The inspectors walked down the system to verify mechanical and electrical equipment line-ups, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of past and outstanding work orders (WO) was performed to verify that any deficiencies did not significantly affect the system function. In addition, the inspectors reviewed the condition report (CR) database to verify that any system equipment alignment problems were being identified and appropriately resolved. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors selected the following system to verify operability and proper equipment line-up for a total of one sample:

- reactor core isolation cooling (RCIC), for the weeks ending March 6, 2004, and March 13, 2004.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors walked down risk significant fire areas to assess fire protection requirements. The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events (IPEEE), the potential to impact equipment which could initiate or mitigate a plant transient, or the impact on the plant's ability to respond to a security event. The inspection activities included, but were not limited to, the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, compensatory measures, and barriers to fire propagation. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors selected the following areas for review for a total of six samples:

- Fire Zone 7-A, 125V Division I battery room, during the week ending January 17, 2004;
- Fire Zone 10, administration building, during the weeks ending January 17, 2004 and January 24, 2004;
- Fire Zone 12-A, lower 4 kv bus area (11, 13 and 15), during the week ending January 24, 2004;
- Fire Zone 12-B, hydrogen seal area, during the week ending January 24, 2004;

- Fire Zone 23-A, intake structure pump room, during the week ending March 20, 2004; and
- Fire Zone 13-B, RX feedwater pump and lube oil reservoir area, during the week ending March 20, 2004.

b. Findings

(1) Failure to Properly Evaluate Fire Protection Strategy and Program Changes

Introduction

The inspectors identified a Non-Cited Violation (NCV) of Technical Specifications (TS) having very low safety significance (Green) for failing to follow Fire Protection Program procedures, which require that changes made to the Fire Protection Program be evaluated for impact on safe-shutdown abilities.

Description

While performing a fire protection inspection of the intake structure area (Fire Zone 23-A), the inspectors noted that the area contained two pressurized water extinguishers intended to extinguish small Class A fires. The licensee's pre-fire strategies identified the combustible loads in Fire Zone 23-A as lubricating oil, cable insulation, and the contents of a storage locker for flammables in the area. The combustible loads were not Class A fire hazards.

The National Fire Protection Association (NFPA) Code No. 10, Standards for Portable Extinguishers, which identifies the proper selection of extinguishers by the class of hazards, does not identify pressurized water extinguishers for protection from Class B hazards (oil and flammable liquids). The NFPA Fire Protection Handbook states that "the extinguishers in any one area should correspond to the hazards of that area." The handbook also states that if non-foam water base extinguishers are used on Class B fires "the fire may flare up, spread, or injure the operator." The inspectors determined that the pressurized water extinguishers placed in the intake structure area were not best suited for controlling the fires associated with the fire hazards in the area.

On December 19, 2003, the licensee issued CR 03011892 which assessed an external operating experience (OE) document, titled, "ABC Dry Chemical Fire Extinguishers Incompatible with Chlorine-Based Oxidizers." The OE document advised against the use of dry chemical and Halon fire extinguishers in certain areas, warning that "ammonium based compounds typically found in multipurpose (ABC) dry chemical fire extinguishers can react violently, igniting or exploding, on contact with strong oxidizers such as the chlorine or bromine based chemicals used in circulating water treatment systems." The corrective measure outlined in the OE document consisted of staging water-filled extinguishers in these areas to supplement the existing dry chemical extinguishers. In response to the CR, the licensee removed two dry chemical fire extinguishers from the intake structure area because sodium hypochlorite interfaced with circulating water through polyvinyl chloride (PVC) piping. The licensee replaced the two dry chemical extinguishers with two pressurized water extinguishers. The licensee generated CR 04003245 to acknowledge that a thorough evaluation had not been completed at the time the extinguishers were replaced.

Analysis

The inspectors determined that a performance deficiency existed because the Engineering Department failed to follow Fire Protection Program procedures which required that changes made to the Fire Protection Program be evaluated for impacts to safe-shutdown capabilities. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” issued on April 29, 2002. The finding involved the attribute of protection against external factors (fire) and could have effected the mitigating systems objective of ensuring the availability of systems that respond to initiating events to prevent undesirable consequences, because an unsuppressed electrical or oil fire could affect both trains of emergency service water.

The inspectors completed a significance determination of this issue using IMC 0609, “Significance Determination Process (SDP),” dated April 30, 2002, Appendix F, “Determine Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Finding,” dated February 2, 2001, Scheme 3. As part of the Phase 2 evaluation the inspectors considered the potential impact on equipment located in the affected fire zone. The inspectors determined that there were electrical cabinets which could ignite the intervening cable trays in the overhead and propagate fire to both trains of emergency service water (ESW) system. The inspectors used the electrical fire as the most limiting scenario, with ignition frequencies of 2.4E-3 per reactor year for all of the electrical cabinets in the intake structure as referenced in the licensee’s IPEEE ($\log_{10}$(IF)= -2.62). The 20-foot separation between the redundant trains was not degraded (FB= -2). The automatic fire suppression capability was assumed to be in a normal operating state because no finding was identified within this capability (AS= -1.25). This finding affected the manual effectiveness and was conservatively considered highly degraded (MS=-0.25). Since the exposure time for the degraded condition existed for more than 30 days, the estimated likelihood rating for the postulated fire event was determined to be less than 1E-6 occurrences per reactor year.

A fire in the intervening cable trays could cause direct damage to the cabling for ESW pumps A and B. These pumps are required to support the operation of the emergency diesel generators (EDG). However, in this case, the EDG’s were not needed because a fire in the intake structure would not cause a loss of offsite power. Therefore, two SDP worksheets, Transients and Transients without Power Conversion System, were used to evaluate the finding. Other redundant safe shutdown equipment would remain available to mitigate the consequences of a fire in that area. Based upon the inspectors’ evaluation of the Fire Protection SDP using these inputs, the finding screened as a finding of very low safety significance (Green).

Enforcement

Technical Specification 6.5.A requires written procedures be established, implemented and maintained. Subsection A.1 requires procedures recommended in Regulatory Guide 1.33, Revision 2, February 1978, and Subsection A.2 requires procedures for the Fire Protection Program Implementation. Appendix A of Regulatory Guide 1.33 requires written procedures for the Plant Fire Protection Program. Administrative Work Instruction 4AWI-08.01.00, “Fire Protection Program Plan,” Section 4.11.2 requires that changes be evaluated to meet the conditions of the license which states, in part, that changes shall be evaluated against “the ability to achieve and maintain safe shutdown in the event of a fire” and that “the change will not alter specific features of the NRC approved program.” Contrary to the above, the Engineering Department failed to follow the Fire Protection Program procedures when they changed the class of extinguishers in a safe-shutdown area. Specifically, the Engineering Department failed to properly evaluate the change for adverse effects on the ability to achieve and maintain safe shutdown in the event of a fire. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, as noted below, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000263/2004002-01). The licensee has entered this into their corrective action program as CR 04002930 and CR 04003245. The licensee also initiated CR 04003007, which required a formal root cause investigation for the potential programmatic breakdown emerging within the Fire Protection Program and/or 10 CFR 50, Appendix R areas.

(2) Inadequate Corrective Action Impair Fire Brigade Response Capabilities

Introduction

The inspectors identified a Non-Cited Violation (NCV) having very low safety significance (Green) for three (3) examples where inadequate corrective actions resulted in incorrect pre-fire strategies which could impair the fire brigade’s ability to respond to a fire.

Description

In April 2002, CR 02003229 was generated to address an adverse trend in discrepancies of fire extinguishers, alarm bells, emergency lights, and phones in the pre-fire strategies. The CR identified the cause of the discrepancies as a “lack of attention to detail when fire strategy maps were revised.” The associated corrective action generated was to revise the fire drill procedure to include a step which would direct the fire brigade members to verify the accuracy of the fire strategy maps in the area drilled. Because the Operations Department has reviewed only 8 of 87 fire zones for accuracy during post-drill activities in the past 2 years and because they do not drill in all of the 87 fire zones, the inspectors concluded that the corrective actions would not have been timely or comprehensive. The inspectors concluded that the change made to the fire drill procedure was not an adequate action to verify and correct discrepancies in the pre-fire strategies because the limited number of drills performed and limited areas of the plant in which drills were performed would not be timely and failed to assess many areas. The licensee documented the inadequate corrective action in CR 04003239.

In August 2003, CR 03008727 was generated to capture inspector observations which identified that the pre-fire strategy for the reactor feed pump (RFP) area only provided direction to isolate a combustible hydrogen gas source at an isolation valve located inside the RFP area fire zone. The NFPA Fire Protection Handbook advises that all combustible gas sources should be isolated prior to entering a fire zone to combat a fire, further stating that "no attempt to extinguish pressurized fuel fires should be made unless the source of fuel can be promptly shut off, otherwise the fuel may explode." The proposed corrective action was to add a statement to the pre-fire strategy regarding the isolation of the hydrogen gas from outside the fire zone. The allowed completion time to correct the strategy was 300 days. The inspectors noted that the Engineering Department had not started corrective actions after approximately 220 days and the Engineering Department indicated that the plans were to start fire map evaluations in June, less than 60 days prior to the completion due date. Because the time to complete the action was delayed and assumed to take less than 60 days the inspectors determined that waiting over 220 days to complete the action did not constitute prompt action to correct this condition adverse to quality. In an attachment to CR 04003007, the licensee acknowledged that the action to revise the RFP area pre-fire strategy should have been more timely. Corrective actions to be taken were under evaluation by the licensee.

In its response to an external operating experience documented in CR 03011892, which discussed the toxic gas, ignition, and explosion hazards associated with the use of Halon and dry chemical fire extinguishers near circulating water treatment systems containing chlorine-based chemicals, the licensee took actions to remove dry chemical extinguishers from the intake structure area and created a training document which cautioned the fire brigade members about using dry chemical extinguishers in the intake structure area. The Engineering Department failed to take prompt and adequate actions to correct conditions adverse to quality when they did not properly update the fire strategies and maps to include the toxic gas, ignition, and explosion hazards in the intake structure area when they were discovered. The corrective actions did not include a step to revise the pre-fire strategy maps to reflect the potential toxic hazard and the fire extinguishants best suited for the intake structure area as specified in 10 CFR 50, Appendix R.III.K.

The fire brigade uses pre-fire strategies to identify additional equipment needed, to identify adjacent resources available, and to determine the hazards in the fire zones during a fire. Failure to have updated and accurate pre-fire strategies could impair the fire brigades' ability to promptly and properly respond to a fire. Therefore for each of the above examples, the actions taken to correct the pre-fire strategies were not considered prompt or adequate.

<u>Analysis</u>

The inspectors determined that a performance deficiency existed because higher priority should have been given to implementing prompt and adequate changes to the fire strategies. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," issued on April 29, 2002, because the fire brigade utilizes the pre-fire strategies to assess hazards, locate alternate equipment, and prepare to combat fires. The lack of adequate strategies impacts the brigades' timeliness and mitigating capabilities.

The inspectors completed a significance determination of this issue using IMC 0609, "Significance Determination Process (SDP)," dated April 30, 2002, Appendix F, "Determine Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Finding," dated February 2, 2001, Scheme 1 (RFP Area) and Scheme 3 (intake structure area). Because the protective scheme for the RFP area utilizes a 3-hour fire barrier separation that provides wall-to-wall and floor-to-floor separation and the barrier was not affected by the degradation of the fire brigade effectiveness, the finding screened out as Green. The protective scheme for the fire zone utilized more than 20 feet of combustible-free horizontal separation between the redundant safe-shutdown trains and an automatic fire suppression system as part of its fire protection methodology. However, since the finding could have impacted the fire brigade's effectiveness, the finding screened out of Phase 1 and a Phase 2 evaluation of IMC 0609, Appendix F, Scheme 3 was needed. A Phase 2 analysis was performed. The result of the Phase 2 analysis is documented in the analysis section of 1R05(1).

Enforcement

The licensee's Fire Protection Program, including the Fire Protection Plan and pre-fire strategies, is committed to 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action" which requires that measures be established to promptly identify and correct deficiencies and other conditions adverse to quality. Contrary to the above, the licensee failed to promptly correct NRC-identified deficiencies within its pre-fire strategies as evidenced by the following examples: inadequate corrective actions to address incorrect pre-fire strategies, untimely actions to update the pre-fire strategies to include a combustible gas isolation statement, and inadequate corrective actions to identify additional hazards in the Area fire zone. This finding is a violation of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action." Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, as noted below, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000263/200402-02). The licensee entered this issue into their corrective action program as CR 04003007, which required a formal root cause investigation of the potential programmatic breakdown emerging within the Fire Protection Program and/or 10 CFR 50, Appendix R areas.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Licensed Operator Simulator Exercise

a. Inspection Scope

The inspectors performed a quarterly review of licensed operator requalification training. The inspection assessed the licensee's effectiveness in evaluating the requalification program, ensuring that licensed individuals operate the facility safely and within the conditions of their license, and evaluated licensed operator mastery of high-risk operator actions. The inspection activities included, but were not limited to, a review of high risk activities, emergency plan performance, incorporation of lessons learned, clarity and formality of communications, task prioritization, timeliness of actions, alarm response actions, control board operations, procedural adequacy and implementation, supervisory oversight, group dynamics, interpretations of TSs, simulator fidelity, and licensee critique of performance. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors observed the following requalification activity for a total of one sample:

- a training crew during an evaluated simulator scenario that included a loss of Bus 15, a loss of circulating water and bypass valves coupled with a failure to scram, which resulted in the operators entering applicable abnormal response procedures including emergency operating procedures and the emergency plan, during the week ending March 13, 2004.

b. Findings

No findings of significance were identified.

.2 Written Examination and Operating Test Results

a. Inspection Scope

The inspectors reviewed the pass/fail results of individual written tests, operating tests, and simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee during calender year 2004. This represents one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed systems to assess maintenance effectiveness, including maintenance rule activities, work practices, and common cause issues. Inspection activities included, but were not limited to, the licensee's categorization of specific issues including evaluation of performance criteria, appropriate work practices, identification of common cause errors, extent of condition, and trending of key parameters. Additionally, the inspectors reviewed implementation of the Maintenance Rule (10 CFR 50.65) requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, functional failure determinations associated with reviewed condition reports, and current equipment performance status. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors performed the following maintenance effectiveness reviews for a total of two samples:

- a function-oriented review of the 11 and 12 EDG system because it was designated as risk significant under the Maintenance Rule, during the weeks ending January 31 through February 14, 2004; and
- an issue-oriented review of the RCIC system because it was designated as risk significant under the Maintenance Rule, during the weeks ending February 21, 2004, and February 28, 2004.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed maintenance activities to review risk assessments (RA) and emergent work control. The inspectors verified the performance and adequacy of RA’s, management of resultant risk, entry into the appropriate licensee-established risk bands, and the effective planning and control of emergent work activities. The inspection activities included, but were not limited to, a verification that licensee RA procedures were followed and performed appropriately for routine and emergent maintenance, that the RA’s for the scope of work performed were accurate and complete, that necessary actions were taken to minimize the probability of initiating events, and that activities to ensure that the functionality of mitigating systems and barriers were performed. Reviews also assessed the licensee's evaluation of plant risk, risk management, scheduling, configuration control, and coordination with other scheduled risk significant work for these activities. Additionally, the assessment included an evaluation of external factors, the licensee's control of work activities, and appropriate consideration of baseline and cumulative risk. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors observed maintenance or planning for the following activities or risk significant systems undergoing scheduled or emergent maintenance for a total of three samples:

- investigate and repair 13 diesel generator (DG) lockout, during the weeks ending January 24, 2004, and January 31, 2004;
- failure of control room emergency ventilation system compressor seal, during the week ending February 28, 2004; and
- service water piping corrosion, during the weeks ending February 28, 2004, and March 6, 2004.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors reviewed personnel performance to planned evolutions to review operator performance and the potential for operator contribution to the evolution. The inspectors observed or reviewed records of operator performance during the evolution. Reviews included, but were not limited to, operator logs, pre-job briefings, instrument recorder data, and procedures. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors observed the following evolution for a total of one sample:

- planned back-seating of the high pressure coolant injection (HPCI) inboard steam isolation valve MO-2034 to reduce drywell leakage, during the week ending February 7, 2003.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors performed operability evaluations of degraded or non-conforming systems that potentially impacted mitigating systems or barrier integrity. The inspectors reviewed operability evaluations affecting mitigating systems or barrier integrity to ensure that operability was properly justified and that the component or system remained available. The inspection activities included, but were not limited to, a review of the technical adequacy of the operability evaluations to determine the impact on TS, the significance of the evaluations to ensure that adequate justifications were documented, and that risk was appropriately assessed. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors reviewed the following operability evaluations for a total of four samples:

- operability of HPCI with MO-2063 in the closed position, during the week ending January 21, 2004;
- operability of 250 VDC buses with non-safety fuses, during the weeks of February 14, 2004, through March 30, 2004;
- relay device lockout on the non-safeguards 13 diesel generator output breaker, during the week ending February 28, 2004; and
- drywell containment air monitor alarm and loss of flow, during the weeks ending February 14, 2004, and February 28, 2004.

b. Findings

(1) Introduction

The inspectors identified a finding of very low safety significance (Green) with no associated violation for failure of the Electrical Maintenance Department to identify and correct a damaged output breaker for the non-safeguards 13 DG. Since the 13 DG has a cumulative impact over time on the plant's safety due to its contribution to core damage frequency (CDF), the inspectors concluded that the finding was more than minor because this finding would become a more significant safety concern if left uncorrected.

Description

The 13 DG is a non-TS and non-safety related system. The 13 DG does provide a function to backfeed equipment, including 125 VDC and 250 VDC battery chargers, during a station blackout (SBO) event. Because of this function, the 13 DG was determined to be risk-significant in the Monticello Probabilistic Risk Assessment (PRA) model and, therefore, affects the CDF of the Monticello plant. The 13 DG is also included under the Maintenance Rule of 10 CFR 50.65 based on its impact on risk for a SBO event.

On January 22, 2004, while attempting to synchronize the 13 DG to load center LC-107 during a monthly test, the operator received a relay device lockout on the output breaker and the output breaker failed to shut. The 13 DG was shut down and the Operations Department wrote CR 04000750 to document the issue. The Electrical Maintenance Department performed an investigation which included a walk-down of the 13 DG, interviews with operations personnel, a review of 13 DG related technical manuals and drawings, and discussions with Cummins/Ziegler diesel generator vendor representatives. A work order was written to remove the output breaker from service for inspection and maintenance at a later time. On January 23, 2004, the 13 DG was declared functional with the cause not being identified and the work order to inspect the output breaker still outstanding.

On February 18, 2004, the 13 DG output breaker again failed to shut during synchronization with load center LC-107 during a monthly test. The failure of the output breaker to shut was caused by a relay device lockout of the breaker. The Operations Department wrote CR 04001791 to document the issue. On February 20, 2004, an investigation of the output breaker revealed a damaged interlock bracket that was determined to be the cause of the relay device lockout events on both January 22, and February 18, 2004. The Electrical Maintenance Department repaired the bent interlock bracket and returned the 13 DG to service on February 20, 2004.

Analysis

The inspectors determined that the failure to identify and correct the damaged 13 DG output breaker after the first failure was a performance deficiency warranting further evaluation. The inspectors reviewed this finding using the guidance contained in Appendix B, "Issue Disposition Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." As a result, the inspectors compared this performance deficiency to the minor questions contained in Section 3, "Minor Questions," to Appendix B of IMC 0612. Since the 13 DG has a cumulative impact over time on the plant's safety due to its contribution to CDF, the team concluded that the finding was more than minor because this finding would become a more significant safety concern if left uncorrected.

The inspectors reviewed this finding in accordance with IMC 0609, "Significance Determination Process (SDP)," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the finding affected the mitigation systems cornerstone; however, the finding was not a design or qualification deficiency, did not represent an actual loss of safety function of a system or the loss of safety function of single train TS equipment for greater than the allowed outage time, or the loss of safety function of non-TS equipment, nor was there risk due to external events. Therefore, the finding was considered to be of very low safety significance (Green).

Enforcement

The 13 DG is a non-safeguards, non-TS system and is not required to cope with a station blackout; therefore, no violations of regulatory requirements occurred. This issue was considered to be a finding of very low safety significance (FIN 05000263/2004002-03). The licensee entered the issue into its corrective action program as CR 040001791, "Received 86 lockout on 52-710 [DG is output breaker] while trying to synchronize 13 DG to LC-107 during monthly operability test of 13 DG," on February 19, 2004, and repaired the bent interlock bracket on the 13 DG output breaker.

(2) Introduction

The inspectors identified a Non-Cited Violation (NCV) having very low safety significance (Green) of 10 CFR 50, Appendix B, Criterion III, "Design Control" for failure to maintain the qualification of switchgear when non-safety related under-voltage re-flash alarm modules were installed on the Division I and Division II 250 VDC buses without an appropriate interface. This finding impacted both the barrier integrity and mitigating systems cornerstones.

Description

On February 18, 2004, while performing a review of the under-voltage alarm units for 250 VDC motor control centers (MCC) D311, D312, and D313, the Engineering Department discovered that alarm modules were isolated from the safety-related bus with non-safety related fuses. The inspectors reviewed the equipment supported by the affected MCCs and found that HPCI, RCIC, and containment isolation valves in both divisions, all 10 CFR 50 Appendix B components, were powered from the buses.

Analysis

Because both the mitigating systems cornerstone and the barrier integrity cornerstone were affected, the inspectors recognized that a Phase 2 significance determination would be required and the inspectors could not readily ascertain the significance of the finding. The inspectors consulted Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," which indicated that a licensee-identified finding appearing to have more than a very low safety significance should be treated as an NRC-identified finding, analyzed using the SDP found in IMC 0609, and dispositioned in accordance with the Enforcement Policy.

The inspectors reviewed the finding and determined that a performance deficiency existed because the installed alarm units failed to provide an appropriate safety-related interface between the safety and non-safety systems. The inspectors determined that the issue was more than minor because it directly impacted the design control attributes for both the mitigating systems and barrier integrity objectives. Because both the mitigating systems and barrier integrity cornerstones were affected, the SDP Phase 1 worksheet required a Phase 2 analysis.

The initial Phase 2 risk assessment characterized this finding as potentially risk significant, using the benchmarked site specific Risk-Informed Inspection Notebook. However, a Phase 3 analysis, performed by a senior reactor analyst, determined the issue was a Green finding, after providing additional consideration for robust design and installation of the modification, and because the fuses had in the past blown to protect the source and adequately isolated the non-safety equipment from the bus. Therefore, after assessing the licensee's operability evaluation, the senior reactor analyst confirmed the licensee's conclusion that the qualification deficiency did not result in a loss of function.

The Engineering Department determined that the design of the system was robust and preserved the operability of the equipment for the following reasons: the wiring for the non-safety components was equivalent to safety grade; the wiring was installed in conduit that would maintain the environmental qualifications of the buses; the alarm modules, construction, and installations were of a similar design to the MCC, which preserved the environmental and seismic qualifications; and the non-safety related fuses were Underwriters' Laboratories qualified, providing reasonable expectation for the fuses to appropriately isolate faults on the equipment. Additionally, the fuses had in the past blown to protect the source and adequately isolated the non-safety equipment from the bus.

Enforcement

Criterion III, "Design Control," of 10 CFR 50, Appendix B, states, in part, that design changes "shall be subject to design control measures commensurate with those applied to the original design." Contrary to this requirement, the licensee failed to maintain the qualification of safety-related switchgear when they installed non-safety related alarm modules on the Division I and Division II 250 VDC buses without an appropriate interface. 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 an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000263/2004002-04). The licensee has entered this into their corrective action program as CR 04001787. Completed corrective actions included establishing a WO to install safety-related fusing with acceptable interrupt ratings. Additionally, the licensee initiated actions to review similar past design changes associated with this condition.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors verified that the post-maintenance test procedures and activities were adequate to ensure system operability and functional capability. Activities were selected based upon the structure, system, or component's ability to impact risk. The inspection activities included, but were not limited to, witnessing or reviewing the integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use and compliance, control of temporary modifications or jumpers required for test performance, documentation of test data, system restoration, and evaluation of test data. Also, the inspectors verified that maintenance and post-maintenance testing activities adequately ensured that the equipment met the licensing basis, TS, and USAR design requirements. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors selected the following post-maintenance activities for review for a total of four samples:

- post-maintenance test for replacement of failed average power range monitor (APRM) card, during the week ending January 17, 2004;
- post-maintenance testing of No. 12 EDG engine driven fuel oil pump following realignment of the pump coupling, during the week ending January 31, 2004;
- post-maintenance testing of two control rods after speed adjustments to compensate for potentially stuck open check valve, during the weeks ending February 14, 2004, and February 28, 2004; and
- post-maintenance test for replacement of anticipated transient without SCRAM (ATWS) electrical relay, during the week ending February 28, 2004.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed surveillance testing activities to assess operational readiness and to ensure that risk-significant structures, systems, and components were capable of performing their intended safety function. Activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a system, structure, or component could impose on the unit if the condition were left unresolved. The inspection activities included, but were not limited to, a review for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, TS applicability, impact of testing relative to performance indicator reporting, and evaluation of test data. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors selected the following surveillance testing activities for review for a total of six samples:

- emergency core cooling system (ECCS) high drywell pressure sensor test, during the week ending January 17, 2004;
- technical support center-emergency ventilation system (TSC-EVS) quarterly operability test, during the weeks ending January 17, 2004, through January 31, 2004;
- reactor core isolation cooling (RCIC) quarterly pump and valve tests, during the week ending February 14, 2003;
- anticipated transient without SCRAM (ATWS) - recirculation trip for reactor pressure and level trip unit test and calibration, during the week ending February 28, 2004;
- reactor high pressure scram functional test and instrument calibration, during the week ending March 13, 2004; and
- local power range monitor (LPRM) calibration, during the week ending March 6, 2004.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors selected emergency preparedness exercises that the licensee had scheduled as providing input to the Drill/Exercise Performance Indicator. The inspection activities included, but were not limited to, the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critiques. Observations were compared with the licensee's observations and corrective action program entries. The inspectors verified that there were no discrepancies between observed performance and performance indicator reported statistics. As part of this inspection, the documents in Attachment 1 were utilized to evaluate the potential for an inspection finding.

The inspectors selected the following emergency preparedness activity for review for a total of one sample:

- the inspectors observed a licensed operator weekly examination scenario that was performed on March 8, 2004, in conjunction with licensed operator requalification training. Drill notifications were made with state, county, and local agencies for an alert classification.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Initiating Events

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors' review of performance indicators (PI) used PI guidance and definitions contained in Nuclear Energy Institute (NEI) Document 99-02, Revision 2, "Regulatory Assessment Performance Indicator Guideline," to verify the accuracy of the PI data. The inspectors' review included, but was not limited to, conditions and data from logs, licensee event reports, condition reports, and calculations for each PI specified. As part of the inspection, the documents listed in Appendix 1 were utilized to evaluate the accuracy of PI data.

The following PIs were reviewed for a total of three samples:

- unplanned scrams per 7000 critical hours, for the period of January 2003 through December 2003;
- unplanned scrams with loss of normal heat removal, for the period of January 2003 through December 2003; and
- unplanned power changes per 7000 critical hours, for the period of January 2003 through December 2003.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

For inspections performed and documented in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the licensee's corrective action system as a result of inspectors' observations are included in the list of documents reviewed attached to this report.

b. Findings

A Green finding of low safety significance and associated violation for inadequate corrective actions were identified when the licensee failed to promptly and adequately correct pre-fire strategies. (Section 1R05)

.2 Review of Open Work Orders with an Age Greater than 30 Days

Introduction

Administrative Procedure 4AWI-10.01.01, "Corrective Action Program," states that the work control process, including work orders, is a corrective action process. Additionally, 4AWI-04.05.05, "WO Closeout and Disposition," states that the work order preparer shall review completed work orders to determine if conditions adverse to quality exist. The inspectors noted that the open work order list contained many work orders where the work had been completed yet the final review had not been done in excess of 700 days. This condition raised the concern that issues documented in the work process may exist which contain conditions adverse to quality, yet the condition had not been entered into the corrective action program or corrected.

a. Inspection Scope

The inspectors selected approximately 30 work orders from the list of open work orders related to risk significant systems for review. From their review the inspectors identified and followed-up on six work orders that had technician comments which potentially impacted quality.

b. Issues

The inspectors identified that the technicians had written condition reports for identified problems and annotated such on the work orders. Further, the licensee had developed a procedure change to perform the post-work review when the work was completed. The proposed procedure change appeared to be an effective means to ensure issues were properly evaluated and if appropriate entered into the corrective action program.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. Jack Purkis and other members of licensee management on April 2, 2004. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Licensed Operator Requalification Testing for Calendar Year 2004 and Applicability of NRC Inspection Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)," with Mr. G. Lashinski on March 10, 2004;

4OA7 Licensee-Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
T. Palmisano, Site Vice President
J. Purkis, Plant Manager
R. Baumer, Licensing
G. Bregg, Manager, Quality Services
K. Jepsen, Radiation Protection Manager
D. Neve, Regulatory Affairs Manager
E. Sopkin, Director of Engineering

Nuclear Regulatory Commission
B. Burgess, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000263/2004002-01	NCV	Failure to Follow Fire Protection Program Procedures Which Require that Changes Made to the Fire Protection Program be Evaluated for Impacts to Safe-Shutdown Capabilities (Section 1R05(1))
05000263/2004002-02	NCV	Failure to take Prompt and Adequate Corrective Actions to Correct Pre-Fire Strategies (Section 1R05(2))
05000263/2004002-03	FIN	Failure to Identify and Correct a Damaged 13 DG Output Breaker Results in Increased Plant Risk (Section 1R15(1))
05000263/2004002-04	NCV	Failure to Maintain the Qualification of Safety-Related Switchgear when Non-Safety Related Alarm Modules were Installed on the Division I and Division II 250 VDC Buses Without an Appropriate Interface (Section 1R05(2))

Closed

05000263/2004002-01	NCV	Failure to Follow Fire Protection Program Procedures Which Require that Changes Made to the Fire Protection Program be Evaluated for Impacts to Safe-Shutdown Capabilities (Section 1R05(1))
05000263/2004002-02	NCV	Failure to take Prompt and Adequate Corrective Actions to Correct Pre-Fire Strategies (Section 1R05(2))
05000263/2004002-03	FIN	Failure to Identify and Correct a Damaged 13 DG Output Breaker Results in Increased Plant Risk (Section 1R15(1))

05000263/2004002-04	NCV	Failure to Maintain the Qualification of Safety-Related Switchgear when Non-Safety Related Alarm Modules were Installed on the Division I and Division II 250 VDC Buses Without an Appropriate Interface (Section 1R05(2))

Discussed

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 reports.

1R04 Equipment Alignment

Documents and Procedures:
0137-10-OCD; Primary Containment Local Leak Rate Test RCIC System Exhaust Isolation Valves; Revision 12
2121; Plant Prestart Checklist RCIC System; Revision 13
0255-08-IA-8; RCIC Cold Shutdown Check Valve Test; Revision 20
4120-PM; RCIC System Inspection; Revision 22
2154-13; RCIC System Prestart Valve Checklist; Revision 25

Drawings and Prints:
NH-116629; Hard Pipe Vent System; Revision F
NH-36258; Primary Containment & Atmospheric Control System; Revision AZ
NH-36049-10; Alternate Nitrogen Supply System; Revision A
NH-36665; Service Water and Make-up Intake Structure; Revision CF
NH-36664; RHR Service Water & Emergency Service Water Systems; Revision BK
NH-36251; Reactor Core Isolation Cooling (Steam Side); Revision AQ
NH-36252; Reactor Core Isolation Cooling (Water Side); Revision AD

Updated Safety Analysis Report:
Section 10.4; Residual Heat Removal System Service Water System; Revision 20
Section 5.2; Hard Pipe Vent System; Revision 20
Section 6.2; Emergency Core Cooling System (ECCS); Revision 20

Operations Manual:
B.04.01; Hard Pipe Vent System; Revision 4
B.08.01.03; RHR Service Water System; Revision 24
B.02.03-01; Reactor Core Isolation Cooling Functional and General Description of System; Revision 3
B.02.03-03; Reactor Core Isolation Cooling Description of Equipment; Revision 8

Condition Reports:
03004415; RCIC-10 Failed LLRT Test (RCIC-Acceptable)
03004575; RCIC-16 Did Not Meet Acceptance Criteria During 0255-08-IA-8
03004853; RCIC-10 As Left Test Results Skewed Due To Leakage Past Test Boundary Valve RCIC-72
03005350; Add Contingency Steps To 0137-10 For Determining if RCIC-72 May Be Excessively Contributing to RCIC-10 Measured Leakage
04001414; RCIC Steam Line Noted To Be Slightly Repressurized During Isolation Activities

1R05 Fire Protection

Pre-Fire Fighting Procedures and Strategies:
Fire Strategy 7-A; 125V Division I Battery Room; Revision 3
Fire Strategy A.3-10; Administrative Building
Fire Strategy 12-A; Lower 4 KV Bus Area (11, 13 & 15); Revision 7
Fire Strategy 12-B; Hydrogen Seal Area; Revision 4
Fire Strategy A.3-23-A; Intake Structure Pump Room; Revision 6
Fire Strategy A.3-13-B; Rx Feedwater Pump and Lube Oil Reservoir Area; Revision 6

Documents and Procedures:
0275-01; Fire Barrier Penetration Seal Visual Inspection; Revision 10
0275-02; Fire Barrier Wall, Damper, and Floor Inspection; Revision 20
4AWI-08.01.00; Fire Protection Program Plan
Appendix A To Branch Technical Position APCSB 9.5-1 "Guidelines For Fire Protection For Nuclear Power Plants Docketed Prior to July 1, 1976" (August 23, 1976)
SWB-00373; Comparison of Existing Fire Protection Provision to the Guidelines Contained in Standard Review Plan 9.5.1 (December 10, 1976)
M780002A; Fire Protection Functional Responsibilities, Administrative Controls (May 18, 1978)
NFPA Code No. 10; Portable Fire Extinguishers; 1975 and 2003

Updated Safety Analysis Report:
Section 10.3.1.5; Safe Shutdown System Analysis; Revision 20

Operations Manual:
Operational Quality Assurance Plan, Appendix C; Revision 20

Condition Reports:
02003229; Adverse Trend on the Accuracy of the Fire Strategy Maps (NRC-identified issue)
03011892; OE11088 - Dry ABC Dry Chemical Fire Extinguisher Incompatible (NRC-identified issue)
04002837; Strategy A.3-100, Symbols, Was Missing From Fire Strategy Manual Outside the NRC Resident Office (NRC-identified issue)
04002930; Fire Protection Issues Identified by NRC Resident Concerning Fire Strategies and Extinguishers (NRC-identified issue)
04002938; During NRC Walkdown, Halon Extinguisher Discovered in Intake Structure When Dry Chem & Halon Extinguisher Should Have Been Replaced (NRC-identified issue)
04003007; Recognition, Capture, and Response to Fire Protection and Appendix R Program Issues Has Not Been Consistently Effective (NRC-identified issue)
04003007 Attachment; Evaluation of NRC Fire Protection Concerns (NRC-identified issue)
04003058; Halon Extinguisher Removed From Strategy A.3-23-A Map Figure When Revised from Revision 5 to Revision 6 (NRC-identified issue)
04003234; No Fire Extinguisher in Intake Tunnel (NRC-identified issue)
04003236; Commitment to M76029A to Verify Adequacy of Extinguishers for Plant Areas (NRC-identified issue)

04003237; Clear Process Not Defined for Changing Out Fire Extinguishers (NRC-identified issue)
04003239; Periodic Review of Fire Strategies as Part of Fire Drill Proc Does Not Cover All Strategies in a Timely Manner (NRC-identified issue)
04003242; Revision to Fire Strategies Not Completed in a Timely Manner (NRC-identified issue)
04003243; Several Errors/Omissions Found in Fire Strategies Including Intake Structure and Reactor Feed Pump Area (NRC-identified issue)
04003245; Intake Structure Extinguisher Changed Out Without Proper Evaluation (NRC-identified issue)
04003265; NRC Question Concerning Fire Protection Review For 1980's Mod in Intake Structure (PVC Pipe Install) (NRC-identified issue)

1R11 Licensed Operator Requalification Program

Documents and Procedures:
RQ-SS-68; Simulator Exercise Guide - Lockout Bus 15, Loss of Circulating Water, ATWS, Failure of #2 Turbine Bypass Valve; Revision 0
C.5-1100; RPV Control; Revision 9
C.5-2007; Failure to SCRAM; Revision 12
C.5-3101; Alternate Rod insertion; Revision 3
C.4.B.09.06.C; Loss of Bus 15 or 16; Revision 6

1R12 Maintenance Effectiveness

Documents and Procedures:
4106-01-PM; Emergency Diesel 1 Cycle Maintenance; Revision 10
4106-01-PM; Emergency Diesel 1 Cycle Maintenance; Revision 11; dated October 15, 2003
4106-01-PM; Emergency Diesel 1 Cycle Maintenance; Revision 12
0187-01; 11 Emergency Diesel Generator/11 Emergency Service Water Pump System Tests; Revision 43; dated December 31, 2002
Maintenance Rule Database Entries for #11 and #12 EDGs for January 1, 2002 through January 12, 2004
B.9.8; Monticello Maintenance Rule Program System Basis Document; Diesel Generators; Revision 1
Maintenance Rule Database Entries for RCIC for January 1, 2002 through January 20, 2004
DBD B.02.02.03; Revision C; Design Basis Document: RCIC System
0255-08-IA-1; Revision 56; RCIC Quarterly Pump and Valve Tests
3278; Revision 3; NMC Standard 10 CFR 50.59 Screening Form: Condition
Report 02003642 on MO-2076 Failed to Fully Open During Step 41 of Test 0062 RCIC Steam Line High Area Temperature Test and Calibration

Drawings and Prints:
NH-36251; Revision AQ; RCIC (Steam Side)
NH-36252; Revision AD; RCIC (Water Side)

Updated Safety Analysis Report:
USAR Section I.2; Revision 20; RCIC (Steam) System

Condition Reports:
02012499; 11 EDG Electric Fuel Oil Pump Coupling is 20 to 25% Engaged
02012489; Electric Motor Driven Fuel Oil Pump on 12 EDG Failed During Monthly Test 0187-2
03010596; Coupling for Engine Driven Fuel Pump for #12 EDG Does Not Appear to be Installed Per Good Maintenance Practices
03000984; Implement Procedural Guidance for Performing Inspections and Maintenance on EDG Lovejoy Coupling Pump
02006005; 12 EDG Governor Will Not Lower From Local Control or Remote Control. 7 Day Limiting Condition of Operation (Unplanned) per TS 3.9.B.3.a
02008741; 12 EDG Generator Bearing Vibrations Exceed EMD Acceptance
04001211; NRC Questioned Potential Adverse Trend and Potential Ineffective Corrective Action on EDG Coupling (NRC-identified issue)
02002403; The EDG Backwater Check Valves Have Not Been Inspected and Cleaned Since 1995. Corrective Action Not Completed in a Timely Manner
02004311; Found Small Oil Leak on #12 EDG Governor Housing. Appears Leakage is Coming From the Gasket Area of the Housing.
03005021; 12 EDG Generator Bearing Axial Vibration Level Failed PMT Acceptance Criteria
03012811; A 1½" Y-Strainer is Installed on 11 EDG, #2 Air Start System, All Other Air Start Systems Have 2" Strainers
03002769; Adverse Trend Over Last 4 Quarters for 12 EDG NRC Performance Indicators and Unplanned Unavailable Hours Increasing
02000273; During RCIC Testing, Could Not Lower Flow to Approximately 200 gallons per Minute as Called for in the Procedure. Document Operability.
02003642; MO-2076 Failed to Fully Open During Step 41 of Test 0062, RCIC Steam Line High Area Temperature Test and Calibration
03000396; RCIC [Pump Suction Pressurization During Isolation Activities]
03000463; MO-2096 Exceeded the LST During Performance of 0255-08-IA-1
03000478; RCIC Barometric Condenser Vacuum Pump Did Not Provide Proper Light Indication Upon Initial Start and Subsequent Stop

Work Orders:
0200644; PM 4108-2 (12 Emergency Diesel Generator G-3B)
0311217; Adjust Engine Driven Fuel Oil Pump on 12 EDG
0205812; 12 EDG DC Motor Driven Fuel Pump Problem
0205822; Adjust/Replace 11 EDG Electric Fuel Oil Pump
0205828; Replace 12 EDG Speed Sensing Panel DG2/SSP2

1R13 Maintenance Risk Assessments and Emergent Work Control

Documents and Procedures:
Monticello Maintenance Rule Program; System Basis Document: 480 VAC Station Auxiliary; Revision 2
0465-01; Emergency Treatment Filtration System; Revision 25
1047-02; Operations Control Room Checklist; Revision 85
2194; EFT Daily Log Sheet and Administration Building Checks; Revision 34

Operations Manual:
B.09.15; Non-Essential Diesel Generator

Condition Reports:
04000750; Received 86 Lockout on 52-710 While Trying to Sync 13 DG to LC-107 During 1374 (Monthly Operability Test of 13 ED)
04000267; SW-42-1 Pin-Hole Leak Valve Body. Service Water Return Isolation Valve From V-AC-3A Spraying/Dripping Onto V-AC-3A Motor
04000576; Leak Noted on 6" to 3" Reducer on Mechanical Vacuum Pump Seal Water Cooler
ACC 04001836; Small Leak on Service Water Piping Found in the Torus Room
04000834; Small Leak on Service Water Piping Found in the Torus Room
04000864; Adverse Trend: Unexpected Leaks in Service Water Pipes
04002088; Small Water Leak From Insulation in Vicinity of SE-239-1, SW Supply to 11 Emergency Diesel Generator
04000063; Service Water Pipe to MVP Seal Water Cooler Has a Leak. Does Not Appear to be From a Mechanical Joint
04002142; Unplanned 30 Day CRV LCO Due to V-EAC-14B Shaft Seal Failure Causing Oil/Freon Leak

Work Orders:
0400419; Investigate and Repair 13 DG Lockout
0400835; Large Shaft Seal Leak on V-EAC-14B

1R14 Personnel Performance During Non-Routine Plant Evolutions and Events

Documents and Procedures:
3034; Jumper Bypass Form; Jumper 04-03
B.03.02-05; Operations Manual - HPCI; Revision 21
3427-D; OC Subcommittee D Review Distribution List for HPCI - System Operation Revision 21; Revision 13
3274; Procedure Preparation Checklist for B.03.02-05, Operations Manual - HPCI, Revision 21; Revision 27
5792; Training Needs Checklist for B.03.02-05, Operations Manual - HPCI, Revision 21; Revision 11
Cycle 22 Drywell Leakage Graphs; 07/01/03 to 02-10-04, 01/01/04 to 03/04/04
3108; Pump/Valve/Instrument Record of Corrective Action for MO-2034 Performed 02/05/04; Revision 12

Condition Reports:
04000648; Operability of HPCI with MO-2063 in Closed Position Has Not Been Analyzed but LCO Not Entered for Closed Stroke Timing

1R15 Operability Evaluations

Documents and Procedures:
Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 137 to Facility Operating License No. DRP-22 Nuclear Management Company, LLC Monticello Nuclear Generating Plant Docket No. 50-263

Monticello Maintenance Rule Program System Basis Document B.9.15; Non-Essential Diesel Generator
76M039; Design Change - Direct Current (DC) Motor Starters

Drawings and Prints:
NF-36969-1; #133-250V DC MCC's D313; Revision H
NF-36969-2; #133-250V DC MCC's D313; Revision H
NX-9297-1-6; Elementary Wiring Diagram for DC Motor Starter; Revision D

Technical Specifications:
3.5/4.5; Core and Containment Spray/Cooling Systems
3.6/4.6 and Basis; Primary System Boundary

Updated Safety Analysis Report:
Section 6.2; Emergency Core Cooling System (ECCS)
Section 8.4; Non Safeguards Diesel Generator
Section 8.12; Station Blackout

Condition Reports:
04000648; Operability of HPCI with MO-2063 in Closed Position Has Not Been Analyzed But LCO Not Entered for Close Stroke Timing
04001107; DW Continuous Air Monitor Trouble Alarm and No Flow at CAM Results in DW CAM Declared Inoperable & Unplanned 12 Hour Grab Sampling LCO Entry
04001680; NRC Disagrees With TS Basis Change 137a on Reactor Coolant System Leakage if DW Sumps Overflow (NRC-Identified issue)
04001791; Received 86 Lockout on 52-710 While Trying to Synchronize 13 DG to LC-107 During 1374 (Monthly Operability Test of 13 DG)
04000750; Received 86 Lockout on 52-710 While Trying to Synch 13 DG to LC-107 During 1374 (Monthly Operability Test of 13 DG)
04001867; Potential Degradation of Breaker 52-710 May Not Have Been Considered in CDF Risk Since Previous Failure
04001787; Standard Grade Undervolt Alarm Reflash Units On 250 VDC MCCs Do Not Have Appropriate Isolation From Safety Related Equipment

1R19 Post-Maintenance Testing

Documents and Procedures:
0012; APRM/Rod Block Scram Surveillance Test; Revision 28
0187-02; 12 Emergency Diesel Generator / 12 ESW Quarterly Pump and Valve Tests; Revision 48
1054; Control Rod Drive Normal Drive Timing Test; Revision 12
Design Basis Document; Anticipated Transient Without SCRAM; Revision 2
0278-B; ATWS - Recirculation Trip for Reactor Pressure and Level Trip Unit Test and Calibration; Revision 11

Drawings and Prints:
NX-16162-1-8; ATWS Elementary Diagram; Revision C

Updated Safety Analysis Report:
USAR Section 7.6.2; ATWS System; Revision 20

Operations Manual:
B.05.01.02-02; Power Range Neutron Monitoring Range (PRM): Descriptions of Equipment; Revision 4
B.05.01.02-06; Power Range Neutron Monitoring Range (PRM): Figures; Revision 3

Condition Reports:
04000441; Figure 2 in Procedure 0012 Does Not Match Actual Switch Positions of APRM (NRC-identified issue)
04000904; Discussion With the Resident Inspector Identifies Concern With PMT Adequacy on 12 EDG Engine Drive Fuel Oil Pump, WO 0311217 (NRC-identified issue)
04001380; During Performance of 1054, Two Control Rods Withdrew Faster Than 35 Seconds and Were Fully Inserted Per 1054

Work Orders:
0400281; Average Power Range Neutron Monitor
0311217; Adjust Engine Driven Fuel Oil Pump on 12 EDG
0205812; 12 EDG DC Motor Driven Fuel Pump Problem
0400475; Increase Speed of 2 Control Rods Using 1054
0310751; Replace or Recondition K101C

1R22 Surveillance Testing

Documents and Procedures:
0030; ECCS High Drywell Pressure Sensor Test; Revision 11
CA-94-106; Determination of Drywell Pressure Instrument Setpoints; Revision 0
5790-001-01; Emergency Response Organization; Revision 42
5790-106-02; TSC Staffing and Organization Chart; Revision 7
A.2-213; Responsibilities of the Emergency Director; Revision 11
Letter to Director Office of NRR; Post-TMI Requirements for the Emergency Operations Facility (Response to GL 81-10); December 1, 1981
1432; TSC-EVS Quarterly Operability Test; Revision 11
Generic Letter 2003-01 Control Room Habitability 60 Day Response; August 5, 2003
LER 96-013; Failure to Comply with Tech Spec Requirement to Verify the Control Room Ventilation System Maintains a Positive Pressure with Respect to Adjacent Areas
EP-7; Emergency Plan; Revision 24
0255-08-IA-1; RCIC Quarterly Pump and Valve Tests; Revision 56
0060; RCIC Hi Steam Flow and Low Steam Pressure Sensor Test and Calibration Procedure; Revision 27
0278-B; ATWS - Recirculation Trip for Reactor Pressure and Level Trip Unit Test and Calibration
Calculation CA-95-019; ATWS High Reactor Pressure; Revision 7
Calculation CA-95-023; ATWS Low Low Water Level; Revision 11
0002/0019; Reactor High Pressure Scram Instrument Test and Calibration Procedure; Revision 13
7210; LPRM Calibration; Revision 3
5528; Radiation Protection Survey Record; Revision 14

Technical Specifications:
3.5/4.5 and Bases; RCIC
3/4.2 and Bases; Protective Instrumentation

3/4.5 and Bases; Core and Containment Spray/Cooling Systems
3.2/4.2 and Bases; Protective Instrumentation
3.1/4.1 and Bases; Reactor Protection System

Updated Safety Analysis Report:
Section 10.2.5; Reactor Core Isolation Cooling System (RCIC)
Section 7.6.2; ATWS System; Revision 20

Operations Manual:
B.05.03-02; Area Radiation Monitoring; Revision 1
B.03.03-05; Automated Transversing Incore Probe; Revision 3

Condition Reports:
04000344; TSC-Outside Differential Pressure was Negative During Testing
04000393; TSC-EVS Quarterly Procedure 1432 Most Recent Revision May Have Caused Out of Specification Reading During 1/24/04 Procedure Run
04000523; Instructions for TSC/BOSC EVS Operation in A.2-106 are Not Explicit in Requiring an Initial Differential Pressure Check Upon Activation
00003720; Compensatory Measures for TSC Emergency Ventilation System Failure Appear Inadequate
04001465; Problems with Stop Watch Prevent Obtaining Accurate Time From MO-2078 Opening to 400 gallons per minute. Not a RCIC Problem
04001466; RCIC Pump Outboard Bearing Vibration is Outside of the Trend Range for Quarterly Testing
04001470; Recorder for RCIC Testing Stopped Recording Prior to Being Shut Off by the Operator on Step 50
02001013; Documentation of NRC Resident Question Regarding the Application of Tech Spec Deviations in As-Found Acceptance Criteria

Work Orders:
0311165; TSC-EVS Trouble Alarm During System Start-up
0307635; TSC-EVS Trouble Alarm During System Start-up

1EP6 Drill Evaluation

Documents and Procedures:
RQ-SS-68; Simulator Exercise Guide - Lockout Bus 15, Loss of Circulating Water, ATWS, Failure of #2 TBPV; Revision 0
C.5-1100; RPV Control; Revision 9
C.5-2007; Failure to SCRAM; Revision 12
C.5-3101; Alternate Rod Insertion; Revision 3
C.4.B.09.06.C; Loss of Bus 15 or 16; Revision 6

4OA1 Performance Indicator Verification

Documents and Procedures:
Monticello Performance Indicator Data Summary Reports; 1st Quarter 2003 through 1st Quarter 2004
NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 2

3530-11; NRC Performance Indicator Initiating Events Worksheet (1st Quarter 2003 through 4th Quarter 2003); Revision 3
3530-13; NRC Performance Indicator Unplanned Power Changes per 7,000 Critical Hours Worksheet (1st Quarter 2003 through 4th Quarter 2003); Revision 1
Monticello Thermal Power History Graphs for January 2003 through December 2003
Operator Logs for April 26, 2003, and May 23 through 26, 2003

4OA2 Identification and Resolution of Problems

Documents and Procedures:
Active Work Order; 92 Level Review Needed; February 3, 2004
Active Work Order; 93 Level Review Needed; February 3, 2004

Condition Reports:
0308735; MO-2036 Manual Handwheel Will Not Stay Engaged
0308769; Refurbish Main Stage Assembly for SRV ‘B’
0308220; Reactor Water Leakage Through SBLC
0307514; B3124 Breaker Trip
0205822; Adjust/Replace 11 EDG Electric Fuel Oil Pump
0307980; Verify Proper Torque on P-202D Pump to Floor Bolts
04003058; Halon Extinguisher Removed From Strategy A.3-23-A Map Figure When Revised From Revision 5 to Revision 6 (NRC-identified issue)
04002938; During NRC Walk Down, Halon Extinguisher Discovered in Intake Structure Area When Dry Chemical and Halon Extinguisher Should Have Been Replaced (NRC-identified issue)
04002617; SRI Question Regarding Work Package Temporary Changes That May Warrant a Condition Report (NRC-identified issue)
04002453; This CR Written to Document a Question Raised By MNGP Resident Inspector on Conduct of SEC Continuing Training (NRC-identified issue)
04002378; NRC Resident Inspector Questioned if Radiological Postings are Needed for the TIP Drive Room During a Full TIP Core Scan (NRC-identified issue)
04002124; Inadequate Documentation of no Decrease in Effectiveness Determination for EAL Change Made in Revision 22 of E-Plan (NRC-identified issue)
04002036; NRC Identified Concern With Operator Use of Operations B Manual Section 1 as a Controlled Document That Could Effect Exams (NRC-identified issue)
04001680; NRC Disagrees with TS Bases Change 137A on RCS Leakage if DW Sumps Overflow (NRC-identified issue)
04001211; NRC Questioned Potential Adverse Trend and Potential Ineffective Corrective Action on EDG Coupling (NRC-identified issue)
04000842; Question Presented by Resident NRC About Sump Cover Securing Issues (NRC-identified issue)
04000915; NRC Notification Did Not Disclose That Conditions Had Been Corrected for Inoperable EFT trains (NRC-identified issue)
04000932; Evaluation of Rototorque MOV Switch Setting (11/11/03 Discovered re: MO-2107) Did Not Identify 6 Hour Period of LCO Entry (NRC-identified issue)
04000940; ADVERSE TREND - A Number of CR’s Related to Configuration Management Have Been Self-Identified (NRC-identified issue)
04000941; ADVERSE TREND - Recent Incidence of PMT’s That Appear Not to Bound Scope of Work or Sufficiently Quantify Function (NRC-identified issue)

04000942; Content and Basis for TSC-EVS Surveillance Does Not Appear to Support Design Commitments (NRC-identified issue)
04000873; NRC Resident Questioned Use of Ops Manual - 01 Section as Controlled Reference Material in Accordance With NUREG-1021 ES-602 (NRC-identified issue)
04000904; Discussion With Resident Inspector Identifies Concern With PMT Adequacy on 12 EDG Engine Driven Fuel Pump, WO 0311217 (NRC-identified issue)
04000699; Scaffolding Parts/Equipment Stored Under the Torus Stacked Above the Holding Brackets (NRC-identified issue)
04000642; Document NRC Questions Regarding Reportability for TSC-EVS Failure to Pressurize All Adjacent Spaces on 01/12/04 (NRC-identified issue)
04000560; During 4th Quarter NRC Exit, NRC Asked Question Concerning Using CCDP in Risk Assessment - Follow up With NRC (NRC-identified issue)
04000441; Figure 2 in Procedure 0012 Does Not Match Switch Position of APRM (NRC-identified issue)
04000263; Document NRC Question Regarding Retirement of Procedure 1447 and Unit Heater PM Checks (NRC-identified issue)

LIST OF ACRONYMS USED

APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AWI	Administrative Work Instruction
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
CRV	Control Room Ventilation
CY	Calendar Year
DBD	Design Basis Document
DG	Diesel Generator
DRP	Division of Reactor Projects
DW	Drywell
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
EP	Emergency Plan
ESW	Emergency Service Water
EVS	Emergency Ventilation System
FIN	Finding
HPCI	High Pressure Core Injection
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report
kV	Kilovolt
LC	Load Center
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLRT	Local Leak Rate Testing
LPRM	Local Power Range Monitor
MCC	Motor Control Centers
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
OE	Operating Experience
PI	Performance Indicator
PM	Planned or Preventative Maintenance
PMT	Post-Maintenance Testing
PRA	Probabilistic Risk Assessment
PVC	Polyvinyl Chloride
RA	Risk Assessment
RCIC	Reactor Core Isolation Cooling
RFP	Reactor Feed Pump
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water

LIST OF ACRONYMS USED

RP	Radiation Protection
RWP	Radiation Work Permit
Rx	Reactor
SBLC	Standby Liquid Control
SBO	Station Blackout
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SRV	Safety Relief Valve
TS	Technical Specification
TSC	Technical Support Center
UL	Underwriters' Laboratories
USAR	Updated Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current
WO	Work Order