April 27, 2007

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

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT
NRC INTEGRATED INSPECTION REPORT 05000263/2007002

Dear Mr. Conway:

On March 31, 2007, 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 3, 2007, 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 NRC-identified and two self-revealed findings of very low safety significance, of which three involved a violation of NRC requirements. However, because these violations were of very low safety significance and because the issues were entered into the licensee's corrective action program, the NRC is treating the three violations as non-cited violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Monticello Nuclear Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), 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 05000263/2007002
w/Attachment: Supplemental Information

cc w/encl: M. Sellman, President and Chief Executive Officer
D. Cooper, Senior Vice President and Chief
Nuclear Officer
Manager, Nuclear Safety Assessment
J. Rogoff, Vice President, Counsel, and Secretary
Nuclear Asset Manager, Xcel Energy, Inc.
State Liaison Officer, 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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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:	05000263/2007002
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello, Minnesota
Dates:	January 1 through March 31, 2007
Inspectors:	S. Thomas, Senior Resident Inspector L. Haeg, Resident Inspector J. Neurauter, Senior Reactor Inspector T. Bilik, Reactor Inspector S. Sheldon, Reactor Engineer T. Go, Health Physicist J. Cassidy, Health Physicist
Observers:	N. Feliz Adorno, Reactor Engineer
Approved by:	B. Burgess, Chief Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

Inspection Report 05000263/2007002; 01/01/2007- 03/31/2007; Monticello Nuclear Generating Plant. Adverse Weather, Operability Evaluation, Identification and Resolution of Problems, and Event Follow-up.

This report covers a three month period of baseline resident inspection and announced baseline radiation protection and inservice inspections. The inspections were conducted by Region III reactor inspectors, health physics 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. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was self-revealed when, due to the licensee's inadequate preparation to address the potential loss of the station heating boiler prior to the onset of extreme cold outside ambient temperatures, a temporary heating boiler could not be expeditiously installed subsequent to the loss of the station heating boiler on February 16, 2007. The licensee entered this issue into their corrective action program and performed a condition evaluation to determine specific items required to ensure that the installation of a temporary heating boiler could be performed, if needed, without jeopardizing plant operation. No violation of regulatory requirements occurred.

  The inspectors determined that the finding was more than minor because it affected the procedure adequacy attribute of the Initiating Events cornerstone objective of limiting those events that upset plant stability and challenge critical safety functions during power operations. The finding was of very low safety significance because the finding: (1) was not associated with the likelihood of a primary or secondary system loss of coolant accident (LOCA) initiation; (2) did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation systems would be unavailable; and (3) was not associated with a fire or flood. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having resource components, and involving aspects associated with having adequate and available facilities and equipment. (Section 1R01)

- Green. A self-revealed finding of very low safety significance was identified for a violation of 10 Code of Federal Regulations (CFR) 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," when the Division II essential 4160 Vac Bus 16 was lost, resulting in a reactor protection system (RPS) trip, scram, and Group II containment isolation on March 17, 2007. Specifically, while in Mode 5 refueling with Division I equipment protected, the loss of Bus 16 occurred due to the licensee's failure to fully evaluate the scheduling and operational impact of removing a potential transformer (POT) drawer associated with isolation activities for the 1AR transformer while Bus 16 was energized. After the loss of Bus 16, the licensee took immediate corrective actions, including: restoring plant equipment alignment, placing all electrical clearance orders/isolations on hold pending independent reviews, and generating corrective action program (CAP) document 01082734 to review the root cause of the event.

The inspectors determined that the finding was more than minor because it affected the procedure quality attribute of the Initiating Events cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown operations. The inspectors evaluated the finding using IMC 0609, Appendix G, Phase 1 Screening, and determined that Checklist 8, "Boiling Water Reactor (BWR) Cold Shutdown or Refueling Operation Time to Boil > 2 Hours: reactor coolant system (RCS) Level < 23' Above Top of Flange," applied. However, because all qualitative criteria within the Core Heat Removal, Inventory Control, Power Availability, and Containment guidelines were met; because the finding did not meet the Checklist 8 criteria for Phase 2 or Phase 3 quantitative analysis (the finding did not: increase the likelihood of a loss of RCS inventory; degrade the licensee's ability to terminate a leak path or add RCS inventory when needed; significantly degrade the licensee's ability to recover decay heat removal once it was lost; or involve only one or less safety relief valves available to establish a heat removal path to the suppression pool due to the vessel head being removed); and because no event occurred that could be characterized as a loss of control as listed in Table 1 of IMC 0609, Appendix G, the finding was considered to be of very low safety significance. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having work control components, and involving aspects associated with the failure to appropriately coordinate the work activity by incorporating actions to address the operational impact of the testing. (Section 4OA3.3)

Cornerstone: Mitigating Systems

- Green. An inspector-identified finding of very low safety significance was identified for a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," when the licensee failed to provide written instructions for the compensatory actions denoted in Operability Recommendation (OPR) 01076631-03 associated with a degraded condition for the 14 emergency service water (ESW) pump and equipment supported by that pump. Upon being notified of these deficiencies by the inspectors, the shift manager implemented procedure change requests for the required procedure revisions, prepared an Operations Memorandum which discussed the limitations imposed by the OPR, and entered the issue into the corrective action program.

  The finding was more than minor because the performance deficiency affected the procedure quality attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance because the finding did not represent a loss of system safety function nor an actual loss of safety function of a single train for more than its Technical Specification allowed outage time. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having resource components, and involving aspects associated with having complete, accurate and up-to-date procedures. (Section 1R15)

- Green. An inspector-identified finding of very low safety significance was identified for a violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” when the licensee failed to maintain written instructions which adequately verified that sufficient ESW flow was supplied to the ‘B’ residual heat removal (RHR) room to maintain the operability of the 12 core spray (CS) pump, 14 RHR pump, and the ‘B’ RHR room cooler under worst case ESW system operating conditions. Upon being notified of the issue by the inspectors, the licensee performed an apparent cause evaluation to evaluate the issue. As a result of the apparent cause evaluation, several corrective actions were developed to correct procedural and equipment deficiencies associated with the ‘B’ ESW system.

  The finding was more than minor because the performance deficiency affected the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance because the finding did not represent a loss of system safety function nor an actual loss of safety function of a single train for more than its Technical Specification allowed outage time. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having resource components, and involving aspects associated with maintaining long-term plant safety by the maintenance of design margins and the minimization of long-standing equipment issues. (Section 4OA2.4)

B. <u>Licensee-Identified Violations</u>

None.

REPORT DETAILS

Summary of Plant Status

Monticello operated at full power during the assessment period except for brief down-power maneuvers to accomplish rod pattern adjustments and to conduct planned surveillance testing activities with the following exceptions:

- On January 10, 2007, the reactor scrammed from full power. The licensee determined that the cause of the scram was a turbine control valve system failure, which caused the control valves to open fully; and as a result, reactor pressure dropped, the main steam isolation valves closed, and the reactor scrammed. All safety systems functioned as required subsequent to the scram. Following the implementation of required corrective actions to address the turbine control valve system failure, the reactor was restarted on January 23, 2007. The generator was connected to the grid on January 24, 2007, with full power achieved on January 25, 2007. This event is further discussed in Section 4OA3 of this report.

- On March 4, 2007, fuel cycle coast-down began. On March 14, 2007, the reactor was shut down for a planned refueling outage. The reactor remained shut down for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors performed a review of the licensee's preparations to respond to the loss of the station heating boiler during periods of extreme cold outside ambient temperature conditions. The inspectors focused on plant specific design features and the ability to implement procedures for responding to and mitigating the effects of the loss of the station heating boiler under these conditions. Inspection activities included: a review of the licensee's abnormal procedure for addressing the loss of the station heating boiler; the procedure for installing a temporary heating boiler; the most current temporary modification package associated with temporary heating boiler installation; and several emergent station heating boiler maintenance issues. The inspectors also observed licensee implementation of the loss of station heating boiler abnormal procedure during a period of time when the boiler was shut down for several hours, coinciding with an outside ambient temperature well below freezing.

The inspectors evaluated overall site preparedness for extreme cold weather for a total of one sample.

b. Findings

Introduction: A finding of very low safety significance was self-revealed when Monticello licensee staff demonstrated that the site was inadequately prepared to cope with the long-term loss of the station heating boiler prior to the onset of extreme cold outside ambient temperatures.

Description: During January and early February 2007, several minor maintenance issues resulted in brief losses of the station heating boiler. On each of these occasions, the station heating boiler was restored with little or no impact on reactor or turbine building heating or ventilation systems. On February 16, 2007, at 9:18 p.m., the licensee experienced a sustained loss of the station heating boiler (approximately nine hours), requiring entry into Abnormal Procedure C.4-B.08.03.A, “Loss of Heating Boiler,” Revision 3. Because outside air temperatures were well below freezing, plant operators expeditiously isolated secondary containment in accordance with the procedure to maintain reactor building temperatures and protect plant equipment. Additional actions taken included the utilization of several portable space heaters and the use of temporary ventilation to prevent average steam chase temperature from exceeding 165°F (an administrative limit; if temperature could not be maintained below the limit, a rapid plant power reduction would have been required). In parallel with trouble-shooting efforts to restore the station heating boiler, the licensee pursued the acquisition and installation of a temporary heating boiler. Approximately nine hours after the station heating boiler was lost, the licensee restored the station heating boiler to operation, enabling exit from C.4-B.08.03.A, and allowing for restoration to normal winter heating and ventilation to the reactor and turbine buildings.

Monticello Abnormal Procedure C.4-B.08.03.A provides directions for plant staff to accomplish in the event of a sustained station heating boiler failure. Specifically, this procedure discusses: manual isolation of secondary containment; initiation of strict plant status control; control of steam chase temperature; isolation and drainage of heating coils; the use of portable heaters; and specific temperature limits for the steam chase, 11 and 12 emergency diesel generator (EDG) rooms, recombiner room, intake structure, diesel fire pump house, and the radioactive waste shipping building. Specific guidance associated with the installation of a temporary heating boiler is located in Step 4 of C.4-B.08.03.A. Step 4 states, “IF heating boiler cannot be restored, THEN notify the System Engineer, to coordinate temporary boiler hook up.” A note preceding Step 4 states that “the time required to install a temporary boiler is 12 hours.” Further clarification of the note is provided in the procedure’s bases section, which states “a temporary boiler can be in place and running in 12 hours.”

The licensee attempted to use existing guidance to acquire and install a temporary heating boiler. This guidance consisted of Temporary Modification (TMOD) 04-016, “Installation of a Temporary Auxiliary Heating Boiler,” and Procedure 8047, “Temporary Heating Boiler Installation.” The licensee quickly discovered that insufficient preparation and planning had occurred to facilitate the installation of an operational temporary heating boiler within 12 hours. Specific issues encountered included:

- the temporary heating boiler installation procedure did not provide sufficient detail to facilitate the expedited installation of a temporary heating boiler;

- plant procedures did not document a specific vendor or the specific type of temporary heating boiler to be used;
- the work packages needed to install the temporary heating boiler had not been prepared in advance and required significant engineering, maintenance, and planning resources to prepare; and
- when the temporary heating boiler arrived on-site, the licensee determined that the vendor had not supplied the necessary piping to connect to the plant heating boiler feedwater piping and that a tie-in to existing plant fuel oil systems would be required. TMOD 04-016, developed as a recurring TMOD to be used as the basis document for Procedure 8047 and to support the installation and removal of a temporary heating boiler, could not be used as written. Since the TMOD assumed that the temporary heating boiler vendor supplied the necessary piping to connect to the station heating boiler feedwater system and that the temporary heating boiler would not require a tie-in to existing plant fuel systems, no preparations had been made to support temporary tie-ins to these plant systems.

Subsequent to the inspectors questioning the licensee's ability to install a temporary heating boiler within 12 hours of it being requested, the licensee informed the inspectors that 12 hours was unrealistic given that existing instructions in Procedure 8047, which directs the generation of installation work orders, did not list approved temporary heating boiler vendors.

<u>Analysis</u>: The inspectors determined that the inadequate preparation to address the potential loss of the station heating boiler prior to the onset of extreme cold outside ambient temperatures was a performance deficiency warranting significance evaluation in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on June 22, 2006. The inspectors concluded that the finding was more than minor because it affected the procedure adequacy attribute of the Initiating Events cornerstone objective of limiting those events that upset plant stability and challenge critical safety functions during power operations. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having resource components, and involving aspects associated with having adequate and available facilities and equipment.

Utilizing the Phase 1 Screening Worksheet, per IMC 0609, "Significance Determination Process," the inspectors determined that this performance deficiency impacted the Initiating Event cornerstone because it constituted a transient initiator contributor. The inspectors answered "no" to the Phase 1 Initiating Event questions because the finding: (1) was not associated with the likelihood of primary or secondary system loss of coolant accident (LOCA) initiation; (2) did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation systems would be unavailable; and (3) was not associated with a fire or flood. Therefore, this finding was considered to be of very low safety significance (Green).

<u>Enforcement</u>: Procedure C.4-B.08.03.A, "Loss of Heating Boiler," Revision 3, was not required by 10 Code of Federal Regulations (CFR) Part 50, Appendix B; therefore, no violation of regulatory requirements occurred. This issue was considered to be a finding of very low safety significance (FIN 05000263/2007002-01). The licensee entered the issues into their corrective action program as CAP 01078606, and performed a condition

evaluation to determine specific items required to ensure that the installation of a temporary heating boiler could be performed, if needed, without jeopardizing plant operation.

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 system equipment. The inspectors reviewed equipment alignment to identify any discrepancies that could impact the function of the system and potentially increase risk. 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 a review of the licensee's procedures, verification of equipment alignment, and an observation of material condition, including operating parameters of equipment in-service.

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

- Division I residual heat removal (RHR) system with Division II RHR system out-of-service for planned maintenance;
- high pressure coolant injection (HPCI) system with reactor core isolation cooling (RCIC) out-of-service for planned maintenance; and
- shutdown cooling Division I protected equipment post-shutdown.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete walkdown of equipment for one risk significant mitigating system. The inspectors walked down the system to review electrical equipment line-ups, component labeling, component and equipment cooling, 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 CAP database to ensure that any system equipment alignment problems were being identified and appropriately resolved.

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

- 125 Vdc batteries and distribution system.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Zone Walkdowns (71111.05Q)

a. Inspection Scope

The inspectors walked down risk significant fire areas to assess fire protection requirements. The inspectors reviewed each area to ensure that 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 potential to impact equipment which could initiate or mitigate a plant transient. The inspection activities included 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.

The inspectors selected the following areas for review for a total of 11 samples:

- Fire Zone 2-F, (main steam chase);
- Fire Zone 12-A, (lower 4KV bus area, 11, 13 and 15);
- Fire Zone 12-D, (mechanical vacuum pump (MVP) room);
- Fire Zone 12-E, (steam air ejector room);
- Fire Zone 13-A, (lube oil storage tank room);
- Fire Zone 15-E, (diesel oil pump house);
- Fire Zone 19-C, (feedwater pipe chase);
- Fire Zone 21-C, (radwaste shipping building);
- Fire Zone 21-B, (radwaste trash compactor area);
- Fire Zone 21-A, (radwaste control room); and
- Fire Zone 40, (screen house).

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors performed an annual review of flood protection barriers and procedures for coping with internal flooding. The inspection focused on determining whether flood mitigation plans and equipment were consistent with design requirements and risk analysis assumptions. The inspection activities included a review and walkdown to assess design measures, seals, drain systems, contingency equipment condition and availability of temporary equipment and barriers, performance and surveillance tests, procedural adequacy, and compensatory measures.

The inspectors selected the following equipment for a total of one sample:

- No.16 250 Vdc battery and D100 125/250 Vdc distribution panel areas after it was identified that a drain line in the emergency filtration treatment (EFT) building was frozen and plugged due to extreme cold weather on February 5, 2007.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors performed an annual review of the licensee’s testing of heat exchangers. The inspection focused on potential deficiencies that could mask the licensee’s ability to detect degraded performance, identification of any common cause issues that had the potential to increase risk, and ensuring that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspection activities included a review of the licensee’s observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing criteria.

The inspectors selected the following equipment for a total of one sample:

- 12 EDG emergency service water (ESW) heat exchangers.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

From March 19 to March 23, 2007, 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 on-site 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 HPCI water side discharge piping welds, weld W-20 (pipe-to-elbow) and weld W-23 (elbow-to-pipe);
- Magnetic particle examination (MT) of a HPCI steam line support (H-3); and
- Visual examination (VT-3) of a HPCI steam line support (H-3, double spring/4 lug hanger).

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 records:

- Visual examination records of the upper surveillance sample holder (at 30°). During the examination, the licensee identified a distorted auxiliary bracket welded to the surveillance sample holder bracket (evaluated and found to be acceptable per ASME Code).
- A pressure boundary weld for a Code Class 1 system which was 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 Feedwater Loop A, Valve FW-97-1, seal weld for hinge pin plug replacement.
- Piping system ISI-related issues 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 issues. 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 reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

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 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.

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

- a training crew during a simulator scenario that included a loss of shutdown cooling and subsequent loss of coolant accident, which resulted in entry into the emergency operating procedures, an unusual event emergency action level classification, and operation of alternate means to provide shutdown cooling and reactor vessel level control.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed various areas and 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.

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

- motor-operated valve (MOV) limit and torque switch setting issues because many MOVs are designated as risk significant under the Maintenance Rule and there were several switch setting issues over the past two years;
- a function-oriented review of the ‘Structures’ maintenance rule system because it was designated as risk significant under the Maintenance Rule and was classified as a 10 CFR 50.65 (a)(1) system during the inspection period; and
- an issue/problem-oriented review of post-maintenance electrical breaker racking performance for various safety-related breakers. Breaker operation was reviewed for those designated as risk significant under the Maintenance Rule.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed maintenance activities to evaluate 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 effectiveness of planning and control for emergent work activities. The inspection activities included verification that; licensee RA procedures were followed and performed appropriately for routine and emergent maintenance; RAs for the scope of work performed were accurate and complete; necessary actions were taken to minimize the probability of initiating events; and activities that ensured the functionality of mitigating systems and barriers were completed. Additionally, the assessment included an evaluation of external factors, the licensee's control of work activities, configuration control, and appropriate consideration of baseline and cumulative risk.

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

- electronic pressure regulator failure;
- main steam turbine control valve actuator support modification;
- emergent work schedule planning during elevated grid risk conditions;
- loss of plant heating boiler on February 15, 2007; and
- replacement of reactor water clean-up (RWCU) valve RC-41-2.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations for components associated with mitigating systems or barrier integrity to ensure that operability was properly justified and that the component or system remained available. The inspection activities included a review of the technical adequacy of each operability evaluation, its impact on TSs, and the significance of each evaluation to ensure that adequate justifications were documented, and that risk was appropriately assessed.

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

- CAP 01070238 (no carbon in EFT test canister removed 1/4/07);
- CAP 01071663 (bent hanger rod on main steam pressure averaging manifold);
- CAP 01073031 (abnormal behavior of control rod drive (CRD) 42-11);
- CAP 01071697 (spurious self-actuation of safety relief valve (SRV) 'F' occurs following scram);
- CAP 01076631 (P-111D, 14 ESW flow to RHR 'B' room less than required band);
- CAP 01079794 (potential leakage past MO-2029 and MO-2030); and
- CAP 01082169 (scram setpoint calculation doesn't account for steam separator differential pressure).

b. Findings

Introduction: The inspectors identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance (Green) for failing to provide written instructions for the compensatory actions denoted in Operability Recommendation (OPR) 01076631-03 associated with a degraded condition for the 14 ESW pump and equipment supported by that pump. The issue was considered to be NRC-identified because two days after the approval of the OPR, inspectors identified that procedure changes delineated as compensatory actions to restore operability still had not been implemented. Additionally, the operating crew was unaware of the operating restrictions outlined in the OPR.

Description: Subsequent to the performance of 0255-11-III-8 [14 ESW comprehensive pump and valve test] on February 9, 2007, and the performance of 0255-11-III-4 [ESW quarterly pump test] on February 13, 2007, the licensee declared 14 ESW inoperable based on a calculated flow rate of less than 36 gpm to the 'B' RHR room. Based on an ESW inlet temperature that was significantly below the TS requirement of 90°F, the on-shift operating crew determined that there was a reasonable expectation that the 14 ESW could perform its required function at the reduced flow rate. The 14 ESW system was declared operable and an OPR was requested to further evaluate the issue.

Operability Recommendation 01076631-03 provided justification that the 14 ESW system was operable but degraded and was approved on February 20, 2007, with a compensatory action of procedurally restricting the 14 ESW inlet temperature to 85°F. On February 22, 2007, inspectors reviewed OPR 01076631-03. As part of this review, the inspectors attempted to verify that the compensatory measure taken to restore operability, as stated in the OPR had been implemented. The compensatory actions consisted of modifying Procedure A.6, “Acts of Nature,” and Procedure B.08.01.04-05, “Emergency Service Water System Operation,” to reflect the new maximum inlet temperature of 85°F. The inspectors discovered that no actions had been completed to implement the procedure changes and that none of the licensee operators which comprised the on-duty crew were aware of the restricted operating limits imposed by OPR 01076631-03. Upon being notified of these deficiencies by the inspectors, the shift manager implemented procedure change requests for the required procedure revisions, prepared an Operations Memorandum which discussed the limitations imposed by the OPR, and entered the issue into the corrective action program.

Analysis: The inspectors determined that failing to provide written instructions for the compensatory actions denoted in OPR 01076631-03 associated with a degraded condition for the 14 ESW pump and equipment supported by that pump was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issues Disposition Screening,” issued on June 22, 2006. The finding was more than minor because it affected the attribute of procedure quality, which could have impacted the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having resource components, and involving aspects associated with having complete, accurate and up-to-date procedures.

Utilizing the Phase 1 Screening Worksheet, per IMC 0609, “Significance Determination Process,” the inspectors determined that the finding did not represent a loss of system safety function nor an actual loss of safety function of a single train for more than its TS allowed outage time. Therefore, this finding was considered to be of very low safety significance (Green).

Enforcement: Title 10 CFR 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances. Contrary to this requirement, on February 20, 2007, the licensee approved and implemented OPR 01076631-03 without providing written instructions for the compensatory actions denoted in the OPR. The required procedure changes were initiated on February 22, 2007, after the licensee was informed by the inspectors of the deficiency. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program as CAP 01078881, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000263/2007002-02).

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 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 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.

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

- reactor building crane upgrade to support spent fuel cask loading activities.

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 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 updated safety analysis report (USAR) design requirements.

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

- 'A' EFT filter efficiency and leak testing following planned maintenance;
- scram discharge volume level switch (LS-7428F) testing following replacement;
- MO-2101, RCIC outboard torus suction valve testing following planned maintenance;
- 11 RWCU pump testing following seal replacement;
- RHR Loop 'A' quarterly pump and valve test following planned maintenance on 13 RHR pump motor breaker; and

- 12 Emergency Diesel Generator (EDG) testing following the 12 year periodic maintenance.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Forced Outage (January 10, 2006, to January 24, 2006)

a. Inspection Scope

The inspectors evaluated activities for a forced outage that began on January 10, 2006, and ended on January 24, 2006. The inspectors reviewed outage 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 following represents a partial list of the major outage activities the inspectors reviewed/observed, all or in part:

- control room operator response to a reactor scram and plant cooldown without the availability of the normal heat sink (main condenser);
- review of outage plans and the ready-backlog;
- control room turnover meetings and selected pre-job briefings;
- repair activities associated with the main steam turbine control valve actuator support modification;
- review of the licensee’s root cause report, extent of condition evaluation, and scram report; and
- startup and heatup activities, including criticality, feed pump startup, main turbine generator startup and synchronization, and elements of power escalation to full power.

b. Findings

No findings of significance were identified.

.2 Refueling Outage

a. Inspection Scope

The inspectors evaluated outage activities for a refueling outage that began on March 14, 2007, at 00:06, and continued through 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.

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

- reactor shutdown and cooldown;
- review of the outage plan;
- control room turnover meetings and selected pre-job briefings;
- steam dryer and separator removal;
- refueling activities;
- replacement of 12 recirculation pump;
- torus coating work;
- main steam isolation valve (MSIV) work; 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 structures, systems, and components were capable of performing their intended safety function. Activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a system, structure, or component could impose on the unit if the condition was left unresolved. The inspection activities included a review for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, TS applicability, impact of testing relative to performance indicator (PI) reporting, and evaluation of test data.

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

- turbine generator operational tests (routine surveillance);
- MSIV exercise test (in-service test);
- alternate shutdown system functional test for Division II RHR, RHR service water, ESW switches and control room annunciator for alternate shutdown system master transfer switch (routine surveillance);
- control rod drive 42-11 scram insertion time testing after identification of increased settling times (routine surveillance); and
- core spray (CS) shutdown valve operability test (routine surveillance).

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's occupational exposure control cornerstone performance indicators (PIs) to determine whether or not the conditions surrounding the PIs had been evaluated and identified problems had been entered into the corrective action program for resolution.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed radiologically significant work areas within the plant and reviewed work packages associated with these areas to determine if radiological controls including surveys, postings, and barricades were acceptable. The following four areas were reviewed and were controlled as radiation areas, high radiation areas (HRAs), or airborne areas:

- refuel floor in-service inspections (ISI) activities;
- hydro lazing of reactor vessel;
- drywell nozzle ISI; and
- torus repair and internal coating.

The inspectors reviewed the radiation work permits (RWPs) and WOs used to access these four 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 assess whether they were aware of the actions required when their electronic dosimeters noticeably malfunctioned or alarmed.

The inspectors walked down and performed radiological survey measurements in these four 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.

The inspectors reviewed RWPs for any airborne radioactivity areas that existed during the inspection 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. There were no airborne radioactivity areas in the plant during this outage. 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 four inspection samples

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, Licensee Event Reports, and Special Reports related to the access control program to verify that identified problems were entered into the corrective action program for resolution.

The inspectors reviewed 20 CAP reports related to access controls and two high radiation area radiological incidents (non-performance indicators identified by the licensee in high radiation areas 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.

The inspectors evaluated the licensee's process for problem identification, characterization, and prioritization to assess whether problems were entered into the corrective action program and resolved. For repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, the inspectors assessed whether the licensee's self-assessment activities were capable of identifying and addressing these deficiencies

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.

These reviews represented four inspection samples

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews and Review of Work Practices in Radiologically Significant Areas

a. Inspection Scope

The inspectors evaluated the licensee's radiological controls, job coverage and radiation worker practices for the following jobs:

- torus recoat maintenance and the associated support work;
- drywell in service inspection and nozzle windows and main steam relieve valves maintenance and associated support;
- hydrolazing and vessel preparations; and
- drywell general support and associated work.

This review included radiation survey information to support these work activities; the radiological job requirements; the adequacy of the information exchanges during pre-job briefings; and the access control provisions for these areas was assessed for conformity with TS and with the licensee's procedures.

Job performance was observed to determine if radiological conditions in the work areas were adequately communicated to workers through the pre-job briefings and area postings. The inspectors also evaluated the adequacy of the oversight provided by the radiation protection staff including the performance of radiological surveys and air sampling, the work oversight provided by the radiation protection technicians (RPTs), and the administrative and physical controls used over ingress/egress into these areas.

The inspectors also reviewed the licensee's procedure and generic practices associated with dosimetry placement and the use of multiple whole body dosimetry for work in high radiation areas having significant dose gradients for compliance with the requirements of 10 CFR 20.1201(c) and applicable industry guidelines. Additionally, previously completed work in areas where dose rate gradients were subject to significant variation such as on the reactor nozzles activities were reviewed to evaluate the licensee's practices for dosimetry placement.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.5 High Risk Significant, High Dose Rate HRA and VHRA Controls

a. Inspection Scope

The inspectors held discussions with the Acting Radiation Protection 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.

The inspectors discussed with radiation protection (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.

The inspectors conducted plant walkdowns to verify the posting and locking of entrances to high dose rate HRAs, and very high radiation.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors assessed radiation worker performance with respect to stated radiation protection work requirements and assessed 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.

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 Radiation Protection Manager.

These reviewed represented two inspection samples.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician Proficiency

a. Inspection Scope

The inspectors evaluated RPT performance during job performance observations, with respect to radiation protection 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.

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.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

2OS2 As Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history and current exposure trends, along with ongoing and planned outage activities, in order to assess current performance and exposure challenges. This included reviewing the plant's current 3-year rolling average collective exposure and comparing the site's radiological exposure on a yearly basis for the previous three years.

The inspectors reviewed the outage work activities scheduled during the inspection period along with associated work activity exposure estimates, including the four work activities which were likely to result in the highest personnel collective exposures:

- torus recoat maintenance and the associated support work;

- drywell in service inspection and nozzle windows and main steam relieve valves maintenance and associated support;
- hydrolazing and vessel preparations; and
- drywell general support and associated work.

Procedures associated with maintaining occupational exposures ALARA and processes used to estimate and track work activity specific exposures were reviewed.

These reviews represented three inspection samples

b. Findings

No findings of significance were identified.

.2 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 selected the four-work activities of highest exposure potential.

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements, in order to determine if the licensee had established procedures, along with engineering and work controls, that were based on sound radiation protection principles, in order to achieve occupational exposures that were ALARA. This also involved determining if the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, or special circumstances.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the assumptions and bases for the current annual collective exposure estimate including procedures, in order to evaluate the licensee's methodology for estimating work activity-specific exposures and the intended dose outcome. Dose rate and man-hour estimates were evaluated for reasonable accuracy.

This review represented one sample.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Controls

a. Inspection Scope

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

- torus recoat maintenance and the associated support work;
- drywell in service inspection and nozzle windows and main steam relieve valves maintenance and associated support;
- hydrolazing and vessel preparations; and
- drywell general support and associated support work.

The licensee's use of engineering controls to achieve dose reductions was evaluated to assess whether 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 inspection sample.

b. Findings

No findings of significance were identified.

.5 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 developing contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry.

This review represented one sample.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

Radiation worker and RPT performance were observed during work activities being performed in radiation areas, airborne radioactivity areas, and HRAs that presented the greatest radiological risk to workers. The inspectors assessed whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and tools to be used, by utilizing ALARA low dose waiting areas, and by complying with work activity controls.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.7 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and special reports related to the ALARA program since the last inspection. The inspectors assessed the adequacy of the licensee's overall audit program's scope and frequency to meet the requirements of 10 CFR 20.1101(c).

Corrective action reports generated during the licensee's outage (RFO-24) that related to the ALARA program were selectively reviewed, and staff members were interviewed to verify that follow-up activities were being conducted in a timely manner commensurate with their importance to safety and risk using the following criteria:

- 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; and
- identification and implementation of effective corrective actions.

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies in problem identification and resolution were being addressed.

This review represented two inspection samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Initiating Events

Reactor Safety Strategic Area

a. Inspection Scope

The inspectors' review of PIs used guidance and definitions contained in Nuclear Energy Institute (NEI) Document 99-02, Revision 4, "Regulatory Assessment Performance Indicator Guideline," to assess the accuracy of the PI data. The inspectors' review included, but was not limited to, conditions and data from logs, licensee event reports (LERs), CAP documents, and calculations for each PI specified.

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

- Unplanned Scrams per 7000 Critical Hours, for the period of March 2006 through March 2007 (Initiating Events);
- Unplanned Scrams with Loss of Normal Heat Removal, for the period of March 2006 through March 2007 (Initiating Events); and
- Unplanned Power Changes per 7000 Critical Hours, for the period of March 2006 through March 2007 (Initiating Events).

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: 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 occurrence 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. <u>Findings</u>

No findings of significance were identified.

.2 <u>Daily Corrective Action Program Reviews</u>

a. <u>Inspection Scope</u>

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. <u>Findings</u>

No findings of significance were identified.

.3 <u>Annual Sample: Operable-But-Degraded/Non-Conforming Items - Non-Resolution during Refueling Outage 23</u>

a. <u>Inspection Scope</u>

On January 3, 2007, the licensee wrote CAP 01069830, titled "Eight OBN items will not be resolved during RFO23." The CAP identified that of the 13 outstanding operable-but-degraded (OBD) and operable-but-non-conforming (OBN) items, eight would not be resolved before the end of the next refueling outage beginning in mid-March 2007. Based on guidance contained in NRC Inspection Manual 9900, the licensee determined that extending resolution of the outstanding OBD/OBN items past the next refueling outage required documented and approved justification. The inspectors chose to perform an in-depth review of the licensee's bases for not resolving the outstanding OBD/OBN items, as well as assessing whether continued operation under the various circumstances was appropriate. During the review, the inspectors considered the overall safety and degradation significance, operational impact, and timing/resource aspects of the conditions.

Of the eight OBD/OBN items, the inspectors reviewed the following two CAP documents, both of which were not planned on being resolved before the end of the March 2007 refueling outage:

- CAP 0878626; MO-2374, main steam line drain outboard valve, shows dual indication when tested in closed direction; and
- CAP 0717451; low pressure coolant injection (LPCI) loop selection logic may not meet USAR break size detection requirement.

Assessment and Observations

The inspectors reviewed documentation and interviewed licensee personnel to determine the adequacy of the licensee's justification for not resolving the OBD/OBN items at the next available opportunity, i.e., prior to the end of the March-April 2007 refueling outage 23.

Regarding CAP 0878626, the licensee identified that in August 2005, outboard main steam line drain motor-operated valve MO-2374 showed dual indication when it was cycled closed during a surveillance test. The valve was declared inoperable due to its potential inability to close on a Group I main steam line isolation signal, and its inboard counterpart MO-2373, was closed and electrically isolated to meet TS. The valve (MO-2374) was determined to be OBN because it could not meet all design basis requirements. An apparent cause evaluation was performed and compensatory measures were taken. As a corrective action, WO 0507786 was written as a contingency to adjust the torque switch on the valve should a forced shutdown of the unit occur. During the inspector's review after the January 10, 2007, automatic shutdown as discussed in Section 1R20, the inspectors questioned whether WO 0507786 was performed. Licensee personnel stated that performance of the WO was considered during an outstanding WO review during the forced outage, but determined that leaving the inboard valve in its current configuration (closed and electrically isolated) would not pose an undue workaround for operations staff during plant startup. The decision was made to defer the WO and cancel it due to planned replacement of the valve during the March 2007 Refueling Outage 23. The inspectors determined that this course of action was appropriate for the circumstances.

Regarding CAP 0717451, the inspectors conducted interviews with engineering personnel and reviewed documentation associated with the condition. The CAP, generated in May 2004, concerned LPCI recirculation loop selection logic not meeting USAR break size detection requirements. The USAR described that the LPCI recirculation loop selection logic was designed to select the unbroken recirculation loop for break areas greater-than-or-equal-to 0.1 square feet during a postulated recirculation line (loop) break. This break, resulting in a differential pressure of approximately 15 inches of water column, was deemed to be overly conservative after a preliminary calculation was performed for a 2-year fuel cycle extension project. This calculation revealed that when applicable uncertainties were considered, the trip setting could exceed 27 inches of water column. The licensee determined that the logic's contribution to the operability of the LPCI recirculation loop selection function was OBN with respect to the USAR description. The inspectors determined that an outstanding license amendment request was in-process and that pending approval, changes to the USAR and Improved Technical Specifications (ITS) would be made after the plant operations review committee approved changes to the USAR and ITS. The inspectors determined that no adverse condition existed, and that completion of the March 2007 refueling outage should not necessarily be contingent upon approval of the amendment.

Overall, the inspectors concluded that although some OBD/OBN items were not resolved, documentation/evaluation justifying continued operation was either in-place, or scheduled to be completed before the end of refueling outage 23. The licensee was aware that documentation of justified operation would, in some cases, be conducted at-risk, i.e., the outcomes of the evaluations may result in requiring resolution before the end of refueling outage 23; however, the licensee believed that all OBD/OBN items would be evaluated and approved. Based on the inspector's review of these items, the licensee's actions appeared to be in accordance with guidance contained in NRC Inspection Manual 9900.

b. Findings

No findings of significance were identified.

4. Annual Sample: 'B' ESW System Testing Issues

a. Inspection Scope

Inspectors reviewed licensee actions to resolve issues associated with the recent inability of 'B' ESW system to provide the minimum system and component flows to maintain the operability of RHR and CS pumps in the 'B' RHR room. It should be noted that the 14 ESW pump provides flow through the 'B' ESW system. Recent occurrences where sufficient flow had not been obtained included:

- On January 26, 2007, during the performance of the 14 ESW quarterly pump and valve test, Revision 41, the reference flow required to calculate 14 ESW pump differential (124-128 gpm) could not be achieved;
- On February 9, 2007, during a partial performance of the 14 ESW comprehensive pump and valve test, 'B' ESW could not achieve the required 'B' RHR room flow of at least 36 gpm;
- On February 13, 2007, during the performance of the 14 ESW quarterly pump and valve test, Revision 41, 'B' ESW could not achieve the required 'B' RHR room flow of at least 36 gpm; and
- On February 13, 2007, during the performance of the 14 ESW quarterly pump and valve test, Revision 42, 'B' ESW could not achieve the required 'B' RHR room flow of at least 36 gpm.

Also included as part of this inspection was a review of the licensee's corrective action documents associated with this issue and the licensee's testing methodology, used in both the 14 ESW quarterly and comprehensive pump and valve tests, to verify that adequate ESW flow was present in the 'B' RHR room to maintain operability of the RHR and CS pumps.

b. Assessments and Observations

Introduction: The inspectors identified an NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance (Green) for failing to have written procedures which adequately verify that sufficient ESW flow was supplied to the 'B' RHR room to maintain the operability of the 12 CS pump, 14 RHR pump, and the 'B' RHR room cooler under worst case ESW system operating conditions. The issue was considered to be NRC-identified because, subsequent to reviewing actual flow distribution data for ESW-cooled components in the ‘B’ RHR room, the inspectors discovered that the minimum acceptable surveillance indicated ‘B’ RHR ESW flow would have been insufficient to supply the required actual ESW flow required by the ‘B’ RHR room cooler.

Description: On February 20, 2007, the licensee approved OPR 01076631-03 to document an operable but degraded condition associated with the ‘B’ ESW system. The inspectors reviewed this OPR and the inspection results are documented in Section 1R15 of this report.

To evaluate past operability of ‘B’ RHR room components, the inspectors reviewed ESW data obtained from past performances of 14 ESW quarterly pump and valve tests (2nd Quarter 2005 through 1st Quarter 2007). This procedure states that ESW flow to the ‘B’ RHR room must be at least 36 gallons per minute (gpm) (indicated). The bases section of the procedure provides the following clarification:

COMPONENT	MINIMUM COOLING WATER FLOW REQUIRED
V-AC-4 (‘B’ RHR Room Cooler)	26 gpm at 90°F to maintain RHR room temperatures less than 140°F
12 RHR Pump Motor	Does not require cooling water
14 RHR Pump Motor	4 gpm
12 CS Pump Motor	2 gpm
Flow Instrument Inaccuracies	4 gpm
TOTAL REQUIRED FLOW (indicated)	36 gpm

Based on indicated ESW flow of 36 gpm, incorporating instrument inaccuracies, the actual flow to the ‘B’ RHR room could be as low as 32 gpm. Utilizing the documented flow distribution to the cooled components in that room, actual ESW flow rates would be as follows: 23.1 gpm (72.2%) to V-AC-4; 4.1 gpm (12.8%) to the 14 RHR pump motor; and 4.8 gpm (15.0%) to the 12 CS pump motor. Based on this information, the inspectors determined that the current requirement of 36 gpm (indicated) was inadequate and that the 14 ESW quarterly and comprehensive pump and valve tests should require an indicated ‘B’ RHR room ESW flow rate of at least 40 gpm, assuming current licensee testing assumptions, to ensure that V-AC-4 received 26 gpm (actual). The inspectors also determined that these procedures were not conservative because they did provide a flow rate correction factor to account for reduced 14 ESW pump performance during design basis minimum river level.

The inspectors focused their review of the 14 ESW quarterly pump and valve testing data (2nd Quarter 2005 through 1st Quarter 2007) on determining actual ESW flow rates delivered to V-AC-4. Operability Recommendation 01076631-03 concluded, in part, that V-AC-4 could perform its required function as long as it received an ESW flow rate of 20 gpm with an inlet temperature of 85°F. Although the inspectors determined that there were several quarterly tests where the actual ESW flow rate was less than 26 gpm, and additional quarterly tests where very little margin existed, there was only one quarter that was not bounded by the OPR. During that quarter, even though river temperature exceeded 85°F, adequate ESW flow existed to maintain V-AC-4 operability.

As part of the licensee's corrective action process, they performed an apparent cause evaluation to determine why 'B' ESW, on several occasions, was unable to provide minimum system and component flow, as required by surveillance procedures. As a result of the apparent cause evaluation, the licensee developed several corrective actions to address: the deficient ESW testing methodology and its impact on other surveillance procedures; ESW surveillance procedure adequacy; and 'B' ESW system improvements focused on regaining flow margin.

Analysis: The inspectors determined that failing to have written procedures which adequately verified that sufficient ESW flow was supplied to the 'B' RHR room to maintain the operability of the 12 CS pump and the 14 RHR pump under worst case ESW system operating conditions was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issues Disposition Screening," issued on June 22, 2006. The finding was more than minor because it affected the attribute of equipment performance, which could have impacted the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having resource components, and involving aspects associated with maintaining long term plant safety by the maintenance of design margins and the minimization of long-standing equipment issues.

Utilizing the Phase 1 Screening Worksheet, per IMC 0609, "Significance Determination Process," the inspectors determined that the finding did not represent a loss of system safety function nor an actual loss of safety function of a single train for more than its TS allowed outage time. Therefore, this finding was considered to be of very low safety significance (Green).

Enforcement: Title 10 CFR 50, Appendix B, Criterion V requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances. Contrary to this requirement, the licensee failed to maintain written procedures which adequately verified that sufficient ESW flow was supplied to the 'B' RHR room to maintain the operability of the 12 CS pump, 14 RHR pump, and the 'B' RHR room cooler under worst case ESW system operating conditions. Because the event was of very low safety significance and because the issue was entered into the licensee's corrective action program (CAP 01076631), this violation is being treated as an NCV, consistent with Section VI.A.1 of Enforcement Policy (NCV 05000263/2007002-03).

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report 50-263/2007-001-00: "Reactor Scram due to Turbine Control Valve Housing Support Failure"

On January 10, 2007, the reactor shut down automatically from 90 percent power after all four turbine control valves unexpectedly opened. The open control valves caused a decrease in main steam line pressure, which led to the automatic shutdown. Control room operators responded appropriately to the plant transient and all safety systems functioned as designed.

The licensee subsequently identified that welds on the supports for the turbine control valve actuator box had failed and that the box shifted downward about six inches. The downward shift caused the control valves to open and overrode the signals from the pressure regulating system.

The licensee determined that the failure of the control valve support structure was caused by inadequate design of the structure combined with shortcomings in weld quality dating from original construction. The final failure occurred after a series of intermediate weld failures, which caused loads to be redistributed to other portions of the support structure, eventually resulting in failure of the remaining welds.

The licensee inspected the main steam lines, located beneath the actuator box, and found no damage to the piping. The licensee staff also inspected other support structures to determine whether conditions similar to those identified on the actuator box supports existed, and no deficiencies were identified.

The resident inspectors, assisted by two inspection specialists from the Region III office, performed a visual examination of the failed support structure welds and reviewed the metallurgical report for failed welds that were tested, the licensee's modification to the actuator box support structure, associated engineering calculations, and a sample of the licensee's corrective actions related to condition evaluation. The resident inspectors also observed portions of the plant restart, including the approach to criticality and the restart of the turbine.

The LER was reviewed by the inspectors and no findings of significance were identified.

.2 Operator Performance During Division II RHR Logic Testing on February 7, 2007

a. Inspection Scope

On February 7, 2007, licensee operations staff were performing surveillance testing of Division II RHR system logic. During the conduct of the test, one test jumper wire was landed inadequately on a terminal, and a second jumper was installed in the wrong electrical panel. The inspectors evaluated the licensee's response to each of these events where personnel error was the initiating cause, including: assessment of the incorrect jumper configuration, removal of the jumper, and evaluation of the potential adverse impacts from the jumper misplacement events. The inspectors reviewed operator logs, surveillance test procedures, and operator conduct expectations.

b. Findings

Introduction: The inspectors identified an unresolved item (URI) associated with the operators' performance and troubleshooting activities associated with the conduct of OSP-RHR-0545-02, "RHR Containment Spray/Cooling Logic Test - Division II," on February 7, 2007. The inspectors identified that the surveillance test, FP-COO-OP-01, "Conduct of Operations,", and management expectations were not adhered to by several licensed operators subsequent to the identification of the misplaced jumpers.

Description: During the performance of OSP-RHR-0545-02, the control room supervisor (CRS, a senior-licensed operator), and a licensed balance-of-plant (BOP) operator, were assigned to install test jumpers across various terminals in the cable spreading room to simulate different plant conditions for logic testing. Prior to performing the test, both individuals performed a walkdown of the cable spreading room where the majority of the duties within the test were to take place. After declaring the Division II RHR system inoperable, the testing commenced. During Step 38 of the procedure, a test jumper wire was incorrectly landed in panel C-33 by the operators. The purpose of this jumper was to simulate high drywell pressure on the 'D' RPS logic channel. The operators, not knowing that the jumper was incorrectly landed, proceeded to Step 39, which included a note stating that the jumper installation in Step 39 was to be installed in panel C-32 (a different panel), and that installation of the jumper would result in receipt of a control room annunciator. The BOP operator did not acknowledge the need to change panels, nor did the CRS verify this requirement. The BOP operator proceeded to land the jumper across similarly-labeled terminals in the wrong panel, C-33. The CRS contacted the control room to verify receipt of the annunciator noted in the procedure; however, the lead control room reactor operator (RO) replied that an annunciator was not received.

At this time, the CRS and BOP operator left the cable spreading room to review RHR logic prints. During this review, they discovered that the jumper, installed in Step 39, had been installed in the wrong electrical panel. The CRS then independently determined that the misplaced jumper had no immediate adverse configuration implication and both individuals returned to the cable spreading room. The BOP operator, as instructed by the CRS, removed the jumper from panel C-33, and installed it in the correct location in panel C-32. The control room was again contacted to determine whether the noted annunciator was received; it was, and the surveillance test was continued.

Later in the test procedure, during Step 50, a control room operator attempted to open valve MO-2011 per the procedure and it did not respond. At this point, the surveillance test was discontinued and the CRS and BOP operator reported to the control room. Shortly thereafter, senior management was notified of the issues and circumstances surrounding the testing and troubleshooting efforts began. The licensee initiated a human performance investigation and removed the involved licensed individuals from duty. The licensee also entered the issue into the corrective action program as CAPs 01075924 and 01075923 and an apparent cause evaluation was conducted.

At the conclusion of the inspection period, the NRC was continuing evaluation and review of the issues surrounding the operators' performance. Therefore, this issue will be considered an URI pending further NRC review (URI 05000263/2007002-04).

.3 Loss of Division II Bus 16, Group II Isolation on March 17, 2007

a. Inspection Scope

On March 17, 2007, with the plant in Mode 5 refueling and Division I equipment protected, a full reactor protection system (RPS) trip, scram, and Group II containment isolation occurred when a potential transformer (POT) drawer associated with 1AR transformer work was removed while Division II Bus 16 was energized.

The inspectors evaluated the licensee's response to this event including: control room operator response and use of abnormal procedures, conduct of personnel in removing the POT drawer under a clearance order, and scheduling impact associated with timing of the work. The inspectors reviewed operator logs, work and clearance orders, and operator conduct expectations.

b. Findings

Introduction: A finding was self-revealed when Division II essential 4 KV Bus 16 was de-energized due to the failure to fully evaluate the scheduling and operational impact of removing a POT drawer associated with isolation activities with Bus 16 energized. This issue was considered to be of very low safety significance (Green) and was dispositioned as a non-cited violation.

Description: On March 17, 2007, with the plant in Mode 5 refueling and Division I equipment protected, plant operators initiated clearance order 17605, "Isolate Bus Metering #16 4KV Bus," to support a control room metering modification. When the operators arrived near Bus 16 to hang the isolation, they noticed that sequence 040 of the clearance order was not specific, in that it directed removal of a POT drawer, but did not list the specific breaker cubicle in which the drawer was located. The POT drawer located within breaker cubicle 152-610, Bus 16 to 1AR supply, was already removed under 4858-04-OCD, "1AR Reserve Transformer Maintenance Isolation"; a separate procedure. Because the clearance order, in part, referenced only breaker 152-610, and because sequence 040 of the clearance order lacked specificity, the operators contacted the Work Control Center (WCC) senior reactor operator (SRO). The operators questioned whether a separate POT drawer within breaker cubicle 152-601, Bus 15 to Bus 16 cross-tie, should be pulled (152-601 contained a POT that supplied signals to bus undervoltage relays). The operators did not notify the WCC SRO that labels stating: "DO NOT OPEN WHEN ENERGIZED" were located below each POT drawer on breaker cubicles 152-601 and 152-610. The operators assumed that the warning label referred to the breaker itself being de-energized, and not the entire bus. In addition, the clearance order and associated outage impact statement did not list any precautions for verifying that the bus was de-energized prior to removing the bus POT drawer.

After the WCC SRO agreed that the clearance order was likely lacking information (i.e., sequence 040 should have referred to cubicle 152-601), the operators were directed to proceed with the isolation and remove the POT drawer from cubicle 152-601. Immediately after the operators removed the drawer, Bus 16 sensed an under-voltage condition due to the POT signal being removed from bus undervoltage relays within cubicle 152-601. Bus 14, which was supplying power to Bus 16, isolated from Bus 16 as designed, resulting in de-energization of Bus 16 due to no other electrical sources supplying power. The 12 EDG did not start on a loss of voltage to the bus because the 12 EDG was in pull-to-lock for planned maintenance. The main control room received a full RPS trip and reactor scram (all rods were already inserted). A Group II isolation resulted primarily in the loss of RWCU and spent fuel pool cooling, and other lighting and ventilation loads were lost.

After unexpected alarms were received, control room operators entered several abnormal operating procedures to restore equipment (e.g. Division II load center 104, which deenergized, was cross-tied to Division I load center 103), performed other troubleshooting activities including a root cause investigation, and made an 8-hour non-emergency report under 10 CFR 50.72 due to the actuation of engineered safety feature equipment. Additional immediate corrective actions included implementation of strict plant status control through the control room, re-assessing shutdown risk due to the loss of RWCU in the heat rejection mode, placing Division II electrical isolations on hold pending independent reviews prior to performing work, and Operations and WCC personnel briefings.

Analysis: The inspectors determined that the failure to fully evaluate the scheduling and configuration control impact, as well as provide adequate instructions to manipulate equipment that could affect equipment credited as critical safety systems, was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” issued on June 22, 2006. The finding was more than minor because it affected the procedure quality attribute of the Initiating Events cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown operations.

The licensee determined inadequate clearance order preparation was a primary cause of the event due to a lack of understanding of the potential impact of removing the POT drawer. In addition, the clearance order and associated outage impact statement did not appropriately contain precautions and/or warning steps before removing the bus POT associated with breaker cubicle 152-601 to verify that Bus 16 was de-energized. Furthermore, communication between the operators in the field and the WCC was inadequate, in that the warning labels on the breaker cubicles was not discussed, i.e., identification of the consequences of pulling the POT drawer could have been identified. The inspectors determined that per the licensee's work planning guidelines and outage impact statement review practices, precautions would have been appropriate for the circumstances considering the complex shutdown plant configuration, infrequent nature of the task, and schedule timing requirements to preclude loss of credited shutdown equipment (e.g., impact on shutdown plant risk). The inspectors determined that the performance deficiency affected the cross-cutting area of Human Performance, having work control components and involving aspects associated with the failure to appropriately coordinate the work activity by incorporating actions to address the operational impact of the testing. In particular, during development and implementation of the clearance order and outage impact statement, several barriers to avoid adversely impacting defense in depth were ineffective due to the lack of operator knowledge in the work planning process and insufficient communication between the work groups. The licensee failed to identify the need for robust precautionary statements in the clearance order or the outage impact statement considering that the testing was an infrequent task and had the potential to affect shutdown defense in depth equipment credited in the critical safety system checklist.

Using IMC 0609, Appendix G, Phase 1 Screening, the inspectors determined that Checklist 8, "Boiling Water Reactor (BWR) Cold Shutdown or Refueling Operation Time to Boil > 2 Hours: RCS Level < 23' Above Top of Flange," applied based on plant conditions at the time of the event and during recovery actions. However, because all qualitative criteria within the Core Heat Removal, Inventory Control, Power Availability, and Containment guidelines were met; because the finding did not meet the Checklist 8 criteria for Phase 2 or Phase 3 quantitative analysis (the finding did not: increase the likelihood of a loss of RCS inventory; degrade the licensee's ability to terminate a leak path or add RCS inventory when needed; significantly degrade the licensee's ability to recover decay heat removal once it was lost; or involve only one or less safety relief valves available to establish a heat removal path to the suppression pool due to the vessel head being removed) and because no event occurred that could be characterized as a loss of control as listed in Table 1 of IMC 0609, Appendix G, the finding was considered to be of very low safety significance (Green).

Enforcement: Title 10 CFR 50, Appendix B, Criterion V requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances. Contrary to this requirement, on March 17, 2007, clearance order 17605 was used to perform a partial isolation of Bus 16, an activity affecting quality. The clearance order and associated outage impact statement were not appropriate for the circumstances because: (1) the documents did not have an adequate level of review to verify the impact of the isolation on shutdown equipment configuration considering the infrequent nature of the task, and (2) the documents did not contain precautions or other information to ensure that the isolation was conducted during an appropriate time within the outage schedule, or to verify that Bus 16 was de-energized before performing the work. Because the finding was of very low safety significance and because the issue was entered into the licensee's corrective action program (CAP 01082734), this violation is being treated as an NCV, consistent with Section VI.A.1 of Enforcement Policy (NCV 05000263/2007002-05).

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. Conway and other members of licensee management on April 3, 2007. 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:

- Baseline procedure 71111.08 with technical staff members on March 23, 2007. The licensee confirmed that none of the potential report input discussed was considered proprietary.
- Access control to radiologically significant areas and the ALARA planning and controls program with Mr. J. Conway, Site Vice President on March 30, 2007.

4OA7 Licensee-Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
J. Conway, Site Vice President
B. Sawatzke, Plant Manager
J. Grubb, Site Engineering Director
S. Radebaugh, Maintenance Manager
B. MacKissock, Operations Manager
W. Guldemond, Nuclear Safety Assurance Manager
K. Jepson, Radiation Protection - Chemistry Manager
R. Latham, Acting Radiation Protection Manager
S. Nelson, Fleet Radiation Protection Manager
S. Halbert, Training Manager
R. Baumer, Regulatory Compliance Engineer
R. Deopere, ISI Coordinator
T. Jones, NDE Coordinator
G. Lofthus, Fleet NDE Lead

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000263/2007002-01	FIN	Inadequate Preparation to Address the Potential Loss of the Station Heating Boiler Prior to the Onset of Extreme Cold Outside Ambient Temperatures (Section 1R01)
05000263/2007002-02	NCV	Failure to Provide Written Instructions for the Compensatory Actions Denoted in Operability Recommendation (Section 1R15)
05000263/2007002-03	NCV	Inadequate 14 Emergency Service Water Surveillance Procedure (Section 4OA2.4)
05000263/2007002-05	NCV	Inadequate Clearance Order for Bus 16 Isolation Resulting in Loss of Bus 16 and Group II Isolation (Section 4OA3.3)

Opened

05000263/2007002-04	URI	Operator Performance During Division II RHR Logic Testing (Section 4OA3.2)

Closed

05000263/2007-01	LER	Reactor Scram Due to Turbine Control Valve Housing Support Failure (Section 4OA3.1)

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.

Section 1R01: Adverse Weather

1151; Winter Checklist; Revision 52
8047; Temporary Heating Boiler Installation; Revision 1
TMOD 04-016; Installation of a Temporary Auxiliary Heating Boiler
C.4-B.08.03.A; Loss of Heating Boiler; Revisions 0, 1, 2, and 3
C.4-B.08.07.A; Part I; Loss of H&V (Heating and Ventilation) to Main Steam Chase; Revision 25
CAP 01077567; Inadequate Procedure Exists for Temporary Heating Boiler Installation
CAP 01077568; Unplanned Entry of TS Actions Due to Boiler Loss
CAP 01077564; Heating Boiler Trip Cause Plant Transient, C.4 and C.5 Entry
CAP 01078193; Component Level Issues Found During Boiler Troubleshooting
CAP 01078606; NRC Question Concerning Temporary Heating Boiler Install Time

Section 1R04: Equipment Alignment

2126-02; Plant Prestart Checklist, Batteries and DC Power System 125 Vdc; Revision 14
0194; 11 and 12 125 Vdc Battery Operability Check - Weekly Test; Revision 22
CAP 01072646; NRC Question on D10 Battery Charger
2154-12; RHR System Prestart Valve Checklist; Revision 41
2154-10; HPCI System Prestart Valve Checklist; Revision 27
2118; Plant Restart Checklist, HPCI System; Revision 14
2154-35; HPCI Hydraulic Control and Lubrication System Prestart Valve Checklist; Revision 8
9111-01; Shutdown Cooling Division I Protected System Ticket Checklist; Revision 2

Section 1R05: Fire Protection

Strategy A.3-02-F; Main Steam Chase; Revision 3
Strategy A.3-12-E; Steam Air Ejector Room; Revision 1
Strategy A.3-12-D; MVP Room; Revision 5
Strategy A.3-15-E; Diesel Oil Pump House; Revision 4
Strategy A.3-40; Screen House; Revision 0
Strategy A.3-19-C; Feedwater Pipe Chase; Revision 5
Strategy A.3-21-C; Radwaste Shipping Building; Revision 6
Strategy A.3-21-B; Radwaste Trash Compactor Area; Revision 3
Strategy A.3-21-A; Radwaste Control Room; Revision 3
Strategy A.3-13-A; Lube Oil Storage Tank Room; Revision 5
Strategy A.3-12-A; Lower 4 KV Bus Area (11, 13 and 15); Revision 11

Section 1R06: Flood Protection Measures

CAP 01075497; Drain Line to North Retention Pond Frozen
Calculation (CA)-91-136; East Turbine Building Pipe Break Analysis; Revision 1
CA-03-202; Evaluation of the Fire Protection Piping in the EFT Building; Revision 0
CA-03-200; Internal Flooding Evaluation Due to a Postulated Break in 2.5" Fire Line; Revision 0

Drawing NF-36697; Yard Work Grading Drainage and Utilities - Area 1
Drawing NF-93262; EFT Building Plumbing, Drainage, and Fire Protection
Design Basis Document (DBD) T.08; Internal Flooding; Revision 3

Section 1R07: Heat Sink Performance
4109-01-PM; EDG 12 Year Maintenance; Revision 13
3843; Eddy Current Results Evaluation; Revision 0
3802; Visual Inspection of Heat Exchanger Condition; Revision 0
3590; Service Water Component Inspection; Revision 6
WO 150978; 1404-02 12 EDG ESW Heat Exchanger Performance Test; July 26, 2006
CAP 01083629; 12 EDG Jacket Coolers require tube plugs

Section 1R08: Inservice Inspection Activities
Work Order 0505187; Feedwater (FW)-97-1 ; March 5, 2005
WP-6; Carbon Steels Group P-1 to P-1 GTAW (Gas Tungsten Arc Welding) - Pipe Size Over 1" OD; Revision 1
NX-9235-37; Monticello Nuclear Generating Plant 14" 600# Swing Check Valve; Revision H
FP-PE-NDE-401; Ultrasonic Examination of Ferritic Pipe Welds - Supplement 3; Revision 0
PEI-02.02.01; Dry Powder Magnetic Particle Examination; Revision 0
PEI-02.05.02; Visual Examination of Components and Their Supports; Revision 0
CAP 0821013; ISI NDE (Non-destructive Examination) Examiner Procedural Errors
CAP 0830468; Numerous Small Leaks Found During Class 1 Pressure Test in Packing and Joints
CAP 0830991; ISI VT Indication, FW-97-1 Disc Stem Wear, Followup to Actions taken
CAP 0838103; ISI Indication, FW-97-1, Erosion on Hinge Pin Plug Hole Threads
CAP 0828290; ASME Section XI Repair/Replacement Plan not Prepared for New P-200A Stuffing Box
CAP 01036371; Code Case Revision not Current in Pressure Test AWI and Proc’s
CAP 0817645; CRD Pipe Thru wall Leakage
CAP 0830983; ISI/IWE Followup Exam of X-107 Interior
CAP 0815605; Undervessel Leakage Inspection Identifies Four CRD Flange Leaks

Section 1R11: Licensed Operator Requalification Program
Simulator Exercise Guide M-8117S-047; Loss of shutdown cooling with LOCA
FP-OP-COO-01; Conduct of Operations; Revision 01

Section 1R12: Maintenance Effectiveness
4901-01-PM; Limit Switch Setting Procedure for Limitorque Valve Operators; Revision 15
4900-01-PM; PM for Limitorque MOVs; Revision 23
CAP 0825017; CV-3267 and CV-3269 Showed Dual Indication During Appendix J PMT
CAP 0831391; Limit Switches on CV-3269 Not Installed or Tested Properly by Valve Contractor
CAP 0891723; HPCI Steam Admission Valve MO-2036 Has a Seat Leak
CAP 01024874; Limit Switches for AO-12-4-41B Need Adjustment
Monticello Maintenance Rule Program Document; System Basis Document; Structures; Revision 2
CAP 01081243; Maintenance Rule Expert Panel Determined Structures System (a)(1)
Maintenance Rule Evaluation Associated with CAP 01071128; Reactor Scram Number 119 Occurred on Jan 10, 2007

1385; Periodic Structural Inspection; Revision 4
Monticello Maintenance Rule Program Document; System Basis Document; 4.16 KV Station Auxiliary; Revision 5
B.09.06-05, Racking In 4 KV Circuit Breakers; Revision 20
CAP 01080681; 13 RHR pump fails to start after 4850-503-PM
CAP 01082726; Breaker 52-404 did not close on first attempt on MCC-143A
CAP 0844746; Breaker 52-602 did not close on first attempt

Section 1R13: Maintenance Risk Assessments and Emergent Work Control
WO 310671; Investigate Electrical Pressure Regulator Shutdown
WO 00311552; Main Steam Control Valve Actuator Support Modification
Engineering Change (EC) 9904; Main Steam Control Valve Actuator Support Modification
Monticello Station Logs; February 3 - February 5, 2007
Operations Work Instruction (OWI)-01.04; Operations General Procedural Guidance; Revision 13
CAP 01075426; 3N4 Low Hydraulic System Pressure
Work Week 0705 T-1 Schedule; February 4 - February 10, 2007
8047; Temporary Heating Boiler Installation; Revision 1
C.4-B.08.03.A; Loss of Heating Boiler; Revision 3
Monticello Station Logs; February 15 - February 16, 2007
WO 315054; Replace Bent Valve Stem on RC-41-2

Section 1R15: Operability Evaluations
CAP 01070238; No Carbon in EFT Test Canister Removed 1/4/07
EWI-08.02.03; Ventilation Filter Testing Program; Revision 3
CAP 01071663; Bent Hanger Rod on Main Steam Pressure Averaging Manifold
CAP 01073031; Abnormal Behavior of CRD 42-11
CAP 01071697; Spurious Self Actuation of SRV ‘F’ Occurs Following Scram
CAP 01079794; Potential leakage past MO-2029 and MO-2030
WO 314958; Switch Diaphragm Failed and Leaking
Monticello Station Logs; February 21 - March 1, 2007
0137-11; Shutdown Cooling Suction Line Isolation Valve Leak Test; Revision 12
OPR 01076631; 14 ESW Operability
FP-OP-OL-01; Operability Determination; Revision 2
CAP 01078881; NRC Comments on OPR for 14 ESW Pump
CAP 01076631; P111D, 14 ESW Flow to RHR ‘B’ Room Less Than Required Band
CAP 01078447; Operability Recommendation Not Completed in a Timely Manner
CAP 01077117; Division II ESW Flow to ‘B’ RHR Room Less Than Minimum
CAP 01078193; Procedure Step Performed Incorrectly
CAP 01079568; Rx Water Level Calculation CA-95-073 May Not Be Conservative
CA-95-073; Reactor Low Water Level SCRAM Setpoint; Revision 2
CA-95-073; Reactor Low Water Level SCRAM Setpoint; Revision 3
GE-NE-0000-0066-0631-R0; MNGP Evaluation of Bernoulli Error for L3 Setpoint; March 2007
SC04-14; 10 CFR 21 Communication: Narrow Range Water Level Instrument Level 3 Trip Final Report; October 11, 2004

Section 1R17: Permanent Plant Modifications
NUREG-0612; Control of Heavy Loads at Nuclear Power Plants; July 1980
EC 785; Reactor Building Crane Upgrade for ISFSI; Revision 2
EC 9647; Reactor Building Crane Load Test Engineering Evaluation
8151; Heavy Load Movement Procedure; Revision 11
Modification 04Q162; Reactor Building Structural Upgrades for ISFSI; Revision 0
CAP 01070508; H-2 Reactor Building Crane Overload Switch Tripped During 125% Test
SCR-06-0059; Reactor Building Crane Upgrade for ISFSI; Revision 0

Section 1R19: Post-Maintenance Testing
0466-01; 'A' EFT Filter Efficiency and Leak Tests; Revision 29
WO 139476-53; Re-terminate and Calibrate LS-7428F
0006; Scram Discharge Volume Hi Level Scram Test and Calibration Procedure; Revision 24
4900-01-PM; PM for Limitorque MOVs; Revision 23
WO 293729; MO-2101, PM 4900-01 for MO-2101
WO 313629; 11 RWCU Pump Seal Replacement
4311; 11 RWCU Recirculation Pump Rotating Assembly Replacement; Revision 8
CAP 01079797; Failed Vibration PMT on 11 RWCU Pump
0255-04-IA-1-1; RHR Loop 'A' Quarterly Pump and Valve Tests; Revision 72
WO 320677; P-202C (13 RHR Pump) Failed To Start From C03
4850-503-PM; 152-503, 13 RHR Pump Relay Maintenance, Calibration, and Test Tripping; Revision 5
B.09.06-05; 4.16 KV Station Auxiliary System Operation; Revision 20
1079-02; 12 Emergency Diesel Generator Overspeed Trip Check; Revision 7
CAP 01084801; Emergency shutdown of 12 EDG due to oil leak
CAP 01084691; Defective NFLD

Section 1R20: Outage Activities
CAP 01072425; Steam Jet Air Ejector (SJAE) Room Items of Concern Identified by NRC
CAP 01073520; 2-271C, 'C' SRV Tailpipe Temperature was Elevated During Startup
CAP 01073517; Potentially Unanalyzed Release Path During Inerting
CAP 01073451; Low Meggar Readings on Hydrogen Seals During 4118-PM
CAP 01073543; Failed Capacitor Identified in Shaft Voltage Suppression
CAP 01071128; Reactor Scram Number 119 Occurred on January 10, 2007
Operations Manual C.1; Startup Procedure; Revision 49
Operations Manual C.3; Shutdown Procedure; Revision 45
2150; Plant Startup Checklist; Revision 33
2159; Predicted Critical for Plant Startup; Revision 7
2165; SCRAM Report; Revision 21
WO 317805; Inspect Cell 42-11 for the causes of blade failure to settle

Section 1R22: Surveillance Testing
1040-02; Turbine Generator Monthly Operational Tests; Revision 56
1040-04; Turbine Generator Quarterly Operational Tests; Revision 56
CAP 01051814; CV2 and CV3 Failed to Operate Properly During 1040-01 Test
CAP 01004519; CV1 Test Stroke Issues
0160-A; MSIV Exercise Test; Revision 4

0395; Alternate Shutdown System (ASDS) Cycle Functional Test for Division II RHR, RHR Service Water, ESW Switches and Control Room Annunciator for ASDS Master Transfer Switch; Revision 10
Reactivity Maneuvering Steps for February 4, 2007, Reactivity Adjustment; Revision 0
2300; Reactivity Adjustment; Revision 0
WO 314372; Operate 42-11 Per the Attached Work Plan
0081; CRD Scram Insertion Time Test; Revision 49
0255-03-IA-2A; CS - Shutdown Valve Operability Test; Revision 21

Section 2OS1: Access Control to Radiologically Significant Areas

AR 01078286; Personnel Contamination on Individual Working on Spare Reactor Water Cleanup Pump Rotating Assembly; dated February 20, 2007
AR 01078593; Two Workers Performing Work Order under a Rwp 267 in the Clean Tool Crib Received Ed Dose Alarms; dated February 21, 2007
AR 01074629; Routine Dose in Work Week March 2007, Higher than the Estimated; dated March 26, 2007
CAP 01084697; Worker Entered Drywell on Wrong Work Order and Rwp; dated March 29, 2007
AR 01080071; Worker Shows Co-60 on Coat During Incoming Fastscan; dated March 26, 2007
AR 01060796; Personnel Contamination on Shoes; dated May 23, 2006
AR 01061334; Personnel Contamination on the Right Arm; dated February 13, 2007
AR 01064139; Individual Shows Cs-137 Activity on Incoming Wholebody Count; dated December 18, 2006
AR 01082217; Workmen in Non-posted, Roped-off Contaminated Area During Setup; dated March 26, 2007
AR 01082639; Individual Contamination During Working on Torus Drain Pump; dated March 26, 2007
R.13.03; Radiography Procedure; Revision 9
4 AWI-08.04.04; Respiratory Protection, Administrative Work Instruction; Revision 10
FP-RP-JPP-01; Radiation Protection Job Planning, Fleet Procedure; Revision 2, Fleet Procedure; dated November 10, 2006
FP-RP-RWP-01; Radiation Work Permit; Revision 5; Fleet Procedure; dated November 10, 2006
RFP 701; Radiological Diving Operations; Revision 0; dated February 23, 2007
ACP 1411.13; Control of Locked High Radiation Areas and above; Revision 18
ACP 1407.2; Material Control in The Spent Fuel Pool and Cask Pool; Revision 15

Section 2OS2: ALARA Planning and Controls

Radiological Work Assessment (RWA) Form on Work Order (WO) No. 306896; Main Steam Chase System Leakage Check Procedure; dated March 21, 2007
RWA Form on WO No. 140418; No. 12 Recirc Pump Replacement and Associated Activities; dated November 28, 2006
RWA Form on WO No. 288426; Drywell Nozzle Inservice Inspection (ISI); dated November 18, 2006
RWA Form on WO 134819; 139885; and 291040; Torus Project; dated February 28, 2007
RWP 691; Grit Blasting and Vacuum Blasting Inside Torus; dated March 28, 2007

RWP 786; Locked High Radiation Area-dose Rates less than 500 Mrem per Hour; dated March 20, 2007
RWP 708; General Drywell ISI, IWE, and Flow Accelerated Corrosion Inspection; dated December 6, 2007
RWP 390; Radiography at Locked High Radiation Area (LHRA); dated November 30, 2006
RWP 593; Steam Area Work in Contaminated and LHRA Areas; dated January 16, 2006
RWP 652; Recirculation Pump No. 12 Replacement and Associated Activities; dated March 14, 2007
RWP 679; Reactor Cavity Decontamination; dated December 8, 2006
RWP 694; Torus Draindown and Decontamination Activities; dated March 18, 2007
RWP 701; Nozzle Inservice Inspections in Drywell; dated March 23, 2007
HPP 3102.02; ALARA Job Planning; Revision 21
HPP 3104.07; Diving Operations within Radiological Areas; Revision 15
RP-AA-460-1001; Additional High Radiation Exposure Control; Revision 1
RFP 607; Removal and Movement of Materials Within the Spent Fuel Pool and Cask Pool; Revision 9
RFP 403; Performance of Fuel Handling Activities; Revision 27
2006-004-5-012; Nuclear Oversight Observation Report, Radiological Protection Assessment; dated November 21, 2006
2007-001-011; Nuclear Oversight Observation Report, Radiological Protection Assessment; dated February 22, 2007

Section 4OA2: Identification and Resolution of Problems
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 34; dated April 1, 2005
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 37; dated August 5, 2005
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 39; dated October 19, 2005
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 39; dated January 17, 2006
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 40; dated April 20, 2006
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 41; dated July 19, 2006
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 41; dated October 17, 2006
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 41; dated February 13, 2007
0255-11-III-4; 14 ESW Quarterly Pump and Valve Tests; Revision 42; dated February 13, 2007
Mississippi River Temperature for May through August 2005 and 2006
CAP 01081038; ESW Surveillance Test Does Not Verify Performance at Worst Design Condition
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 30; dated January 3, 2005
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 31; dated March 16, 2005
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 33; dated July 22, 2005
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 35; dated October 5, 2005
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 35; dated January 3, 2006
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 36; dated April 3, 2006
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 36; dated July 4, 2006
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 36; dated October 6, 2006
0255-11-III-3; 13 ESW Quarterly Pump and Valve Tests; Revision 36; dated January 9, 2007

Section 4OA3: Event Follow-up
CE 01071663-01; Condition Evaluation: Bent Rod Hanger on Main Steam Pressure Averaging Manifold

CE 01071769-01; Condition Evaluation: Extent of Condition - Evaluation of Components in Immediate Area of Failed Main Stream Control Valve Actuator Support
CA-07-001; Calculation: Main Steam Control Valve Actuator Support Modification; Revision 0
MN07-995-026-100; Calculation: Operability Evaluation of Main Steam (MS) Piping PS1, 2, 3, and 4 Outside Containment; Revision 0
Unnumbered; Failure Analysis of Structural Welds on the Main Steam Control Valve Actuator Enclosure; Revision Draft
Monticello Station Logs; February 7, 2007
CAP 01075923; Jumper Was Installed on Wrong Points During OSP-RHR-0545-02
CAP 01075924; MO-2011 Did Not Respond to Step 50 in OSP-RHR-0545-02
OSP-RHR-0545-02; RHR Containment Spray/Cooling Logic Test - Division II; Revision 0
FP-OP-COO-01; Conduct of Operations; Revision 1
CAP 01082734; Lost Essential Bus 16 during isolation activities
Clearance Order 17605; Hang C/L: 1AR Bus Metering
Monticello Station Logs; March 17, 2007
CAP 01082725; Defense in Depth Color turned Orange for Decay Heat Removal
4858-04-OCD; 1AR Reserve Transformer Maintenance Isolation
3749-02; Monticello Impact Statement - Outage; for work order 288356

LIST OF ACRONYMS USED

AC	Alternating Current
ALARA	As-Low-As-Is-Reasonably-Achievable
ASDS	Alternate Shutdown System
ASME	American Society of Mechanical Engineers
BOP	Balance-of-Plant
BWR	Boiling Water Reactor
CA	Calculation
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CRS	Control Room Supervisor
CS	Core Spray
DBD	Design Basis Document
DC	Direct Current
°F	degrees Fahrenheit
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EC	Engineering Change
ED	Electronic Dosimetry
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
ESW	Emergency Service Water
FIN	Finding
FW	Feedwater
gpm	Gallons per Minute
GTAW	Gas Tungsten Arc Welding
HEPA	High Efficiency Particulate Air
HRA	High Radiation Area
HPCI	High Pressure Core Injection
IMC	Inspection Manual Chapter
IR	Inspection Report
ISI	Inservice Inspection
ISFSI	Independent Spent Fuel Storage Installation
ITS	Improved Technical Specification
KV	Kilovolt
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MNGP	Monticello Nuclear Generating Plant
MOV	Motor-Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle Examination
MVP	Mechanical Vacuum Pump

LIST OF ACRONYMS USED

NCV	Non-Cited Violation
NDE	Non-destructive Examination
NEI	Nuclear Energy Institute
NMC	Nuclear Management Company
NRC	U.S. Nuclear Regulatory Commission
OBD	Operable-But-Degraded
OBN	Operable-But-Non-Conforming
OPR	Operability Recommendation
OWI	Operations Work Instruction
PARS	Publicly Available Records
PI	Performance Indicator
PM	Planned or Preventative Maintenance
PMT	Post-Maintenance Test
POT	Potential Transformer
RA	Risk Assessment
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RO	Reactor Operator
RP	Radiation Protection
RPS	Reactor Protection System
RPT	Radiation Protection Technician
RWP	Radiation Work Permit
RWCU	Reactor Water Cleanup
SDP	Significance Determination Process
SJAE	Steam Jet Air Ejector
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
TMOD	Temporary Modification
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
UT	Ultrasonic Examination
Vac	Volts Alternating Current
Vdc	Volts Direct Current
VT	Visual Examination
WCC	Work Control Center
WO	Work Order