April 29, 2005

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 05000263/2005002

Dear Mr. Palmisano:

On March 31, 2005, 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 7, 2005, with you and other members of your staff.

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

Based on the results of this inspection, there were two findings of very low safety significance (Green) identified; one NRC-identified and one self-revealed finding. Both were determined to involve violations of NRC requirements. However, because these violations were of very low safety significance and because these issues were entered into your corrective action program, the NRC is treating these findings 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/

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

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

Enclosure: Inspection Report 05000263/2005002
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
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:	05000263/2005002
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	January 1 through March 31, 2005
Inspectors:	S. Burton, Senior Resident Inspector R. Orlikowski, Resident Inspector R. Langstaff, Project Engineer D. Melendez, Reactor Engineer M. Mitchell, Radiation Specialist R. Winter, Regional Inspector D. McNeil, Reactor Inspector (Lead Inspector) R. Walton, Reactor Inspector T. Bilik; Reactor Inspector
Observers:	None
Approved by:	B. Burgess, Chief Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000263/2005002; 01/01/2005 - 03/31/2005; Monticello Nuclear Generating Plant. Maintenance Effectiveness.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections of occupational radiation safety, licensed operator requalification program, and inservice 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

Cornerstone: Mitigating Systems

- Green. The inspector identified a finding of very low safety significance involving a failure to follow a procedure, in that the adequacy of illumination was not verified by an examiner for a visual exam being performed on a residual heat removal (RHR) heat exchanger support.

  This finding was greater than minor because the issue involved procedural errors being performed by more than one examiner, involved more than one type of examination, and extended to other systems and components. Specifically, the licensee's subsequent extent of condition (EOC) evaluation (Condition Evaluation CE012073) determined that two examiners had performed visual examinations and system pressure tests without the use of illumination checks as required by procedure and American Society of Mechanical Engineers (ASME) Code. This resulted in numerous inadequate examinations being performed, including those which involved mitigating systems (MS) and primary containment (PC). As a result of the EOC evaluation, the licensee was required to re-perform approximately 60 exams/tests (VT-1, VT-3, pressure tests, or other periodic tests). Because the examinations were re-performed (or relief requested to allow acceptance of several non-repeatable tests) to demonstrate code compliance without revealing any degradation, this issue was considered a finding of very low safety significance. This finding was a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, which required activities to be accomplished in accordance with procedures and 10 CFR 50.55a(g)4, which requires, in part, that components (including supports) must meet the requirements set forth in the ASME Code Section XI. (Section 1R08)

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance was self-revealed for a violation of Technical Specifications for maintenance personnel failing to perform maintenance in accordance with written procedures associated with air-operated valve AO-2381, the drywell purge inboard isolation valve. In February 2005, AO-2381 was declared

inoperable after it was determined that the valve's as-found seating force exceeded that allowed by calculational limits and the valve may not be able to close under a design basis accident condition. During a review of the maintenance history for AO-2381 it was discovered that, in February 2000, maintenance workers failed to complete a step in the procedure used to replace the T-ring seal of this valve. The cause of the failure of this valve was due to interference of the valve disc with the T-ring seat. The primary cause of this finding was related to the cross-cutting area of Human Performance. The licensee replaced the T-ring seat during the March 2005 refuel outage and the valve was declared operable after post-maintenance testing.

The issue affected the Barrier Integrity cornerstone attribute of maintaining the functionality of containment. Specifically, this issue affected the containment isolation system, structure, and component (SSC) reliability/availability element of the SSC and Barrier Performance attribute and, therefore, was determined to be more than minor. This finding was of very low safety significance because there was no degradation of the radiological barrier function provided for the control room, auxiliary building, spent fuel pool, or standby gas treatment system; no degradation of the smoke or toxic gas barrier function of the control room; and the finding did not represent an actual open pathway in the physical integrity of the reactor containment or involve an actual reduction in defense-in-depth for the atmospheric pressure control or hydrogen control functions of the primary containment. The issue was a Non-Cited Violation of Technical Specification 6.5.A, which requires that maintenance that can affect the performance of safety-related equipment should be properly performed in accordance with written procedures, documented instructions, or drawings appropriate for the circumstances. (Section 1R12)

B. <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 with the following exception:

- On January 27, 2005, a fuel cycle coastdown began, followed by a shutdown for a planned refueling outage on March 5, 2005, that continued through the remainder of the period.

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 the attachment were utilized to evaluate the potential for an inspection finding.

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

- control room ventilation train "B" with 13 emergency service water system out-of-service for maintenance;
- No. 12 emergency diesel generator (EDG) during planned maintenance on No. 11 EDG; and
- Division II shutdown cooling with Division I shutdown cooling out-of-service for maintenance.

b. Findings

No findings of significance were identified.

.2 <u>Complete System Walkdown</u>

a. <u>Inspection Scope</u>

The inspectors performed a complete walkdown of equipment for one risk significant mitigating system. The inspectors walked down the system to review 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 (WOs) was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program (CAP) database to ensure that any system equipment alignment problems were being identified and appropriately resolved. As part of this inspection, the documents in the attachment were utilized to evaluate the potential for an inspection finding.

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

- standby liquid control (SBLC) system.

b. <u>Findings</u>

No findings of significance were identified.

1R05 <u>Fire Protection</u> (71111.05)

.1 <u>Quarterly Fire Zone Walkdowns</u> (71111.05Q)

a. <u>Inspection Scope</u>

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), or the potential to impact equipment which could initiate or mitigate a plant transient. 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 the attachment were utilized to evaluate the potential for an inspection finding.

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

- Fire Zone 1-F, suppression pool area;
- Fire Zone 2-B, east hydraulic control unit area;
- Fire Zone 2-C, west hydraulic control unit area;
- Fire Zone 3-B, motor control center and SBLC area;
- Fire Zone 3-C, vessel instrument rack area;
- Fire Zone 3-D, reactor building closed cooling water pump area;
- Fire Zone 3-E, reactor building 962' elevation north;
- Fire Zone 12-C, condenser area; and
- Fire Zone 30, turbine deck.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

From March 14, 2005 to March 18, 2005, the inspector conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and the risk significant piping system boundaries. The inspector selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of IP 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the onsite inspection period.

The inspector conducted an on-site review of the following types of nondestructive examination activities to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements. Specifically, the inspector observed the following examinations:

- Ultrasonic examination (UT) of a vessel shell-to-nozzle weld (weld N-4C, feedwater "B" inlet);

- Magnetic particle examination (MT) of a residual heat removal (RHR) heat exchanger outlet nozzle (weld –1); and

- Visual examination (VT-3) of a RHR heat exchanger support (support "C").

The inspector reviewed an examination completed during the previous outage with relevant/recordable condition/indication that was accepted for continued service to verify that the licensee's acceptance was in accordance with the Section XI of the ASME Code. Specifically, the inspector reviewed the following record:

- The inspector reviewed liquid penetrant (PT) records of the sealing surface on the flange for a head spray nozzle (N6A). During this examination, the licensee identified a rounded relevant indication (indication evaluated and found to be acceptable per ASME Code, Section VIII).

The inspector reviewed pressure boundary welds for Code Class 1 or 2 systems which were completed during the previous refueling outage, to verify that the welding acceptance and preservice examinations (e.g., pressure testing, visual, magnetic particle, and weld procedure qualification tensile tests and bend tests) were performed in accordance with the ASME Code Sections III, V, IX, and XI requirements. Specifically, the inspectors reviewed welds associated with the following work activities:

- Main steam line drain valve relocation and line replacement (welds W-13, CLAJ-9 and W-14, CLAJ-10, class 1 component). The welds were fabricated during the relocation of valve MO-2374.

The inspector performed a review of piping system inservice inspection-related problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. Additionally, the inspector's review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspector evaluated the threshold for identifying issues through interviews with licensee staff and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspector performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspector are listed in the attachment to this report.

The licensee identified a leaking control rod insert line and indicated that a licensee event report (LER) would be issued. The inspector will complete the NRC evaluation of this issue during the LER review.

The reviews as discussed above counted as one inspection sample.

b. Findings

Failure to Verify Adequate Illumination Levels for VT-3 Exam

Introduction: The inspector identified a Non-Cited Violation (NCV) of both 10 CFR Part 50, Appendix B, Criterion V, and 10 CFR 50.55a(g)4 having a very low safety significance (Green), related to an inadequate ASME Code required visual examination of a support bracket.

Description: On March 16, 2005, the inspector identified through direct observation that a licensee contract non-destructive examination (NDE) examiner was not performing a VT-3 examination per procedure. The examiner was conducting an examination on the RHR heat exchanger support (support C) but had failed to verify adequate illumination levels to perform the examination.

The licensee had previously been committed to an earlier edition of ASME Code Section XI which did not contain the same illumination verification requirements. The current ASME code committed to (1995 Edition with 1996 addenda) as well as the current revision of the licensee's VT procedure called for the illumination levels from battery powered lights to be checked before and after each examination or series of examinations. The examiner had relied on his memory to perform the examination. The licensee documented this concern in corrective action number CAP037910 ("ISI NDE Examiner Procedural Errors").

The EOC evaluation (CE012073) indicated that two examiners had performed visual examinations and system pressure tests without the use of illumination checks as required by procedure and ASME Code. This resulted in inadequate examinations being performed on numerous structures, systems or components, including those which involved MS and PC. As a result of the EOC evaluation, the licensee re-performed approximately 60 exams/tests (VT-1, VT-3, pressure tests, or other periodic tests) or requested relief to allow acceptance of a number of non-repeatable tests. The licensee also planned to write a VT-2 procedure to be referenced in future pressure test procedures/surveillances to capture illumination check requirements. These actions are captured in the following corrective action documents resulting from this issue; CA024144, CA024145, and CA024146.

Analysis: The inspector determined that the failure of the examiners to perform illumination checks required by procedure and ASME code, warranted a significance determination. The inspector reviewed this finding against the guidance contained in Appendix B, "Issue Dispositioning Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." In particular, the inspector compared this finding to the findings identified in Appendix E, "Examples of Minor Issues," of IMC 0612 to determine whether the finding was minor and concluded that none of the examples listed in Appendix E accurately represented this example. As a result, the inspector compared this performance deficiency to the minor questions contained in Section 3, "Minor Questions," to Appendix B of IMC 0612. The inspector concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the MS Cornerstone attribute and affected the MS objective of ensuring the availability, reliability, and capability of systems that respond to mitigating events to prevent undesirable consequences. The inspector was concerned that the failure to perform adequate VT exams on MS could have allowed undetected cracks or other deficiencies to remain in service. Returning the plant to service with undetected cracks could increase the probability a MS to be unavailable, unreliable, or incapable to perform its function when called upon. The finding was assigned to the MS Cornerstone because, while multiple systems or components were affected, one of the principle components affected was the RHR system and the finding affected the MS Cornerstone objectives.

The inspector next determined that the finding could not be evaluated using the Significance Determination Process (SDP) in accordance with NRC IMC 0609, "Significance Determination Process," because the SDP for the MS Cornerstone only applied to degraded systems/components, not to deficiencies associated with the procedures that are designed to detect component degradation. Therefore, the finding was reviewed by a regional branch chief in accordance with IMC 0612, Section 05.04c, who agreed with the inspector that this finding was of very low safety significance (Green). Specifically, there was no evidence of actual degradation that had been missed.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documents, instructions, procedures, or drawings and shall be accomplished in accordance with these instructions, procedures, or drawings. Procedure PEI-02.05.02, "Visual Examination of Components and Their Supports," Step 9.3.1c, states that, "The illumination levels from battery powered lights SHALL be checked before and after each examination or series of examinations not to exceed 4 hours between checks."

Title 10 CFR 50.55a(g)4 requires, in part, that throughout the service life of a boiling water-cooled nuclear power facility, components (including supports) must meet the requirements set forth in the ASME Code Section XI. Section XI, IWA-2210, "Visual Examinations," requires that illumination levels from battery powered portable lights shall be checked before and after each examination or series of examinations, not to exceed four hours between checks.

Contrary to the above, on March 16, 2005, while performing a VT-3 examination using procedure PEI-02.05.02 on an RHR heat exchanger support (Support "C" of RHR heat exchanger "A"), the licensee examiner failed to check the illumination levels of the battery powered light before or after the VT-3 examination of the support.

Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (corrective action number CAP037910), it is being treated as a NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000263/2005002-03).

1R11 Licensed Operator Requalification Program (71111.11)

.1 Quarterly Review of Licensed Operator Requalification Training by Resident Staff

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 Technical Specifications (TSs), simulator fidelity, and licensee critique of performance. As part of this inspection, the documents in the attachment 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 a simulator training scenario that included a reactor shutdown with several anomalies which challenged operators.

b. Findings

No findings of significance were identified.

.2 Facility Operating History

a. Inspection Scope

The inspectors reviewed the plant's operating history from January 2002 through January 2004 to assess whether the licensed operator requalification training (LORT) program had identified and addressed operator performance deficiencies at the plant.

b. Findings

No findings of significance were identified.

.3 Licensee Requalification Examinations

a. Inspection Scope

The inspectors performed a biennial inspection of the licensee's LORT test/examination program. The operating examination material reviewed consisted of five operating tests, each containing two dynamic simulator scenarios and six job performance measures. The written examinations reviewed consisted of five written examinations, each containing approximately forty questions. The inspectors reviewed the annual requalification operating test and biennial written examination material to evaluate general quality, construction, and difficulty level. The inspectors assessed the level of examination material duplication from week-to-week during the current year operating test, and compared the operating test material from this year's operating tests (2005) with last year's operating tests (2004). The annual operating tests were conducted in January/February 2004 and January/February 2005. The examiners assessed the amount of written examination material duplication from week-to-week for the written examination administered in January/February 2004. The inspectors reviewed the methodology for developing the examinations, including the LORT program 2-year sample plan, probabilistic risk assessment insights, previously identified operator performance deficiencies, and plant modifications.

b. Findings

No findings of significance were identified.

.4 Licensee Administration of Requalification Examinations

a. Inspection Scope

The inspectors observed the administration of a requalification operating test to assess the licensee's effectiveness in conducting the test. The inspectors assessed the facility evaluators' ability to determine adequate crew and individual performance using objective measurable standards. The inspectors evaluated the performance of one shift crew in parallel with the facility evaluators during two dynamic simulator scenarios and evaluated various licensed crew members concurrently with facility evaluators during the administration of several job performance measures. The inspectors observed the training staff personnel administer the operating test, including conducting pre-examination briefings, evaluations of operator performance, and individual and crew evaluations upon completion of the operating test. The inspectors reviewed the licensee's overall examination security program. The inspectors evaluated the ability of the simulator to support the examinations. A specific evaluation of simulator performance was conducted and documented under Section 1R11.7, "Conformance With Simulator Requirements Specified in 10 CFR 55.46," of this report.

b. Findings

No findings of significance were identified.

.5 Licensee Training Feedback System

a. Inspection Scope

The inspectors assessed the methods and effectiveness of the licensee's processes for revising and maintaining its' LORT program up-to-date, including the use of feedback from plant events and industry experience information. The inspectors reviewed the licensee's quality assurance oversight activities, including licensee training department self-assessment reports. The inspectors evaluated the licensee's ability to assess the effectiveness of its' LORT program and their ability to implement appropriate corrective actions.

b. Findings

No findings of significance were identified.

.6 Licensee Remedial Training Program

a. Inspection Scope

The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the previous biennial requalification examinations and the training planned for the current examination cycle to ensure that they addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans.

b. Findings

No findings of significance were identified.

.7 Conformance With Operator License Conditions

a. Inspection Scope

The inspectors reviewed the facility and individual operator licensees' conformance with the requirements of 10 CFR Part 55. The inspectors reviewed the facility licensees' program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53 (e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators and which control room positions were granted credit for maintaining active operator licenses. The inspectors reviewed the facility licensees' LORT program to assess compliance with the requalification program requirements as described by 10 CFR 55.59 (c). Additionally, medical records for 12 licensed operators were reviewed for compliance with 10 CFR 55.53 (i).

b. Findings

No findings of significance were identified.

.8 Conformance With Simulator Requirements Specified in 10 CFR 55.46

a. Inspection Scope

The inspectors assessed the adequacy of the licensees' simulation facility (simulator) for use in operator licensing examinations and for satisfying experience requirements as prescribed in 10 CFR 55.46, "Simulation Facilities." The inspectors also reviewed a sample of simulator performance test records (i.e., transient tests, scenario test and discrepancy resolution validation test), simulator discrepancy and modification records, and the process for ensuring continued assurance of simulator fidelity in accordance with 10 CFR 55.46. The inspectors reviewed and evaluated the discrepancy process to ensure that simulator fidelity was maintained. Open simulator discrepancies were reviewed for importance relative to the impact on 10 CFR 55.45 and 55.59 operator actions as well as on nuclear and thermal hydraulic operating characteristics. The inspectors conducted interviews with members of the licensee's simulator staff about the configuration control process and completed the Inspection Procedure (IP) 71111.11, Appendix C, checklist to evaluate whether or not the licensees' plant-referenced simulator was operating adequately as required by 10 CFR 55.46 (c) and (d).

b. Findings

No findings of significance were identified.

.9 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 2005. 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 CAP documents, and current equipment performance status. As part of this inspection, the documents in the attachment were utilized to evaluate the potential for an inspection finding.

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

- An issue/problem-oriented review of the primary containment system because it was designated as risk significant under the Maintenance Rule and the inboard drywell vent and purge valve failed to meet its' closing time criteria during quarterly testing.

- An issue/problem-oriented review of the condensate and feedwater system because it was designated as risk significant under the Maintenance Rule and the system experienced a leak on feedwater check valve FW-97-1 just a year after the last repair for a leak on the same area of the valve.

b. Findings

Introduction: A finding of very low safety significance (Green) was self-revealed for a violation of TSs for maintenance personnel failing to perform maintenance in accordance with written procedures associated with air-operated valve AO-2381, the drywell purge inboard isolation valve. This finding was attributed to the cross cutting are of human performance.

Description: The primary containment vent and purge system contains 18-inch butterfly valves that are used to vent the drywell and suppression chamber of the primary containment. These valves also provide a safety-related function to close, to provide a barrier to control the release of fission products in the event of a LOCA. While performing quarterly stroke time testing of air-operated valve AO-2381, the drywell vent and purge valve, it failed to meet the closing time acceptance criteria of the surveillance procedure on the first stroke of the valve. The subsequent stroke time was within the acceptance band.

A work order was written to investigate the valve during stroking and to look for any equipment issues. Because no obvious failures were identified, diagnostic testing was performed. This testing showed that the valve required a significant amount of force to fully seat and unseat the valve disc. Seating load in the valve's design calculation was based on the manufacturer's input of 56 ft-lbs of force. Diagnostic testing showed that Valve AO-2381 required approximately 150 ft-lbs of force to seat. With this additional load, AO-2381 may not be able to close under a LOCA. AO-2381 was declared inoperable and left in the closed position, which is the safety-related position. Prior to repair, Valve AO-2381 passed an as-found leak rate test.

Subsequent investigations determined that the T-ring seat of Valve AO-2381 was replaced in February of 2000. A review of the work order written to perform this maintenance revealed that several steps of Procedure 4321PM, "Primary Containment T-Seated Butterfly Valves," that were required to be completed were marked as "N/A" (not applicable) by the system engineer. Specifically, Step 6 of the procedure required maintenance personnel to measure the torque required to seat the valve disc and use emery paper, as necessary, to eliminate any interference that may exist between the disc and the T-ring seat. This step was marked "N/A" by the system engineer and was not performed. The licensee determined the failure to perform this step was the cause of the valve failure.

Analysis: The inspectors determined that the failure of maintenance personnel to perform maintenance in accordance with written procedures 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." The issue affected the Barrier Integrity cornerstone attribute of maintaining the functionality of containment. Specifically, this issue affected the containment isolation system, structure, and component (SSC) reliability/availability element of the SSC and Barrier Performance attribute. Additionally, this issue affected the Barrier Integrity Human Performance attribute of Routine Maintenance Performance. Because this issue affected both the SSC and Barrier Performance and the Human Performance attributes for the Barrier Integrity cornerstone, this finding was determined to be more than minor.

The inspectors reviewed this finding in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Using the Phase 1 Significance Determination Process (SDP) worksheet for the Barrier Integrity cornerstone, the inspectors determined that there was no degradation of the radiological barrier function provided for the control room, auxiliary building, spent fuel pool, or standby gas treatment (SBGT) system; no degradation of the smoke or toxic gas barrier function of the control room; and the finding did not represent an actual open pathway in the physical integrity of the reactor containment or involve an actual reduction in defense-in-depth for the atmospheric pressure control or hydrogen control functions of the primary containment. Therefore, the finding was considered to be of very low safety significance (Green).

Enforcement: Technical Specification 6.5.A.1 requires written procedures be established, implemented and maintained for the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, February 1978. Appendix A of Regulatory Guide 1.33 requires that maintenance that can affect the performance of safety-related equipment should be performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. Contrary to the above, in February 2000 maintenance workers failed to complete a step in the procedure used to replace the T-ring seat of drywell purge inboard isolation valve AO-2381. As a result, in February 2005, Valve AO-2381 was declared inoperable because the valve may not have been able to close under a design basis accident condition due to the as-found seating force being more than allowed by an engineering calculation. Because this violation was of very low safety significance, and it was entered into the licensee's CAP, this violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000263/2005002-01). The licensee entered this issue into their corrective action program as CAP035915 and have repaired the valve and returned it to operable status after completing post-maintenance testing.

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

a. Inspection Scope

The inspectors reviewed maintenance activities to review risk assessments (RAs) and emergent work control. The inspectors verified the performance and adequacy of RAs, 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 RAs 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 the attachment 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 four samples:

- routine scheduled maintenance and risk management during emergent work that occurred while auxiliary transformer 2R was isolated for a fault;
- routine scheduled maintenance and risk management during emergent work that occurred while the auxiliary 1R transformer was isolated;
- routine scheduled maintenance and risk management during emergent work that occurred while modifications were performed on the EDG ventilation system; and
- routine scheduled maintenance and risk management during emergent work that occurred while personnel were performing trenching operations in the switchyard.

b. Findings

No findings of significance were identified.

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

.1 No. 11 and No. 12 Emergency Diesel Generators Declared Inoperable for Single Failure Vulnerability of 1AR Transformer

a. Inspection Scope

The inspectors reviewed personnel performance when the 11 and 12 EDGs were declared inoperable due to a single failure vulnerability found on the 1AR transformer. The inspectors monitored the licensee's assessment of the condition, root cause investigation, repair activities, and corrective actions relative to the single failure vulnerability. The inspectors monitored the evaluation of the issue to assess the potential contribution to future events that may arise as a result of the licensee's assessment and corrective actions for the degraded condition. The inspectors' reviews included, but were not limited to, pre-job briefings, observations of repair activities, root cause analysis, reporting requirements, and corrective actions. As part of this inspection, the documents in the attachment were utilized to evaluate the potential for an inspection finding. This observation constituted one sample.

b. Findings

No findings of significance were identified.

.2 Reactor Coolant Leakage Identified at the Insert Line and Flange Interface for Control Rod 38-31

a. Inspection Scope

The inspectors reviewed personnel performance when reactor coolant leakage was identified during post reactor shutdown inspections of the drywell at the interface between the control rod drive (CRD) insert line and the control rod flange for Rod 38-31. The inspectors monitored the licensee's assessment of the condition, root cause investigation, repair activities, and corrective actions relative to the leak. The inspectors monitored the evaluation of the issue to assess the potential contribution to future events that may arise as a result of the licensee's assessment and corrective actions for the degraded condition. The inspectors reviews included, but were not limited to, pre-job briefings, observations of repair activities, root cause analysis, reporting requirements, ASME code requirements, and corrective actions. As part of this inspection, the documents in the attachment were utilized to evaluate the potential for an inspection finding. This observation constituted one sample.

b. Findings

Introduction: The inspectors identified an unresolved item (URI) for a potential failure to make an 8-hour notification as required by 10 CFR 50.72(b)(3)(ii).

Description: On March 5, 2005, the licensee identified leakage on four control rod drive mechanism flanges during their post-shutdown drywell entry inspections. Upon removal of the CRD housing ejection support steel, the licensee performed a detailed inspection of the four suspect drives. This inspection revealed that the leakage on three of the mechanisms was mechanical joint leakage from the flange joint; the fourth leakage source was confirmed to be from a location other than a mechanical joint at a location where the insert line met the CRD flange. Subsequent to the inspection, the licensee performed a root cause analysis to determine the exact location and the cause of the leakage.

Because the licensee identified leakage from a CRD line on March 9, 2005, and because there was extensive information related to prior indications and leakage associated with CRD lines, the residents provided the data to regional inspectors performing the baseline inspection of outage-related inservice inspection activities. The regional inspectors reviewed the licensee's evaluation of this issue and any relationship to prior CRD observations using IP 71111.08, "Inservice Inspection Activities" (Section 1R08).

Additionally, the inspectors were concerned that the identified leakage was from a location that constituted reactor coolant pressure boundary leakage as defined in 10 CFR 50.2. The inspectors reviewed this aspect of the issue and determined that further review of the licensee's conclusions was required to determine if the licensee failed to make an 8-hour notification as required by 10 CFR 50.72(b)(3)(ii). During initial discussions with the inspectors, the licensee indicated that licensee event report (LER) was not required. After additional discussions with regional and headquarters management, the licensee proposed a voluntary LER for the issue. The inspectors concluded that the LER would have met reporting requirements outlined in 10 CFR 50.73. As a result, the inspectors were concerned that the regulatory process may have been impeded by the lack of an 8-hour report as required by 10 CFR 50.72(b)(3)(ii). The inspectors consider the potential failure to make a required 8-hour report unresolved (URI 05000563/2005002-02). This issue will be reviewed using IP 71153, upon receipt of the LER from the licensee.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations which affected 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 TSs, 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 the attachment were utilized to evaluate the potential for an inspection finding.

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

- valve times out-of-specification for turbine generator quarterly testing;
- AO-2381, drywell purge inboard isolation valve; and
- bent fitting found on AI-221 for control valve of high pressure core injection (HPCI) minimum flow line.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed an operator workaround (OWA) and focused on verification of the selected workarounds' impact on the functionality of a mitigating system. The inspection activities included, but were not limited to, a review of the selected workaround to determine if the functional capability of the system or human reliability in responding to an initiating event was affected, including a review of the impact of the workaround on the operator's ability to execute emergency operating procedures. As part of this inspection, the documents in the attachment were utilized to evaluate the potential for an inspection finding.

The inspectors reviewed the following OWA for a total of one sample:

- OWA 03-048, apparent thinning torus cooling line downstream of MO-2008.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors' review of permanent plant modifications focused on verification that the design bases, licensing basis, and performance capability of related (SSCs) structures, systems or components were not degraded by the installation of the modification. The inspectors also verified that the modifications did not place the plant in an unsafe configuration. The inspection activities included, but were not limited to, a review of the design adequacy of the modification by performing a review, or partial review, of the modification's impact on plant electrical requirements, material requirements and replacement components, response time, control signals, equipment protection, operation, failure modes, and other related process requirements. As part of this inspection, the documents in the attachment were utilized to evaluate the potential for an inspection finding.

The inspectors selected the following permanent plant modification for review for a total of one sample:

- No. 11 and 12 EDG room ventilation system upgrade.

b. Findings

No findings of significance were identified.

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 the attachment 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 testing of 11 EDG after maintenance;
- post-maintenance testing of Bus 15 outage relay after maintenance;
- post maintenance testing of 11 EDG ventilation modification; and
- post-maintenance testing of AO-2381, the drywell purge inboard isolation valve.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated outage activities for a refueling outage that began on March 5, 2005, and was still in progress at the end of the inspection period. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule, developed mitigation strategies for loss of key safety functions, and adhered to operating license and TS requirements to ensure defense-in-depth. The inspection activities included, but were not limited to, a review of the outage plan, monitoring of shutdown activities, control of outage activities and risk, and observation of reduced inventory operations, maintenance and refueling activities. As part of this inspection, the documents in the attachment were utilized to evaluate the potential for an inspection finding.

In addition to activities inspected utilizing specific procedures, the following represents a partial list of the major outage activities the inspectors reviewed/observed, all or in part:

- review of both outage plans and the ready-backlog;
- control room turnover meetings and selected pre-job briefings;
- control room demeanor, communications, self/peer checking, and equipment panel control;
- outage management turnover meetings;
- reactor shutdown and cooldown;
- CRD piping inspections;
- steam dryer and separator removal and installation;
- initial drywell entry inspection with reactor still under pressure to inspect for reactor coolant system leaks;
- walkdowns of the reactor and turbine building to observe ongoing work activities;
- walkdowns of the main control room to observe alignment of systems important to shutdown risk;
- leak rate testing activities;
- outage equipment configuration and risk management;
- electrical line-ups;
- selected clearances;
- control and monitoring of decay heat removal;
- refueling activities;
- feedwater line thinning discovered during non-destructive examination;
- foreign material exclusion issues associated with the reactor vessel;
- main steam line isolation valve repair; and
- identification and resolution of problems associated with the outage.

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 SSCs 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 SSC could impose on the unit if the condition was 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 PI reporting, and evaluation of test data. As part of this inspection, the documents in the attachment 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:

- primary containment isolation valve exercise;
- transversing incore probe (TIP) explosive valve testing and monitoring;
- anticipated transient without a scram recirculation pump high pressure and low level trip test and calibration;
- No. 16 250Vdc battery capacity test;
- primary containment isolation valve exercise; and
- primary containment purge and vacuum breaker, pressure and isolation valve local leak rate testing (LLRT) test.

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 PI. 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 CAP entries. The inspectors verified that there were no discrepancies between observed performance and PI reported statistics. As part of this inspection, the documents in the attachment 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 annual drill sample:

- the inspectors observed the licensee's annual drill to evaluate drill conduct and the adequacy of the licensee's critique of performance to identify weaknesses and deficiencies. Drill notifications were made with state, county, and local agencies for a general emergency classification.

b. Findings

No findings of significance were identified.

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following three radiologically significant work areas within radiation areas, high radiation areas (HRAs), and airborne radioactivity areas in the plant and reviewed work packages, which included associated licensee controls and surveys of these areas to determine if radiological controls including surveys, postings, and barricades were acceptable:

- reactor coolant pump motor and impeller replacement;
- drywell main steamline isolation valve work; and
- drywell radiation protection (RP) and as-low-as-reasonably-achievable (ALARA) efforts.

This review represented one sample.

The inspectors reviewed the radiation work permits (RWPs) and work packages used to access these three areas and other high radiation work areas to identify the work control instructions and control barriers that had been specified. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. Workers were interviewed to verify that they were aware of the actions required when their electronic dosimeters noticeably malfunctioned or alarmed. This review represented one sample.

The inspectors walked down and surveyed (using an NRC survey meter) these three areas to verify that the RWP, procedure, and engineering controls were in place; that licensee surveys and postings were complete and accurate; and that air samplers were properly located. This review represented one sample.

The inspectors reviewed RWPs for the following two airborne radioactivity areas to verify barrier integrity and engineering controls performance (e.g., high efficiency particulate air (HEPA) ventilation system operation) and to determine if there was a potential for individual worker internal exposures of greater than 50 millirem committed effective dose equivalent:

- drywell main steamline isolation valve work; and
- turbine blade sandblasting.

Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection. This review represented one sample.

The adequacy of the licensee's internal dose assessment process for internal exposures greater than 50 millirem committed effective dose equivalent was assessed. There were no internal exposures greater than 50 millirem. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, LERs, and special reports related to the access control program to verify that identified problems were entered into the CAP for resolution. This review represented one sample.

The inspectors reviewed fifteen corrective action reports related to access controls and two HRA radiological incidents when available (non-performance indicators identified by the licensee in HRAs less than 1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation/consideration of risk significant operational experience feedback.

This review represented one sample.

The inspectors evaluated the licensee's process for problem identification, characterization, and prioritization and verified that problems were entered into the CAP and resolved. For repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, the inspectors verified that the licensee's self-assessment activities were capable of identifying and addressing these deficiencies. This review represented one sample.

The inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if any of these PI events involved dose rates greater than 25 R/hr at 30 centimeters or greater than 500 R/hr at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. There were no PI events occurring since the last inspection. This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following four jobs that were being performed in radiation areas, airborne radioactivity areas, or HRAs for observation of work activities that presented the greatest radiological risk to workers:

- reactor coolant pump motor and impeller replacement;
- drywell main steam line isolation valve work;
- drywell RP and ALARA efforts; and
- turbine fan blade sandblasting.

The inspectors reviewed radiological job requirements for these four activities, including RWP requirements and work procedure requirements, and attended ALARA job briefings. This review represented one sample.

The above review is combined with IP 71121.02, “ALARA Planning and Controls,” and documented in Section 02.02 of this report.

Job performance was observed with respect to these requirements to verify that radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings. The inspectors also verified the adequacy of radiological controls including required radiation, contamination, and airborne surveys for system breaches; RP job coverage, which included audio and visual surveillance for remote job coverage; and contamination controls. This review represented one sample.

Radiological work in high radiation work areas having significant dose rate gradients was reviewed to evaluate the application of dosimetry to effectively monitor exposure to personnel and to verify that licensee controls were adequate. These work areas involved areas where the dose rate gradients were severe (i.e., diving activities and the reactor water cleanup heat exchanger room), which increased the necessity of providing multiple dosimeters and/or enhanced job controls. No jobs observed required multiple dosimeters. This review represented one sample.

b. Findings

No findings of significance were identified.

.4 High Risk Significant, High Dose Rate High Radiation Area and Very High Radiation Area Controls

a. Inspection Scope

The inspectors held discussions with the RP manager concerning high dose rate/high radiation area and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to verify that any procedure modifications did not substantially reduce the effectiveness and level of worker protection. This review represented one sample.

The inspectors discussed with RP supervisors the controls that were in place for special areas that had the potential to become very high radiation areas during certain plant operations, to determine if these plant operations required communication beforehand with the RP group, so as to allow corresponding timely actions to properly post and control the radiation hazards. This review represented one sample.

The inspectors conducted plant walkdowns to verify the posting and locking of entrances to high dose rate HRAs and very high radiation areas. This review represented one sample.

b. Findings

No findings of significance were identified.

.5 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated RP work requirements and evaluated whether workers were aware of the significant radiological conditions in their workplace, of the RWP controls and limits in place, and that their performance had accounted for the level of radiological hazards present. This review represented one sample.

The inspectors reviewed radiological problem reports which found that the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. These problems, along with planned and taken corrective actions were discussed with the RP manager. This review represented one sample.

b. Findings

No findings of significance were identified.

.6 Radiation Protection Technician Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation protection technician (RPT) performance with respect to RP work requirements and evaluated whether they were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities. This review represented one sample.

The inspectors reviewed radiological problem reports, which found that the cause of the event was RPT error, to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. This review represented one sample.

b. Findings

No findings of significance were identified.

2OS2 As Low As Is Reasonably Achievable Planning And Controls (71121.02)

.1 Radiological Work Planning

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities ranked by estimated exposure that were in progress and reviewed the following five work activities of highest exposure significance:

- reactor coolant pump motor replacement;
- reactor coolant pump impeller replacement;
- drywell main steamline isolation valve work;
- drywell RP and ALARA efforts; and
- drywell scaffolding.

This review represented one sample.

For these five activities, the inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements in order to verify that the licensee had established procedures, and engineering and work controls that were based on sound RP principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances. This review represented one sample.

The inspectors compared the results achieved, including dose rate reductions and person-rem used, with the intended dose established in the licensee's ALARA planning for these five work activities. Reasons for inconsistencies between intended and actual work activity doses were reviewed. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the following five jobs that were being performed in radiation areas, airborne radioactivity areas, or HRAs for observation of work activities that presented the greatest radiological risk to workers:

- reactor coolant pump motor replacement;
- reactor coolant pump impeller replacement;
- drywell main steamline isolation valve work;
- drywell RP and ALARA efforts; and
- drywell scaffolding.

The licensee's use of ALARA controls for these work activities was evaluated. Specifically, the licensee's use of engineering controls to achieve dose reductions was evaluated to verify that procedures and controls were consistent with the licensee's ALARA reviews, that sufficient shielding of radiation sources was provided for, and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding. This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Source-Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to determine the historical trends and current status of tracked plant source terms and to evaluate if the licensee was making allowances and had developed contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry. Additionally, the inspectors reviewed the licensee's implementation of PRC2 resin systems, use of additional permanent shielding in the drywell and a slow-fill method to reduce source term movement, and increased contamination control during flood-up. This review represented one sample.

b. Findings

No findings of significance were identified.

.4 Declared Pregnant Workers

a. Inspection Scope

The inspectors reviewed dose records of declared pregnant workers for the current assessment period to verify that the exposure results and monitoring controls employed by the licensee complied with the requirements of 10 CFR Part 20. Currently there are no declared pregnant workers. This review represented one sample.

b. Findings

No findings of significance were identified.

.5 Problem Identification and Resolutions

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and special reports related to the ALARA program since the last inspection to determine if the licensee's overall audit program's scope and frequency for all applicable areas under the Occupational cornerstone met the requirements of 10 CFR 20.1101(c). This review represented one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

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

As part of the routine inspections documented above, the inspectors verified that the licensee entered the problems identified during the inspection into their CAP. Additionally, the inspectors verified that the licensee was identifying issues at an appropriate threshold and entering them in the CAP, and verified that problems included in the licensee's CAP were properly addressed for resolution. Attributes reviewed included: complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization and focus were commensurate with safety and sufficient to prevent recurrence of the issue.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished by reviewing daily CAP summary reports and attending corrective action review board meetings.

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-up (Annual Sample): Review of the Effectiveness of the Corrective Action Program Action Request Screening Process Prior to and During the Refueling Outage

Introduction:

Monticello Nuclear Generation Station maintains an action request screening team. The team function is to screen action requests initiated and identify any inconsistencies or errors. The team also assigns significance level, identifies trends, and handles requests for changes in significance level and evaluations.

a. Inspection Scope

As part of an annual sample the inspectors reviewed fifteen action requests, eight initiated prior to the refueling outage and seven initiated during the refueling outage. The action requests were reviewed after being screened by the screening team. The fifteen action requests reviewed were:

- CAP036318; ALARA: NRC Resident Inspector Questioned the Location of a Fire Extinguisher in "A" Residual Heat Removal (RHR);
- CAP036503; Activity LAR 021809 Overdue and Not Approved by Due Date;
- CAP036517; 3 CAP Activity Due Dates in December Extended Without Management Approval;
- CAP036613; WO Completed Without RWP Being Assigned;
- CAP036891; 250 Volt Battery Charger Parameters Found Out of Spec During 4525 PM;
- CAP036726; Implementation Phase for CA022563 is not in the Corrective Action Program;
- CAP037132; Corrective Action Overdue;

- CAP037163; Two Due Dates Extended in January Without Proper Management Approval;
- CAP037446; Reactor Core Isolation Cooling (RCIC) Operability Challenged by Insulation Removal;
- CAP037389; Opening of Breaker B4117 for WO 0403632 Caused an Inadvertent Closure of AO-2886;
- CAP037847; Condensate/Feedwater Valve Breached Without Notifying Radiation Protection;
- CAP037903; NRC Question - Did Workers Received Pre-Entry High Radiation Area Brief as Require;
- CAP037875; Bagged Material in 1001' Ops LLRT Cage Not Tagged per SSS #78;
- CAP037874; Bagged Material in 1001' Decon Area Not Tagged per SSS #77 (3rd Time in 11 Days); and
- CAP038102; Potentially Improper Grease Issued as Mobilux EP-2 (Safety-Related).

b. Issues

During the review of the action requests the inspectors noticed an increased number of inconsistencies not identified during the screening process for those action requests initiated during the refueling outage. Five out of fifteen action requests had inconsistencies; e.g., incorrect reference of documents. Four of these action requests were initiated during the refueling outage and one was initiated prior to the outage. The inspectors noticed a higher percentage of inconsistencies in those action requests initiated during the outage than in those initiated prior to the outage. The five action requests noted and their associated inconsistencies were:

- CAP036891 - incorrect use of units (volts and amps);
- CAP037389 - should have screened as a potential maintenance rule functional failure because the failure occured as part of a maintenance activity and could have been prevented because Operations Manual B.08.09, "Condensate Storage System," specifically has precautions to prevent the condition from occurring ;
- CAP037446 - an operability evaluation for RCIC was recommended but no senior reactor operator (SRO) review was required;
- CAP037874 - incorrect document referenced in the recommendations; and
- CAP037847 - should have screened as ALARA using the appropriate "hot button;" however, the screening committee indicated "none" for "hot button."

The inspectors concluded that these inconsistencies were minor and related to the process.

.4 Biennial Sample Review for Licensed Operator Requalification Program

a. Inspection Scope

The inspectors reviewed several licensee training department self-assessment reports. The licensee's self-assessments reviewed the licensed operator training program for approximately 12 months prior to this inspection activity. The self-assessments were reviewed to ensure that any issues identified during the self-assessment were appropriately evaluated, prioritized, and controlled.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report 50-263/2004-003: "High Pressure Coolant Injection System Declared Inoperable Due to Loose Oil Plug"

On December 15, 2004, the licensee identified that the oil drain plug on the HPCI turbine drive booster pump was loose and that HPCI operation could not be assured with the degraded condition. The condition was determined to be the result of improper tightening of the drain plug when an oil sample had been taken late on December 14, 2004. The licensee evaluated the issue and determined it to be of very low safety significance because the issue resulted in HPCI being inoperable for a nine-hour period of time, and that this limited period of inoperability had no significant increase in core damage frequency. The licensee attributed the cause of the issue to a lack of programmatic controls to ensure minimum required tightness was applied when reinstalling drain plugs after sampling or changing oil. Corrective actions included revisions to related procedures for oil sampling and changing to include a definition of mechanically tight and a requirement for second verification of tightness upon completion of related activities. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee entered this issue into their corrective action program as CAP 036268.

.2 Trip of the "A" Reactor Protection System Motor Generator Set

b. Inspection Scope

On February 24, 2005, the inspectors observed plant operators response to and plant recovery from a trip of the "A" reactor protection system motor generator set. The trip caused a Group 2 primary containment isolation and an autostart of the SBGT system. The inspectors observed control room operations, procedure usage, plant parameters, alarms and condition related to the event, resolution of the event, and the licensee's management of risk.

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 1R12 of this report had as its primary cause a Human Performance deficiency, in that maintenance personnel failed to complete the required steps of a maintenance procedure for a primary containment isolation valve and it was later determined that the valve had excessive seating force that would prevent it from closing under a LOCA.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. Palmisano and other members of licensee management on April 7, 2005. 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:

- Biennial Operator Requalification Program Inspection with Mr. J. Conway, Site Director of Operations, on January 28, 2005; and
- Occupational Radiation Safety inspection with Mr. J. Conway, Site Director of Operations, on March 18, 2005.
- Baseline procedure 71111.08 with Mr. Sawatzke and other members of your staff on March 18, 2005, then again on March 24, 2005, with Mr. Fields and other members of your staff with regard to the NCV.

4OA7 Licensee-Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
T. Palmisano, Site Vice President
J. Conway, Site Director for Operations
J. Grubb, Plant Manager (Acting)
R. Baumer, Licensing
K. Jepsen, Radiation Protection Manager
D. Neve, Regulatory Affairs Manager
J. Fields, Regulatory Affairs Manager (Acting)
R. Buldoc, Requalification Program Lead
M. Carey, Supervisor, Operations Training
S. Halbert, Training Manager
B. MacKissock, Operations Manager
J. Ruth, Examination Developer
B. Sawatzke, Acting Plant Manager
R. Diopere, ISI Coordinator
T. Jones, NDE Coordinator

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

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000563/2005002-01	NCV	Failure to Complete required Procedure Steps Leads to Inoperable Primary Containment Isolation Valve (Section 1R12)
05000263/2005002-02	URI	Reactor Coolant Leakage Identified at the Insert Line and Flange Interface for Control Rod 38-31 (Section 1R14)
05000263/2005002-03	NCV	Failure to check the illumination levels of the battery powered light before or after the VT-3 examination of an RHR heat exchanger support (Section IR08)

Closed

05000563/2005002-01	NCV	Failure to Complete required Procedure Steps Leads to Inoperable Primary Containment Isolation Valve (Section 1R12)

05000263/2004-003	LER	High Pressure Coolant Injection System Declared Inoperable Due to Loose Oil Plug (Section 4OA3.1)
05000263/2005002-03	NCV	Failure to check the illumination levels of the battery powered light before or after the VT-3 examination of an RHR heat exchanger support (Section IR08)

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:
M-127; SBLC System; Revision W
2154-07; SBLC System Prestart Valve Checklist; Revision 11
2161; Plant Prestart Checklist Process Instrumentation; Revision 25
Emergency Filtration Train and Administrative Building Operator Logs for January 1 to January 4, 2005
Monticello Station Logs for January 3 and January 4, 2005
2124; Plant Prestart Checklist Diesel Generators and Fuel Oil System; Revision 7
4 AWI-04.05.10; Scaffolding Controls; Revision 4
8146; Scaffold Control; Revision 19
4179-01; Loop A RHR - Shutdown Cooling Mode; Revision 9
4179-02; Loop B RHR - Shutdown Cooling Mode; Revision 8

Corrective Action Program Documents:
CAP 033626; Determine If Any Nuclear Inst. Procedures Have Adequate Configuration Control
CAP 036782; Scaffolding Interference would Prevent Access to Manually Bar Over 12 EDG

Surveillances:
0255-02-III; SBLC Quarterly Pump and Valve Tests; dated December 29, 2004
0255-02-III-1A; SBLC Comprehensive Pump and Valve Tests; dated December 29, 2004

1R05 Fire Protection

Pre-Fire Fighting Procedures and Strategies:
A.3-03-C; West Hydraulic Control Unit Area; dated June 24, 2004
A.3-30; Turbine Deck; Revision 7
A.3-12-C; Condenser Area; Revision 5

Documents and Procedures:
4 AWI-08.01.01; Fire Prevention Practices; Revision 21

Corrective Action Program Documents:
CAP 034697; Transient Combustibles Identified in the Reactor & Radwaste Buildings
CAP 036644; NRC Walkdown of Fire Zone 1F (Torus Area) Identified Various Housekeeping Issues (NRC Identified)

CAP 037783; Evaluate Surveillance Frequency Inspection of Extinguishers in HRAs (NRC-identified)

Work Orders:
0403663; Add/Exchange Fire Extinguishers in Plant; dated October 26, 2004

1R08 Inservice Inspection Activities (IP 71111.08)

Corrective Action Documents Prompted by NRC Inspection
CAP037910; ISI Non Destructive Examination (NDE) Examiner Procedural Errors; dated March 17, 2005
CE012073; ISI NDE Examiner Procedural Deficiencies; dated April 6, 2005.

Documents and Procedures
Work Order 0108101; Main Steam Line Drain - Outboard; dated May 16, 2003
P1P1GTNO; Welding Procedure Specification; dated December 10, 1999
ISI-7905-32-A; SYSTEM: RHR HX "A"; Revision 4
ISI-786-A; Main Steam Condensate Leakoff; Revision 6
PEI-02.03.15; Ultrasonic Examination of Reactor Pressure Vessel Welds to Appendix VIII; Revision 0; dated February 11, 2005
PEI-02.02.01; Dry Powder Magnetic Particle Examination; Revision 0; dated April 18, 2003
PEI-02.05.02; Visual Examination of Components and Their Supports; Revision 0; dated April 18, 2003

Corrective Action Program Documents
CAP026463; Liquid Penetrant Exam of Flange; dated May 11, 2003
CAP027041; Loose Nut Found During VT Exam of Hanger SWH-1; dated May 15, 2003
CAP027983; Reg Guide 1.193: ASME Code Case Not Approved for Use; dated June 17, 2003
CAP028067; OE16421 - Preliminary - PDI Qualified Ultrasonic Examination Detects Unusual Construction Welding Flaw; dated June 23, 2003
CAP027023; Loose Base Plate Bolts on HPCI Hangers SR-597 and TWH-47; dated May 16, 2003
CAP028947; Leakage Found on Bottom-Mounted Instrumentation; dated August 14, 2003

1R11 Licensed Operator Requalification Program

Documents and Procedures:
OWI-01.08; NRC License Maintenance Responsibilities; Revision 5
MTCP-02.02; Monticello Plant Modification Review; Revision 5
MTCP-02.12; Simulator Testing; Revision 8
M9100; Licensed Operator Requalification; Revision 11
Operations Department Organization/Qualification; dated January 19, 2005
NRC License Active Status Maintenance; Revision 6
SA021885; Training Self-Evaluation - Simulator Management Process; dated July 23, 2004
Requalification Examination Control, MTCP-03.32; Revision 19

Initial and Requalification Examination Security, MTCP-03.35; Revision 9
Conduct of Training Cycle Evaluations, MTCP-03.40; Revision 8
MNGP Licensed Operator Requalification (LOR) Two-Year Plan, 2004/2005 Training Program; Revision 0
LOR Curriculum Review Committee Meeting Minutes - Various; 2004 - 2005
Nuclear Oversight Observation Report; 2004-003-5-023
Year 2004 Requalification Exam Summary Report
Monticello Training Program Self-Assessment Report; December 2003
MNGP Training Self-Evaluation Report Operations Training Programs Comprehensive Evaluation; dated March 15 - 19, 2004
Various Condition Reports Involving MNGP Training Issues (7)
Various Trainee Feedback Summary Forms (QF-1050-01, Revision 3) (12)

Corrective Action Program Documents:
CAP 036322; Operator Simulator Response Time; dated December 17, 2004

1R12 Maintenance Effectiveness

Documents and Procedures:

Corrective Action Program Documents:
CAP036186; Step N/A'd on 4321PM with no Basis Given
CAP037974; Packing and Seating Torque Higher than Expected in Calculation
CAP038010; AO-2381 Packing Load Found Higher than Expected During Diagnostic Testing
CAP038035; AO-2896 Packing Load Found Higher than Expected During Diagnostic Testing
CAP038040; AO-2383 Packing Load Found Higher than Expected During Diagnostic Testing
CAP038159; AO-2377 Packing Load Higher than Expected
CAP038036; Issues Identified with Work on AO-2380 have not been Captured in Action Request Process
CAP038145; As-Found Seating Torque for AO-2378 Found Higher than Shown in Calculation
CAP036660; Last Two AO-2378 Close Stroke Tests Show Adverse Trend Over Previous Results
ACE004288; AO-2381 Found with Higher than Expected Seating Torque
CAP035915; AO-2381 Found with Higher than Expected Seating Torque
CAP034959; AO-2381 Failed to Meet Closing Time Acceptance Criteria on First Stroke of Valve

Work Orders:
WO0000638; Perform 4321PM on Valve to Resolve LLRT Failure
WO0404038; Repair AO-2381
WO0202791; Replace Cam Valve on AO-2381

1R13 Maintenance Risk Assessments and Emergent Work Control

Documents and Procedures:
Equipment Out of Service Model Risk Evaluation Cutset for a 2R Transformer Failure
Monticello Station Logs for January 20 through January 23, 2005
Monticello Station Logs for the week ending February 12, 2005
Work Week Schedule for Work Week 5110 (February 6 through February 12, 2005)
Work Week Schedule for Work Week 5113 (February 27 through March 5, 2005)
Monticello Station Logs for the week ending March 5, 2005
Monticello Station Logs for the week ending February 2, 2005

Corrective Action Program Documents:
CAP036784; NRC Resident Informed of Wrong Core Damage Frequency (CDF) Associated with 2R XFMR Out
CAP036779; Initial Classification for Loss of 2R Cooling was Incorrect
CAP036773; Loss of Oil Flow and Cooling to the 2R Transformer Requires Transfer to the 1R Transformer
CAP036785; NRC Resident Questioned Extent of Condition Review of Fault on 2R Transformer
CAP036786; Operations Manual Procedure not Initially Used when Attempting to Close Breaker 3N4
CAP036777; Expected Procedure to be Used for Transfer from 2R to 1R is not specified in Alarm Response Procedure

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

Documents and Procedures:
Monticello Station Logs for February 4, 2005

Corrective Action Program Documents:
CAP037264; Alternate Shutdown System Isolation Design Issue Could Prevent Bus 16 from Energizing
CAP037270; Additional Vulnerability Identified for Modification 05Q035 after MOD Turnover
CAP037270; Inadequate Appendix R Review for Mod 05Q035 Results in Engineering Change Notice
CAP037461; Undervessel Leakage Inspection Identified Four CRD Flange Leaks

1R15 Operability Evaluations

Documents and Procedures:
1040-01; Turbine - Generator; Revision 46
0300; Turbine Stop Valve Closure Calibration Checks; Revision 4
0011-A; Turbine Control Valve Fast Closure Scram Test and Calibration (>30% of Rated); Revision 6
QF-1100; Operability Recommendation for AO-2381 Associated with CAP035915
NX-16854; NSP Technical Manual for Direct Acting Diaphragm Actuator Fisher Governor, Type 656; Revision 7
NX-16850; NSP Technical Manual for Type 9200 T-Ring Butterfly Valve; Revision 7

CA-01-154; Calculation for Allowable Leakage Rate for HPCI Minimum Flow Control Valve Accumulator System, Revision 1
M-124; HPCI; Revision AE
M-131 NH-36049-14; Instrument Air Reactor Building; Revision AD

Corrective Action Program Documents:
CAP036057; Valve Times out of Specification for Turbine Generator Quarterly Testing
GEN01000780; During the Performance of Test 1040-01 (Turbine - Generator), Stop Valves Number 1 and 3 Were out of Specification
GEN00004856; Turbine Generator Stop Valves Number 1 and 3 and Turbine Bypass Valve Number 1 Had Problems Meeting Their Designated Stroke Times During Performance of the Turbine Generator Quarterly Test 1040-01
CAP036660; Last 2 AO-2378 Close Stroke Tests Show Adverse Trend over Previous Results
CAP036186; Step N/A'd on 4321PM with No Basis Given
GEN00000432; The Inboard Containment Isolation Leak Valve on the Drywell Purge Line, AO-2381, Failed its As-Found Leak Rate Test
GEN01000090; Duane Arnold Energy Center Informed MNGP That an Analysis of Their Primary Containment Vent and Purge Valves Indicated a Required Torque Value Higher than Actuator Capability for 4 of Their Valves
CAP037215; Accumulator Supply for CV-2065 Damaged
CAP033694; Outage Drywell Walk Down Found Leakage on FW-97-1
CAP033707; Roll Pin Sheared on FW-97-1 Hinge Pin
CAP037464; FW-97-1 Requires Repair Less than a Year After Last Repair

Work Orders:
0505081; Replace Twisted and Bent Fitting Downstream AI-221
0402342; FW-97-1 Has Leak at Hinge Plug
0402354; Repair Weld Check Valve

Drawings:
NX-9235-37, Rev. G

1R16 Operator Workarounds

Documents and Procedures:
03-048; Apparent Thinning Torus Cooling Line Downstream of MO-2008

Corrective Action Program Documents:
CAP028756; Evaluation of Torus Cooling Downstream from MO-2008 for Pipe Thinning
CAP029014; Added to Operator Challenge List as Operator Workaround
RCE000852; Evaluation of Torus Cooling Line Downstream MO-2008

1R17 Permanent Plant Modifications

Documents and Procedures:
Modification 04Q030; Revision 0; EDG Ventilation System Upgrade
8146; Scaffold Control for 11 EDG Room; January 3, 2005
8146; Scaffold Control for 12 EDG Room; January 10, 2005

CA-04-145; EDG Ventilation: Cooling Load and Airflow Determination; Revision 0

Corrective Action Program Documents:
CAP036782; Scaffolding Interference Would Prevent Access to Manually Bar over 12 EDG

Work Orders:
0402306; Modify 12 EDG Ventilation Ducting
0402307; Modify 11 EDG Ventilation Ducting
0403218; Measure 12 EDG Ventilation Room Flow
0403219; Measure 11 EDG Ventilation Room Flow

1R19 Post-Maintenance Testing

Documents and Procedures:
4108-01; Emergency Diesel Generator 6 Year Maintenance; Revision 7
4850-915; Bus 15 Outage Relay Maintenance; Revision 2
3108; Pump/Valve/Instrument Record of Corrective Action for AO-2381; March 22, 2005
4321-PM; Primary Containment T-Seated Butterfly Valves; Revision 4
3062-05; ASME Section XI Repair/Replacement Plan for AO-2381; March 17, 2005
0137-04; Primary Containment Purge and Vacuum Breaker, Pressure and Isolation Valve LLRT Test for AO-2381; March 24, 2005

Corrective Action Program Documents:
CAP037840; #11 EDG Engine Cranking Speed During Air Start Test was Low
CAP037426; Vendor Made Errors When Calculating V-SF-10 (11 EDG room) Airflow

Work Orders:
0307179;4850-915 (Bus 15 Relays)
0404038; Repair AO-2381
0205022; Baseline Test AO-2381 for Air Operated Valve (AOV) Program
0403219; Measure 11 EDG Ventilation Room Flow

1R20 Outage Activities

Documents and Procedures:
2005 Refueling Outage Daily Risk Data Sheets
2005 Outage Daily Shift Turnover Reports
Monticello Nuclear Generating Plant 2005 Refuel Outage Critical Path Schedule

Corrective Action Program Documents:
CAP037391; NRC Resident Questioned Numerous Temporary Cables Routed Throughout Reactor Building 935 Level (NRC Identified)
CAP037393; NRC Resident Questioned Seismic Implications of Material Under Scaffold Legs (NRC Identified)
CAP037618; NRC Questions Contaminated Grating Area Posting above Uncontaminated Area (NRC Identified)
CAP037783; Evaluate Surveillance Frequency Inspection of Extinguishers in High Radiation Areas (NRC Identified)

1R22 Surveillance Testing

Documents and Procedures:
0255-10-IA-1; Primary Containment Isolation Valve Exercise Performed January 11, 2005; Revision 29
Monticello Station Logs for January 11, 2005
0255-18-IC; TIP Explosive Valve Testing and Monitoring; Revision 12
0278-B; ATWS - Recirc Trip for Reactor Pressure and Level Trip Unit Test and Calibration, Revision 13
0197-02; 16 250 VDC Battery Capacity Test; Revision 9
0137-04; Primary Containment Purge and Vacuum Breaker, Pressure and Isolation Valve LLRT Test; Revision 14
0255-10-IA-1; Primary Containment Isolation Valve Exercise; January 11, 2005
Monticello Station Logs for January 11 and 12, 2005

Corrective Action Program Documents:
GEN03006744; While Performing Test 0255-10-IA-1 the Opening Time of AO-2377 Was out of Specification Slow at 12.0 Seconds
ACE003852; AO-2377 Slow Opening Time During Performance of Test 0255-10-IA-1
ACE001658; AO-2377 Failed Closing Stroke Time Testing
GEN02000161; AO-2377 Failed Closing Time During Testing
GEN00005136; During Performance of Primary Containment Isolation Valve Exercise 0255-10-IA-1; Torus Purge Inboard Valve AO-2378 Exceeded the Acceptance Band in the “Closed” Direction
CAP016351; AO-2378 Open Time Exceeded the Reference Value

1EP6 Drill Evaluation

Documents and Procedures:
5790-102-02; Monticello Emergency Notification Report Form for Alert Classification Declaration During Drill on February 10, 2005
5790-102-02; Monticello Emergency Notification Report Form for Site Area Emergency Classification Declaration During Drill on February 10, 2005
5790-102-02; Monticello Emergency Notification Report Form for General Emergency Declaration During Drill on February 10, 2005
5790-104-04; Emergency Call List - Alert/Site Area/General Emergency completed for Drill on February 10, 2005
5790-803-01; EOF Reclassification Call-List for Site Area Emergency and General Emergency Reclassifications completed for Drill on February 10, 2005
5790-204-01; Monticello Off-Site Protective Action Recommendation Checklist completed for Drill on February 10, 2005

Corrective Action Program Documents:
CAP037084; Discrepancy between EFT/EVS boundaries in A.2-106 and Controller Expectations
CAP037089; Inappropriate Actions from Controllers during February 10, 2005 Drill

2OS1 Access Control to Radiologically Significant Areas

Documents and Procedures:
RWP 030454; Postings - Access to Overhead Platform from Contaminated Area Overhangs Clean Area; dated November 18, 2003
MNGP 5851; High Radiation Briefing Log; Revision 1
RWP 73; Radiologically Controlled Areas Excluding Locked HRAs and Highly Contaminated Areas; Revision 8
RWP 564; All Drywell - “A” Recirc Pump; Revison 0

Corrective Action Program Documents:
CA 022688; Radiation Protection/Fire Brigade Staffing Not Met for Approximately 15 Minutes; dated October 20, 2004
CAP 035125; Radiation Protection/Fire Brigade Staffing Not Met for Approximately 15 Minutes; dated October 5, 2005
CAP 035854; CE011159 Condition Evaluation Was Inadequately Performed; dated November 19, 2004
CAP 036252; Workers Accessed Scaffolding Prior to RP Survey; dated December 14, 2004
CAP 037618; NRC Questions Contaminated Grating Area Posting Above Uncontaminated Area; dated March 9, 2005
CAP 037506; Individual Violated RWP Locked HRA Requirements at the Drywell; dated March 6, 2005
CAP 037518; Potential Regulatory Significant Issue Not Communicated to the Outage Control Center; dated March 7, 2005
CAP 037742; Two Workers Failed to Log Into Timekeeping Prior to Entering Drywell Locked HRA Boundary; dated March 12, 2005
CAP 037744; Individual Received Dose Alarm; dated March 13, 2005
CAP 037780; Personnel Contamination - 500 cpm on Face While Mopping and Decontaminating Refuel Floor; dated March 14, 2005
CAP 037845; Turbine Overhaul Personnel Reaching Over Contamination Boundary - Repeat; dated March 15, 2005
CAP 037847; Condensate/Feedwater Valve Breeched Without Notifying RP; dated March 16, 2005
CAP 037896; Individual Did Not Follow Requirements on RWP for Opening Contaminated Systems; dated March 17, 2005
CAP 037903; NRC Question - Did Workers Receive Pre-Entry HRA Brief; dated March 17, 2005
CAP 037936; Unattended and Unlabeled Yellow Bag of Tools on Clean Side of Step-Off Pad; dated March 18, 2005

2OS2 As Low As Is Reasonably Achievable Planning And Controls

Documents and Procedures:
4AWI-08.04.08; ALARA Plan; Revision 6
MNGP 1123; Portable Fire Extinguishers Monthly Test; Revision 50
MNGP 1123; Portable Fire Extinguishers Monthly Test; Revision 52
MNGP 1123; Portable Fire Extinguishers Monthly Test; Revision 53
RP 50515; Perform Leak Rate Testing in Drywell; Revision 0

RWP 50500; Perform RP Surveys; Revision 0
RWP 50502; Operations General Entry to Drywell; Revision 0
RWP 50507; ALARA Efforts In Drywell and Shielding; Revision 0
RWP 50520; Perform Nozzle ISI and Insulation Work; Revision 1
RWP 50522; Breech, Repair and Modify Seat on MSIV "A" and"C"; Revision 0
RWP 50523; Scaffold Installation and Removal; Revision 0
RWP50563; Remove/Install Interferences for No. 11 Recirc Motor/Pump Repair and Replacement; Revision 1
RWP 50564; No. 11 Recirc Pump Repair/Replace; Revision 0

Corrective Action Program Documents:
CAP 036171; Actual Dose Received Was Not Within +/- 25 Percent of Dose Estimate for RWP 178; dated December 9, 2004
CAP 036318; ALARA: NRC Resident Questioned the Location of Fire Extinguisher in "A" RHR Room HRA; dated December 16, 2005
CAP 036613; Work Order Completed Without RWP Being Assigned; dated January 11, 2005
CAP 037659; ALARA - Exposure Greater Than 125 Percent of Prorated Estimate for Drywell Scaffolding; dated March 10, 2005
CAP 037827; ALARA - Exposure Greater Than 125 Percent of Prorated Estimate for CRD Rebuild; dated March 15, 2005
CAP 037839; ALARA - Actual Exposure Greater Than 125 Percent of Estimated for Valve Work in "A" RHR Room; dated March 15, 2005

Work Orders:
WO 0403292; Work Order Disassemble, Remove, Store and Reinstall No. 11 Recirc Pump Motor Stand and Stuffing Box Assembly; Attachment 1

4OA2 Identification and Resolution of Problems

Corrective Action Program Documents:
CAP036547; Initial Operability Screening for CAP035964 did not Consider Single Failure Criteria (NRC Identified)
CAP036644; NRC Walkdown of Fire Zone 1F (Torus Area) Identified VArious Housekeeping Issues (NRC Identified)
CAP036784; NRC Resident Informed of Wrong CDF Associated with 2R Transformer (NRC Identified)
CAP036915; NRC Question on Crystal River Event #41362 (NRC Identified)
CAP036953; Updating of Technical Manual and Drawing for Feedwater Check Valve Done Incorrectly (NRC Identified)
CAP037092; NRC Question Concerning Review of RCIC Procedure Change - Station Review (NRC Identified)
CAP037351; NRC Questions Exclusion of Limited Stroke Time (LST) for Valve Non-Safety Related Stroke Direction (NRC Identified)
CAP037663; Housekeeping, Mop Bucket Found Tied to CGCS Piping (NRC Identified)
CAP038185; Possibility of Having an Unsearched Individual Enter Protected Area Raised (NRC Identified)

4OA3 Event Follow-up

Documents and Procedures:
3195; Event Notification Worksheet for Trip of "A" Reactor Protection System (RPS) Motor Generator (MG) Set Caused "A" Group 2 Isolation and Auto Start of the Standby Gas Treatment System on February 24, 2005
Licensee Event Report 50-263/2004-003; High Pressure Coolant Injection System Declared Inoperable Due to Loose Oil Plug

Corrective Action Program Documents:
CAP037306; Failure of No. 11 RPS MG Set Causes Emergency Safeguards Features (ESF) Actuation

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AWI	Administrative Work Instruction
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CRD	Control Rod Drive
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Core Injection
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report
LER	Licensee Event Report
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
LORT	Licensed Operator Requalification Training
MNGP	Monticello Nuclear Generating Plant
MOV	Motor-Operated Valve
MRFF	Maintenance Rule Functional Failure
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NMC	Nuclear Management Company
OWA	Operator Workaround
PARS	Publicly Available Records
PI	Performance Indicator
PM	Planned or Preventative Maintenance
RA	Risk Assessment
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RP	Radiation Protection
RPT	Radiation Protection Technician
RWP	Radiation Work Permit
SBGT	Standby Gas Treatment
SBLC	Standby Liquid Control
SDP	Significance Determination Process
SRO	Senior Reactor Operator
SSC	Structures, Systems, and Components
TIP	Transversing Incore Probe
TS	Technical Specification

LIST OF ACRONYMS USED

URI	Unresolved Item
USAR	Updated Safety Analysis Report
Vdc	Volts Direct Current
WO	Work Order