August 8, 2007

Mr. Dennis L. Koehl
Site Vice President
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6590 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, NRC INTEGRATED INSPECTION REPORT 05000266/2007003 AND 05000301/2007003

Dear Mr. Koehl:

On June 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Point Beach Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on July 12, 2007, with you and 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 your personnel.

Based on the results of this inspection, five findings of very low safety significance associated with violations of NRC requirements were identified. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, 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 copies 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 at the Point Beach Nuclear Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 5
Division of Reactor Projects

Docket Nos. 50-266; 50-301
License Nos. DPR-24; DPR-27

Enclosure: Inspection Report 05000266/2007003; 05000301/2007003
w/Attachment: Supplemental Information

cc w/encl: F. Kuester, President and Chief
Executive Officer, We Generation
D. Cooper, Senior Vice President and Chief
Nuclear Officer
J. McCarthy, Site Director of Operations
D. Weaver, Nuclear Asset Manager
Plant Manager
Regulatory Affairs Manager
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Site Assessment Manager
Site Engineering Director
Emergency Planning Manager
J. Rogoff, Vice President, Counsel & Secretary
K. Duveneck, Town Chairman
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Public Service Commission of Wisconsin
J. Kitsembel, Electric Division
Public Service Commission of Wisconsin
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Letter to D. Koehl from J. Cameron dated August 8, 2007

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-266; 50-301
License Nos:	DPR-24; DPR-27
Report No:	05000266/2007003; 05000301/2007003
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Plant, Units 1 and 2
Location:	Two Rivers, Wisconsin
Dates:	April 1, 2007, through June 30, 2007
Inspectors:	R. Krsek, Senior Resident Inspector G. Gibbs, Resident Inspector K. Barclay, Reactor Engineer D. Betancourt, Reactor Engineer R. Daley, Senior Reactor Inspector S. Ghasemian, Enforcement Specialist, Office of Enforcement J. Jandovitz, Reactor Inspector W. Slawinski, Senior Health Physicist B. Smith, Reactor Engineer D. Szwarc, Reactor Inspector
Approved by:	J. Cameron, Chief Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000266/2007003 and 05000301/2007003; 04/01/2007 - 06/30/2007; Point Beach Nuclear Plant, Units 1 and 2; Maintenance Effectiveness; Maintenance Risk Assessment and Emergent Work Evaluation; Post Maintenance Testing; Surveillance Testing; and Event Followup.

This report covers a three-month period of inspections by resident inspectors and regional specialists. Five Green findings with associated non-cited violations (NCVs) and one unresolved item were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green: A self-revealing finding and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions , Procedures, and Drawings,” having very low safety significance (Green), was identified for failure to have procedures appropriate to the circumstances for maintenance on air-operated valve positioners, when hardware attaching the connecting link between the Unit 1 “B” feedwater regulating valve positioner and actuator became disconnected resulting in loss of control of the valve. Specifically, there were no procedures that ensured that positioner arm hardware was properly secured. The licensee repaired valve positioners as required, performed an extent-of-condition review for similar valve positioners and is performing a root cause evaluation.

  The inspectors concluded the finding is greater than minor because the finding was associated with the equipment performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The transient initiator contributor was a reactor trip that did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Consequently, the finding is considered to be of very low safety significance (Green). The inspectors also determined that the primary cause of this finding is related to the cross-cutting area of human performance (H.2.(c)). Specifically, under the component of resources, the licensee failed to ensure complete, accurate, and up-to-date procedures and work packages for work on air-operated valve positioners were available. (Section 4OA3.1)

Cornerstone: Mitigating Systems

- Green: The inspectors identified a NCV of 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," after the licensee failed to adequately manage the risk associated with the installation of the Unit 1 Steam Generator Nozzle Dams, which is a reduced inventory and Orange Qualitative Risk Condition. Specifically, the contingency plan stated, in part, that an uncontrolled reactor coolant system inventory loss would be mitigated with the use of Shutdown Emergency Procedure SEP-2, "Cold Shutdown LOCA." However, the inspectors noted that certain critical equipment required in SEP-2 was not available and no contingencies were established for the unavailable equipment. The licensee initiated condition reports and took immediate corrective actions and planned additional corrective actions based on a causal evaluation.

  The finding was greater than minor because the finding affected the cornerstone objective, to ensure the availability of systems that respond to initiating events to prevent undesirable consequences, and the attributes of configuration control and equipment performance, due to the shutdown equipment lineup and unavailability of equipment. In addition, the finding was related to the licensee's failure to effectively manage significant compensatory measures for this Orange Risk condition. The finding screened as very low safety significance (Green), because the finding did not meet the criteria for a Phase 2 or Phase 3 Analysis, as specified in IMC 0609 Appendix G, Attachment 1, Checklist 1, "PWR Hot Shutdown Operation: time to Core Boiling < 2 Hours." The inspectors also determined that the primary cause of this finding is related to the cross-cutting area of human performance (H.3(a)). Specifically, under the component of work control, the licensee did not appropriately plan work activities by incorporating the need for planned contingencies and compensatory actions, ensuring that equipment relied upon for contingencies remained available. (Section 1R13.1)

- Green: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to accomplish required preventive maintenance resulting in the D-108 Station Battery output becoming unstable on several occasions. In January 2007, the D-09 Battery Charger also failed as a result of failure to perform scheduled preventive maintenance. The licensee initiated condition reports, took immediate corrective actions to repair the chargers and is performing an apparent cause evaluation.

  The inspectors concluded that the finding is greater than minor because if left uncorrected, the finding would become a more significant safety concern, in that, failures of safety-related battery chargers can significantly challenge the vital 125V DC system. In addition, the finding is associated with the equipment performance attribute of the Mitigating System cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, (such as, core damage). Since the finding is not a loss of system safety function and is not an actual loss of safety function of a single train for greater than its

Technical Specification allowed outage time, the finding is considered to be of very low safety significance (Green). The inspectors also determined that the primary cause of this finding is related to the cross-cutting area of human performance (H.3(b)). Specifically, the licensee did not appropriately coordinate work activities to support long-term equipment reliability and maintenance scheduling, which was not more preventive than reactive, as critical preventative maintenance for battery chargers was not performed. (Section 1R12.1)

- Green: The inspectors identified a NCV of 10 CFR Part 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," of very low safety significance (Green), for the failure to incorporate available internal and external Operating Experience (OE) pertaining to 4.16kV switchgear cubicle Mechanism Operated Control (MOC) switch assemblies. Preventive maintenance procedures for Westinghouse 4.16kV switchgear cubicles had not been revised to incorporate important MOC switch linkage measurements, adjustments and verification of contact position. The licensee initiated condition reports and is revising procedures to incorporate required preventive maintenance.

  The inspectors concluded that the finding is greater than minor, because, if left uncorrected, the finding would become a more significant safety concern. The finding also affects the procedure quality attribute of the Mitigating System cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (such as, core damage). Since the finding is not a loss of system safety function and is not an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, the finding is considered to be of very low safety significance (Green). Additionally, the inspectors determined that the contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution within the component of OE (P.2(b)). The licensee did not implement and institutionalize OE through changes to station processes and procedures, as appropriate preventive maintenance procedures and routines were not established. (Section 1R19.1)

Cornerstone: Barrier Integrity

- Green: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for the failure to have procedures appropriate to the circumstances, which established the appropriate test conditions for primary coolant sources testing outside containment. Specifically, testing procedures, which satisfied Technical Specification 5.5.2, "Primary Coolant Sources Outside Containment," did not ensure that residual deposits of boric acid on the containment spray, high head and low head safety injection systems were removed, so that active system fluid leaks could be identified as required during the tests. The issue was entered into the licensee's corrective action program (CAP), the licensee took immediate corrective actions, and performed a causal evaluation at the end of this inspection.

The inspectors evaluated the finding using IMC 0609, “Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.” The finding screened as very low safety significance (Green) because the finding did not: represent the degradation of the radiological barrier function provided for the auxiliary building; represent a degradation of the barrier function of the control room; and did not represent an actual open pathway in the physical integrity of reactor containment. The inspectors also determined that the primary cause of this finding is related to the cross-cutting area of human performance (H.2(c)). Specifically, under the component of resources, the licensee failed to ensure that procedures were adequate and accurate to assure nuclear safety. (Section 1R22.1)

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

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period shutdown for the Cycle 30 Refueling Outage (U1R30). Unit 1 remained shutdown until May 7, 2007, when the Unit was returned to power operations. Unit 1 remained at 100 percent power, until June 5, 2007, when operators manually tripped the reactor, in response to a main feedwater system malfunction. Unit 1 was returned to power operations on June 9, 2007, and remained at 100 percent power until June 14, 2007, when a Technical Specification (TS) required shutdown was initiated, due to the Unit 1 turbine driven auxiliary feedwater pump being inoperable in excess of the allowed outage time. On June 23, 2007, following repair of the turbine driven auxiliary feedwater pump, Unit 1 was returned to power operations, where the unit remained until the end of the inspection period.

Unit 2 was at 100 percent power throughout the inspection period with the exception of brief reductions in power during routine auxiliary feedwater and secondary system valve testing.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors walked down the switchyard, and the main and auxiliary transformers which are susceptible to potential missiles (debris) that could be generated by high winds or tornados. The inspectors reviewed the corrective actions and work orders (WOs) written to correct identified problems. The inspectors also walked down areas which had a history of poor external housekeeping. This observation constituted one inspection procedure sample for impending adverse weather conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of risk-significant systems to determine the operability of the systems. The inspectors utilized system valve lineup and electrical breaker checklists, tank level books, plant drawings, and selected operating procedures to determine whether the systems were correctly aligned to perform the intended design functions. The inspectors also examined the material condition of the components and observed operating equipment parameters to determine whether deficiencies existed. The inspectors reviewed completed WOs and calibration records associated with the systems for issues that could affect component or train functions. The inspectors used the information in the appropriate sections of the Final Safety Analysis Report (FSAR) to determine the functional requirements of the system. Partial system walkdowns of the following systems constituted three inspection procedure samples:

- Spent Fuel Pool Cooling during Unit 1 Full Core Offload;
- Unit 2 ‘ B’ Component Cooling Water (CCW) while ‘A’ CCW pump was out of service; and
- Emergency Diesel Generator (EDG) G02 while EDG G01 was out of service.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Walkdown of Selected Fire Zones

a. Inspection Scope

The inspectors conducted fire protection walkdowns which focused on the following attributes: the availability, accessibility, and condition of fire fighting equipment; the control of transient combustibles and ignition sources; and the condition and status of installed fire barriers. The inspectors selected fire areas for inspection based on the area’s overall fire risk contribution, as documented in the Individual Plant Examination of External Events, or the potential of a fire to impact equipment that could initiate a plant transient.

In addition, the inspectors assessed these additional fire protection attributes during walkdowns: fire hoses and extinguishers were in the designated locations and available for immediate use; unobstructed fire detectors and sprinklers; transient material loading within the analyzed limits; and fire doors, dampers, and penetration seals in satisfactory condition. The inspectors also determined whether minor issues identified during the inspection were entered into the licensee’s corrective action program. The walkdown of the following selected fire zones constituted eight inspection procedure samples:

- Fire Zone 511/Fire Area A36; Unit 1 Containment, 21-Foot Elevation;
- Fire Zone 308/Fire Area A27; EDG G01 Room;
- Fire Zone 318/Fire Area A30; Cable Spreading Room;
- Fire Zone 770/Fire Area A71; EDG G03 Room;
- Fire Area A01F; Yard Area;
- Fire Zone 142/Fire Area A01A; CCW Pump Area;
- Fire Zone 321/Fire Area A54; Swing Battery Room D305; and
- Fire Area A17; D04 Electrical Equipment Room.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

The inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary, risk-significant piping system boundaries, and the containment boundary.

From April 2 through April 12, 2007, the inspectors evaluated activities involving non-destructive examination, other examinations, and ISI inspection results with recordable indications and welding. Specifically, the inspectors observed the following:

- Ultrasonic examination of Safety Injection Piping welds SIS-06-CS-1004-09 and SIS-06-CS-1004-12;
- Dye Penetrant examination of Safety Injection Piping welds SIS -06-CS-1004-09 and SIS-06-CS-1004-12; and
- Visual (VT-3) examinations of component support SI-301R-1H8.

The inspectors selected these components in sequence of risk priority as identified in Section 08-03 of Inspection Procedure 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the on-site inspection period. The inspectors evaluated these examinations for compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI and plant TS requirements, to determine whether indications and defects (if present) were dispositioned in accordance with the ASME Code.

The inspectors reviewed the licensee's records related to disposition of recordable indications identified in four examinations. Specifically, the inspectors reviewed the evaluation records with recordable indications accepted for continued service for:

- Indication Disposition Report 2005-031, PT indications found after weld preparation on weld RC-04-PR-1001-05; dated October 22, 2005;
- CAP 00067738, Indications found on pressurizer spray line weld exceed acceptance criteria; dated October 08, 2005;
- Indication Disposition Report 2005-035, Unacceptable Ultrasonic Indications on Steam Generator Weld SG-B-4; dated October 28, 2005; and
- Indication Disposition Report 2006-004, through wall leak on Non-Regent HX Shell Side Outlet Flow Element; dated August 08, 2006.

The inspectors reviewed licensee's records related to pressure boundary welding performed in the following components:

- SI-00845A valve replacement, 2 inch valve from P-15A Safety Injection Pump to Reactor Coolant (RC) Loop A Cold Leg Safety Injection (SI); dated March 22, 2004, and

- RH-00715C valve replacement, 2 inch valve Residual Heat Removal to Letdown cross-connect; dated September 25, 2005.

The inspectors performed this review to determine whether the welding acceptance and pre-service examinations (e.g., pressure testing, visual, dye penetrant, and weld procedure qualification tensile tests and bend tests) were performed in accordance with the requirements of the ASME Code, Sections III, V, IX, and XI.

The above review counted as one inspection sample.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

Point Beach Unit 1 is in the low susceptibility ranking category due to replacement. No control rod drive mechanism non-destructive examinations were performed this outage. Therefore, no inspection sample was credited.

.3 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

From April 2, 2007, through April 12, 2007, the inspectors reviewed the Unit 1 BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary.”

The inspectors observed the licensee conducting a walkdown of borated systems within the Unit 1 containment. The resident inspectors observed the licensee during these examinations to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. In particular, the inspectors performed this observation to determine whether the licensee focused BACC inspections on locations where boric acid leaks can cause degradation of safety significant components and to determine whether degraded or non-conforming conditions were properly identified in the licensee’s corrective action system.

The inspectors reviewed corrective actions and evaluations performed for boric acid found on reactor coolant system connected piping and components to confirm that corrective actions were consistent with requirements of Section XI of the ASME Code and 10 CFR Part 50, Appendix B, Criterion XVI, and that the minimum code required section thickness had been maintained for the affected components. In particular the inspectors reviewed boric acid evaluations for indications at Safety Injection Valve 2SI-829D.

The documents reviewed during this inspection are listed in the Attachment to this report.

The review counted as one inspection sample.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube ISI

a. Inspection Scope

From April 9, 2007 through April 12, 2007, the inspectors performed an on-site review of SG “B” tube examination activities (the “A” SG was not examined) conducted pursuant to TS and the ASME Code Section XI requirements.

The NRC inspectors observed acquisition of eddy current (ET) data, interviewed ET data analysts, and reviewed documents related to the SG ISI program and determined:

- in-situ SG tube pressure testing was not required;
- the numbers and sizes of SG tube flaws/degradation identified was bounded by the licensee’s previous outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to identify tube degradation based on site and industry operating experience by confirming that the ET scope completed was consistent with the licensee’s procedures, plant TS requirements and EPRI 1003138, “Pressurized Water Reactor Steam Generator Examination Guidelines,” Revision 6.”
- the SG tube ET examination scope included tube areas which represented ET challenges such as the tubesheet regions, expansion transitions, and support plates;
- the licensee did not identify new tube degradation mechanisms;
- no tubes required repair;
- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below the detection threshold during the previous operating cycle;
- the licensee did not find loose parts through visual inspections or eddy current examinations;

- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, "Performance Demonstration for Eddy Current Examination," of EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6; and

- the licensee identified deviations from ET data acquisition or analysis procedures.

The inspectors performed a review of SG ISI related problems that were identified and entered into the corrective action program by the licensee, conducted interviews with licensee staff, and reviewed licensee corrective action records to determine whether:

- the licensee had described the scope of the SG related problems;

- the licensee had established an appropriate threshold for identifying issues;

- the licensee had evaluated industry generic issues related to SG tube integrity; and

- the licensee implemented appropriate corrective actions.

The inspectors 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 inspectors are listed in the Attachment to this report.

The review discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Resident Inspector Quarterly Observation of Licensed Operator Requalification

a. Inspection Scope

On June 25, 2007, the inspectors observed the operating crew performance during a simulator as-found requalification examination. Observation of the requalification quarterly evaluation constituted one inspection procedure sample.

The inspectors assessed crew performance in the areas of:

- Clarity and formality of communications;
- Understanding of the interactions and function of the operating crew during an emergency;

- Prioritization, interpretation, and verification of actions required for emergency procedure use and interpretation;
- Oversight and direction from supervisors; and
- Group dynamics.

Crew performance in these areas was also compared to licensee management expectations and guidelines, as presented in nuclear plant procedure NP 2.1.1, “Conduct of Operations.” The inspectors also verified that the licensee and crew assessed and critiqued crew performance accordingly.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 125-Volt DC Battery Chargers

a. Inspection Scope

The inspectors performed maintenance effectiveness reviews of the 125-Volt DC battery chargers. The inspectors reviewed repetitive maintenance activities 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 Part 50.65) requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, functional failure determinations, and current equipment performance status. The review of maintenance effectiveness for the 125-Volt DC battery chargers constituted one inspection procedure sample.

b. Findings

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” having very low safety significance (Green) for failure to accomplish required preventive maintenance (PM) resulting in the 125Vdc, D-108 Station Battery Charger output becoming unstable on several occasions. In January of 2007, the 125Vdc, D-09 Station Battery Charger also failed as a result of failure to perform scheduled PM. In each case the required PM to replace capacitors and printed circuit boards was not performed. The battery chargers became inoperable due to component failures on circuit cards that were not replaced per the PM.

Description: On March 30, 2007, the Power Conversion Products safety-related D-108 Battery Charger became unstable when multiple alarms “D-02/D-04 125V DC Bus Under/over Voltage” occurred. These same alarms had also occurred on

February 19, 2007, and March 24, 2007. Plant Operations considered the charger unstable and removed it from service. Trouble shooting revealed that filter capacitors on the voltage sensing and current limiting board affected the stability of the voltage sensing circuit as well as the charger's output voltage. The output of the voltage sensing circuit is input into an amplification stage. Since the amplifier input contained electrical noise, the amplifier board was feeding forward changes to the firing cards causing changes in the firing angle of the firing pulses to the power thyristors. The result was fluctuations in the output voltage that caused swings in output current on the order of +/- 20 amps.

The inspectors reviewed the failure of the D-108 Battery Charger and determined that the electrolytic capacitors and printed circuit boards were scheduled to be replaced in February 2006 as part of 10-year PM per WO 0415753. This PM was not accomplished as scheduled and the charger failed approximately 13 months later. While it had been over 90 days since the failure and the failed components were still awaiting failure analysis to definitively determine the cause, the inspectors noted that the need to manage the replacement of these battery charger components (capacitors and printed circuit cards) was known to the licensee; however needed PM was not performed.

As part of this maintenance effectiveness review, the inspectors also evaluated a recent failure of the D-09 Westinghouse Station Battery Charger that occurred on January 13, 2007. This charger had accumulated two battery charger related maintenance preventable functional failures within two years therefore exceeding the Maintenance Rule prescribed reliability criteria for station chargers to not exceed accumulatively one maintenance preventable functional failure in 24 months. The licensee considered the safety-related Westinghouse battery chargers to be obsolete and originally scheduled them to be replaced in calendar year 2005/2006.

A review of this recent failure revealed that PM was scheduled to replace battery charger D-09 capacitors and printed circuit boards. The PM was not accomplished and the charger failed on January 13, 2007. The maintenance rule evaluation for this failure stated that the ". . . incident would likely have been avoided if the station could have followed its prescribed maintenance practices, which would have replaced the board with the comparator circuit every 7 to 10 years. However, the normal card replacement intervals were not able to be followed due to obsolescence issues (cards not available)." Also, the licensee had not effectively pursued alternate plans to obtain appropriate replacement parts when it became evident the battery chargers were not going to be replaced as scheduled in 2005 and 2006. As a result, the licensee did not follow prescribed maintenance practices to maintain charger availability and reliability and failures of safety-related, risk-significant equipment occurred.

Analysis: The inspectors determined that the failure to accomplish preventive maintenance to maintain the reliability of the safety-related battery chargers was a performance deficiency and a finding. Failure to accomplish scheduled PMs for safety-related equipment known to have parts reliability issues led to failures and unnecessary unavailability of risk significant equipment. The 125V DC system was already in Maintenance Rule (a)(1) status and the Westinghouse battery chargers were previously identified as needing replacement due to obsolescence and unavailability of parts when the D-09 battery charger failure occurred. These conditions should have resulted in heightened awareness of the need to ensure planned PM was appropriately scheduled and accomplished.

Using IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” dated November 2, 2006, the inspectors concluded that the finding is greater than minor because, if left uncorrected, the finding would become a more significant safety concern in that failures of safety-related battery chargers can significantly challenge the vital 125V DC system. In addition, the finding is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Because these events occurred while the reactor was at power, the significance was evaluated using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 1, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.” Since the inspectors answered “No” to all of the screening questions in the Mitigating System Cornerstone Column, the finding screened as having very low safety significance (Green).

The inspectors also determined that the primary cause of this finding is related to the cross-cutting area of human performance. Specifically, under the component of work control, the licensee failed to plan and coordinate work activities, consistent with nuclear safety. The licensee did not appropriately coordinate work activities to support long-term equipment reliability and maintenance scheduling was not more preventive than reactive as critical preventive maintenance for battery chargers was not performed (H.3(b)).

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to this, established critical preventive maintenance routines to replace capacitors and circuit cards for station battery chargers were not performed when scheduled and the chargers subsequently failed. Because of the very low safety significance of this finding and because the issue was entered into the licensee’s corrective action program (CAP 01085065 and CAP 01071742), the violation is being treated as a NCV, consistent with Section VI.A.1 of NRC Enforcement Policy (NCV 05000266/2007003-01; 05000301/2007003-01).

Licensee corrective actions included replacing the defective parts and returning the chargers to service. An apparent cause evaluation is being performed for the D-108 failure to determine appropriate additional corrective actions. As a result of the D-09 failure, the station is changing its maintenance practices to check operating voltages on circuit cards to make sure that components (such as, Zeners/Diodes/Capacitors) are not degrading. PCRA 1058436-03 was written to change RMP 9359-7C, “DC Station Battery Charger D-09 Maintenance Procedure,” as well as similar procedures for other safety-related Westinghouse battery chargers, to accomplish the actions identified above every 18 months. Additionally, the licensee procured the services of a vendor to re-engineer the affected cards so a supply of parts would be available, included replacement of the obsolete Westinghouse battery chargers on its top 10 equipment list, and is pursuing replacement of the chargers. Scheduled PM routines for battery chargers D-107 and D-108 were completed as required.

.2 Additional Routine Maintenance Rule Samples

a. Inspection Scope

The inspectors performed maintenance effectiveness reviews of the systems listed below. The inspectors reviewed repetitive maintenance activities 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 Part 50.65) requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, functional failure determinations, and current equipment performance status.

For each system reviewed, the inspectors reviewed significant WOs and corrective action program documents to determine whether failures were appropriately identified, classified, and corrected, and if unavailable time was correctly calculated. The reviews of maintenance effectiveness for the following components and systems constituted two inspection procedure samples:

- Unit 1 Turbine Driven Auxiliary Feedwater Pump 1P29; and
- Unit 2 Turbine Driven Auxiliary Feedwater Pump 2P29.

b. Findings

The results of this review are documented in NRC Inspection Report 05000266/2007008.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (71111.13)

.1 Inability to Implement Risk Management Contingency Plan

a. Inspection Scope

The inspectors reviewed the risk assessment and reduced-inventory Orange Path Contingency Plan for the planned SG Nozzle Dam installation during the U1R30 Refueling Outage. The inspectors compared the licensee’s risk management actions contained in the contingency plan, to those actions specified in the licensee’s procedures for the assessment and management of risk associated with maintenance activities. The inspectors assessed whether evaluation, planning, control, and performance of the work, was done in a manner which reduced risk and minimized the duration, where practical, and whether contingency plans could be implemented. The review of the maintenance risk assessment and contingency plan for the reduced-inventory Orange Path condition for April 7 and 8, 2007, constituted one inspection procedure sample.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR 50.65(a)(4) after the licensee failed to adequately manage the risk associated with the installation of the Unit 1 SG Nozzle Dams, a reduced inventory and Orange Qualitative Risk Condition. Specifically, the contingency plan stated, in part, that an uncontrolled reactor coolant system inventory loss would be mitigated with the use of Shutdown Emergency Procedure SEP-2, "Cold Shutdown LOCA." However, the inspectors noted that certain critical equipment required in SEP-2 was not available and no contingencies were established for the unavailable equipment.

Description: On April 7, 2007, the licensee entered a planned reduced inventory condition in the reactor coolant system in order to install the SG Nozzle Dams, which placed Unit 1 in an orange risk condition. In accordance with the licensee's administrative procedures, the orange risk condition required the planning and development of a reduced-inventory orange path contingency plan, which was approved by the Plant Operations Review Committee. The inspectors observed the licensee enter the reduced inventory condition with no issues. Subsequent to this, the inspectors performed walkdowns to ensure that equipment relied upon in the contingency plan for the loss of key safety functions (core cooling or inventory) was available.

While in containment, the inspectors noted that the containment sump isolation valves were not available due to a modification of the containment sump screens in response to GSI-191. The inspectors queried licensee and contract personnel working on the project, to ascertain if contingencies were in place in the field should the reactor coolant system inventory be lost in an uncontrolled manner, to ensure at least one train would be made available. The inspectors determined there were no contingencies in place to ensure the availability of containment sump isolation valves.

The inspectors noted that the U1R30 Reduced-Inventory Orange Path Contingency Plan, Revisions 0 and 1, Section 6, "Loss of Key Safety Function," stated, in part, that should the reactor coolant system inventory be lost in an uncontrolled manner, Shutdown Emergency Procedure 2, "Cold Shutdown LOCA," would be entered and used to mitigate the accident. Step 17 was a continuous action step, which required operators to verify that the containment sump recirculation was not required, and, if the response was not obtained, to align the residual heat removal system for sump recirculation. The inspectors questioned the Shift Manager and licensee Outage Control Center staff, as to why neither sump recirculation path was currently available. The licensee subsequently put immediate contingency actions in place, to ensure, that the containment sump paths would be available.

Finally, the inspectors, also noted that while the licensee had a containment checklist in place to ensure containment penetrations could be secured within the time to boil, licensee's controls to ensure this task could be completed successfully were not rigorous. Specifically, the licensee did not assign control room and auxiliary operators the duty of ensuring containment integrity is set, if required, and walk-throughs to ensure the list of penetrations could be closed within the time to boil were not required. The licensee also initiated immediate corrective actions to ensure the appropriate rigor was assigned to ensure containment integrity would be set within the time to boil.

Analysis: The licensee's failure to adequately manage plant risk during a reduced reactor coolant system inventory period by ensuring that the contingency plan could be implemented, in the event of a loss of a key safety function was a performance deficiency and a finding that warranted a significance evaluation. The inspectors concluded that the finding is greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," November 2, 2006, because the finding is associated with the Mitigating Systems cornerstone of reactor safety. The finding affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences, and the attributes of configuration control and equipment performance due to the shutdown equipment lineup and unavailability of equipment. In addition the finding was related to the licensee's failure to effectively manage significant compensatory measures for this orange risk condition.

The inspectors initially evaluated the finding using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process;" however, in accordance with Section 3 of Appendix K, this appendix does not apply because the shutdown risk assessment is a qualitative analysis of plant risk due to maintenance activities. Therefore, the inspectors used risk insights available in conjunction with NRC management review. The inspectors evaluated the finding using IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The finding screened as very low safety significance (Green), because the finding did not meet the criteria for a Phase 2 or Phase 3 Analysis, as specified in Appendix G, Attachment 1, Checklist 1, "PWR Hot Shutdown Operation: time to Core Boiling < 2 Hours."

The inspectors also determined that the primary cause of this finding is related to the cross-cutting area of human performance. Specifically, under the component of work control, the licensee did not appropriately plan work activities by incorporating the need for planned contingencies and compensatory actions, ensuring that equipment relied upon for contingencies remained available (H.3(a)).

Enforcement: 10 CFR Part 50.65(a)(4), requires, in part, that the licensee assess and manage the increase in risk that may result from proposed maintenance activities. Contrary to the above, on April 7, 2007, the licensee failed to adequately manage the increase in risk associated with the installation of the Unit 1 SG Nozzle Dams. Specifically, the licensee had developed risk management actions contained in the Reduced-Inventory Orange Path Contingency Plan; however, the inspectors identified that the equipment required by the licensee's contingency plan for a loss of a key safety function was not available and no contingencies were established at the time the reduced inventory condition was entered. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program, CAP 01086452, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2007003-02).

The licensee took immediate corrective actions to implement contingencies to ensure the equipment required by the contingency plan would be available. In addition, for the remaining two orange paths in the refueling outage, the licensee revised the contingency plans, such that the equipment relied upon for the loss of key safety function during reduced inventory would be available, or alternative equipment was specified to perform the safety function. At the end of the inspection period, the licensee was continuing to perform a causal evaluation.

.2 Additional Maintenance Risk Assessment and Emergent Work Evaluation Samples

a. Inspection Scope

The inspectors reviewed risk assessments for planned and emergent maintenance activities during the specified work weeks. During these reviews, the inspectors compared the licensee's risk management actions to those actions specified in the licensee's procedures for the assessment and management of risk associated with maintenance activities. The inspectors assessed whether evaluation, planning, control, and performance of the work were done in a manner to reduce the risk and minimize the duration where practical, and whether contingency plans were in place where appropriate.

The inspectors used the licensee's daily configuration risk assessment records, observations of shift turnover meetings, and observations of daily plant status meetings to determine whether the equipment configurations were properly listed. The inspectors also verified that protected equipment was identified and controlled as appropriate, and that significant aspects of plant risk were communicated to the necessary personnel. The reviews of maintenance risk assessment and emergent work evaluation constituted six inspection procedure samples:

- Planned and emergent maintenance during the week of April 1, 2007;
- Planned and emergent maintenance during the week of April 9, 2007;
- Planned and emergent maintenance during the week of April 15, 2007;
- Planned and emergent maintenance during the week of April 23, 2007;
- Reduced Inventory Orange Path Contingency for removal of nozzle dams during the week of April 20, 2007; and
- Planned and emergent maintenance during the week of May 20, 2007.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Failure to Comply with TS 5.6.5

a. Inspection Scope

The inspectors reviewed the operability evaluation associated with a condition report written in response to the inspectors questions concerning a potential failure to update the Pressure Temperature Limit Report, required by TS 5.6.5.c. The inspectors reviewed design basis information, the FSAR, TS requirements, and licensee procedures to determine the technical adequacy of the operability evaluations. In addition, the inspectors determined whether compensatory measures were implemented, as required. The inspectors assessed whether system operability was properly justified and that the system remained available, such that no unrecognized increase in risk occurred. The review of the operability evaluation associated with Condition Reports AR 01092944 and AR 01093301 constituted one inspection procedure sample.

b. Findings

Introduction: The inspectors identified an Unresolved Item for the apparent failure to be in compliance with TS 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)." Specifically, the licensee has not been in compliance with Section 5.6.5(c), which requires, in part, that the licensee shall provide the NRC upon issuance, the PTLR for each reactor vessel fluence period and for any revision or supplement thereto.

Description: Technical Specification 5.6.5.(b), requires, in part, that the analytical methods used to determine RCS pressure and temperature limits be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated October 6, 2000, and July 23, 2001. Technical Specification 5.6.5(c), requires, in part, that the licensee provide the NRC upon issuance, the PTLR for each reactor vessel fluence period and for any revision or supplement thereto.

The July 23, 2001, approval concluded that the fluence for the Unit 1 Reactor Vessel was effective through 25.6 Effective Full Power Years which would be exceeded on October 23, 2003. The licensee issued a letter on June 27, 2003, providing an update which revised the effective date to February 2004. The inspectors also noted that the PTLR has not been submitted to the NRC since December 20, 2002. Finally, the inspectors noted that there were several engineering evaluations performed using the methods that had not been submitted to the NRC. However, on December 14, 2006, the licensee submitted a License Amendment Request to use a different PTLR methodology.

The licensee performed an operability evaluation utilizing the new methodology and determined a new PTLR date of June 2008 for the Unit 1 Reactor Vessel. The inspectors reviewed and verified the licensee's operability determination.

Therefore, the issue will be considered an Unresolved Item pending further NRC review of the licensee's causal evaluations and license amendment request (URI 05000266/2007003-03).

.2 Additional Operability Evaluation Samples

a. Inspection Scope

The inspectors reviewed selected operability evaluations associated with issues entered into the licensee's corrective action program. The inspectors reviewed design basis information, the FSAR, TS requirements, and licensee procedures to determine the technical adequacy of the operability evaluations. In addition, the inspectors determined whether compensatory measures were implemented, as required. The inspectors assessed whether system operability was properly justified and that the system remained available, such that no unrecognized increase in risk occurred. The reviews of the following operability evaluations constituted three procedure samples:

- CAP 01085701; Transformer 1X03, 345-kV to 13.8kV Degraded Voltage;
- CAP 01090876, RCS Pressure Transmitter, 1-PT-40 Configuration Inconsistencies;
- CAP 01098358-01; Operability Evaluation for Moisture in Unit 2 TDAFW pump

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Inadequate Preventive Maintenance of Breaker Mechanism Operated Control Switches

a. Inspection Scope

During completion of the post-maintenance test inspection procedure samples, the inspectors observed in-plant activities and reviewed procedures and associated records to determine whether:

- Testing activities satisfied the test procedure acceptance criteria;
- Effects of the testing were adequately addressed prior to the testing;
- Measuring and test equipment calibration was current;
- Test equipment was within the required range and accuracy;
- Applicable prerequisites described in the test procedures were satisfied;
- Affected systems or components were removed from service in accordance with approved procedures;
- Testing activities were performed in accordance with the test procedures and other applicable procedures;
- Test data and results were accurate, complete, and valid;
- Test equipment was removed after testing;

- Equipment was returned to a position or status required to support the operability of the system in accordance with approved procedures; and
- All problems identified during the testing were appropriately entered into the corrective action program.

The inspectors reviewed the activities associated with the testing of the EDG G02 when aligned to the alternate 4.16kV Bus 1A05, which constituted one quarterly inspection procedure sample.

b. Findings

Introduction: The inspectors identified a NCV of 10 CFR Part 50.65 (a)(3), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," of very low safety significance (Green), for the failure to incorporate available internal and external operating experience (OE) pertaining to 4.16kV switchgear cubicle mechanism operated control (MOC) switch assemblies. Preventive Maintenance procedures for Westinghouse 4.16kV switchgear cubicles had not been revised to incorporate important MOC switch checks to ensure freedom of movement, alignment of the linkage, inspection of contacts for deformation, and verification of contact continuity.

Discussion: During TS-82, "Emergency Diesel Generator G02 Monthly" testing on April 14, 2007, an attempt was made at paralleling G02 to it's alternate bus (1A-05) per the normal procedural sequence. Although conditions established for paralleling the diesel generator were normal, the diesel loading response was not. When the breaker closed, load increased to 300kW without operator action and then continued at a linear rate to pick up load. A decision was made to open the output breaker as load passed thru 1000kW. Subsequent trouble shooting revealed that the MOC switch in breaker cubicle 1A52-57 (normal ac supply to the 4.160kV emergency bus 1A05) had a degraded contact set, which prevented the governor droop circuit from functioning upon breaker closure. Therefore, the governor remained in the isochronous mode, which caused the identified condition. No other degraded contact sets of the cell switch were found.

The inspectors noted that a similar event occurred on July 27, 2003, when the G02 Emergency Diesel Generator to bus 2A05 breaker was closed. The diesel generator immediately assumed load and continued to assume load until load was turned using the governor control switch, at which time load was shed and continued to decrease until corrected by the operator. The diesel generator was unloaded and declared out of service (OOS) by the shift manager. The investigation of this event identified that a slight misadjustment of the operating mechanism/linkage for the breaker cubicle MOC switch precluded closure of the contact that put the G02 governor into droop mode. The licensee concluded this was likely caused by lack of a comprehensive PM program on the subject breaker cubicle. Another event occurred in October 2003, during performance of Procedure ORT-3, "Safety Injection Actuation with Loss of Engineered Safeguards," when the diesel generator load steadily increased. This problem was caused by misalignment of the linkage between the circuit breaker and the cubicle. The licensee developed corrective action, CA032919, to provide an inspection program/procedure for critical Westinghouse switchgear cubicles to include and to obtain design specifications for MOC switch pantograph assemblies from the vendor and develop inspection criteria for the MOC switch assemblies in Westinghouse 50DH350 switchgear cubicles.

The inspectors reviewed the licensee's PM procedures for 4.16kV Westinghouse switchgear and identified that: (1) there were no requirements to inspect the MOC switch assembly for freedom of movement, to verify alignment of the linkage between the circuit breaker and cubicle, or to inspect contacts for deformation or verify continuity; (2) there was some guidance in a 12 year PM per RMP 9370 "Bus Inspection and Cleaning," Section 5.7.11, to check that the MOC switch pivot bracket mounting hardware is snug and tight and that the bracket spring leaves are properly aligned. Inspectors noted that failure to perform appropriate maintenance activities for the 4.16kV cubicle, 1A52-57, resulted in failure of the breaker cubicle MOC switch to function as designed and subsequently resulted in failure of the diesel to load properly. While paralleling of the diesels is not a safety-related function, there are contacts on the same MOC switch that provide a close permissive to the diesel breaker, which if not functional, would prevent automatic closure of the breaker. In this instance, additional degraded adjustment of the switch could have easily affected the breaker permissive contact.

Analysis: The inspectors determined that the licensee's failure to fully incorporate relevant internal and external OE into its maintenance procedures is a performance deficiency and a finding. Using IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated November 2, 2006, the inspectors concluded that the finding is greater than minor, because if left uncorrected the finding would become a more significant safety concern. The finding also affects the procedure quality attribute of the Mitigating System cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage).

The inspectors assessed this finding using IMC 0609, Significance Determination Process, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situation." (Although Unit 1 was shutdown for a refueling outage, Appendix A is equally applicable to this procedure quality performance deficiency). Since the inspectors answered "No" to all of the screening questions in the Mitigation Systems Cornerstone Column, the finding screened as having very low safety significance (Green). Additionally, the inspectors determined the primary cause of the finding is related to the cross-cutting area of Problem Identification and Resolution within the component of OE as the licensee failed to implement and institutionalize OE through changes to station processes, procedures, equipment, and training programs (P.2(b)).

Enforcement: 10 CFR Part 50.65(a)(3) requires, in part, that preventive maintenance activities be evaluated at least every refueling cycle and take into account, where practical, industry-wide OE. An adjustment shall then be made where necessary to ensure that the objective of preventing failures of structures, systems, and components through maintenance is appropriately balanced against the objective of minimizing unavailability of structures, systems, and components due to monitoring or preventive maintenance.

Contrary to the above, as of April 1, 2007, the licensee failed to: (1) incorporate available internal and external OE into its maintenance procedures for Westinghouse 4.16kV breaker cubicle MOC switches and (2) use this important OE information during maintenance activities to ensure the objective of preventing component failures by performing adequate PM. Because this violation is of very low safety significance (Green) and because the issue was entered into the licensee's corrective action program (CAP 01091966), the violation is being treated as a NCV, consistent with Section VI.A.1 of NRC Enforcement Policy (NCV 05000266/2007003-04; 05000301/2007003-04).

Licensee corrective actions included troubleshooting and adjusting the 1A52-57, 4.16kV breaker cubicle MOC switch, performing post maintenance testing and submitting procedure changes to RMP 9370 "Bus Inspection and Cleaning," to implement PM for the 4.16kV Westinghouse breaker cubicle MOC switches. The licensee also evaluated its PM for ABB 4.16kV breaker cubicles and determined it was adequate.

.2 Additional Post Maintenance Testing Samples

a. Inspection Scope

During completion of the post-maintenance test inspection procedure samples, the inspectors observed in-plant activities and reviewed procedures and associated records to determine whether:

- Testing activities satisfied the test procedure acceptance criteria;
- Effects of the testing were adequately addressed prior to the testing;
- Measuring and test equipment calibration was current;
- Test equipment was within the required range and accuracy;
- Applicable prerequisites described in the test procedures were satisfied;
- Affected systems or components were removed from service in accordance with approved procedures;
- Testing activities were performed in accordance with the test procedures and other applicable procedures;
- Jumpers and lifted leads were controlled and restored where used;
- Test data and results were accurate, complete, and valid;
- Test equipment was removed after testing;
- Equipment was returned to a position or status required to support the operability of the system in accordance with approved procedures; and
- All problems identified during the testing were appropriately entered into the corrective action program.

The activities listed below were reviewed by the inspectors and constituted seven quarterly inspection procedure samples:

- Unit 1 Polar Crane Return to Service and Troubleshooting, the week of April 8, 2007;
- Unit 1 Charging Pumps Suction from the Refueling Water Storage Tank Valve CV-112B following preventive maintenance, the week of April 8, 2007;

- Unit 1 Residual Heat Removal Valves RH-720, RH-700, and RH-701 following preventive maintenance the week of April 15, 2007;
- Unit 1 Reactor Coolant Loop ‘A’ Cold Leg Normal Charging Isolation Valve following a breaker replacement the week of April 15, 2007;
- Unit 1 Residual Heat Removal Pump 1P-10A following preventive maintenance the week of April 15, 2007;
- Unit 1 Emergency Core Cooling System Recirculation Isolation Valves SI-850A and SI-850B following preventive maintenance the week of April 23, 2007; and
- Unit 1 Turbine Driven Auxiliary Feedwater Pump 1P-29, following maintenance and repair the week of June 18, 2007.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Routine Refueling Outage Inspection Activities

a. Inspection Scope

The inspectors observed activities during the Unit 1 refueling outage (U1R30) conducted from March 31 through May 7, 2007. These inspection activities constituted one refueling outage inspection sample.

This inspection consisted of an in-office review of the licensee’s outage schedule, safe shutdown plan, and administrative procedures governing the outage; and periodic observations of equipment alignment and plant and control room outage activities. Specifically, the inspectors determined the licensee’s ability to effectively manage elements of shutdown risk pertaining to reactivity control, decay heat removal, inventory control, electrical power control, and containment integrity.

The inspectors conducted the following inspection activities:

- Attended outage management turnover meetings to determine whether the current shutdown risk status was accurate, well understood, and adequately communicated;
- Performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;
- Observed the operability of reactor coolant system instrumentation and compared channels and trains against one another;
- Performed in-plant walkdowns to observe ongoing work activities; and
- Conducted in-office reviews of selected issues that the licensee entered into its corrective action program to determine whether identified problems were being entered into the program with the appropriate characterization and significance.

Additionally, the inspectors performed the following specific in-plant activities:

- Performed Mode 3 walkdowns at the start and end of the refueling outage to check for active boric acid leak indications;
- Observed head lift activities and containment closure and integrity;
- Observed core unloading activities in the containment, spent fuel pool and control room and reactivity control;
- Observed reduced inventory, mid-loop operations and inventory controls;
- Observed outage clearance activities;
- Verified the status and configuration of electrical systems against TSs and the licensees's outage risk management plan;
- Verified that the flow paths, configurations, and alternative means for inventory addition were consistent with the outage risk plan;
- Observed core reload from containment;
- Observed operators align the RHR system for shutdown cooling and verified the system was functioning properly to remove decay heat;
- Observed placement of the over-pressure protection system into operation;
- Observed lifting and transport of the reactor vessel head in preparation for core offload;
- Performed a closeout inspection of the Unit 1 containment, including a review of the emergency core cooling sump final installation ;
- Reviewed shutdown margin calculations;
- Reviewed spent fuel pool cooling and service water pump configurations during partial core offload;
- Observed operation of the fuel handling bridges in containment and over the spent fuel pool;
- Performed a containment closeout inspection;
- Reviewed mode-change checklists to verify that selected requirements were met while transitioning from the refueling mode to full power operation;
- Observed portions of low power physics testing and approach to criticality; and
- Observed portions of the plant ascension to full power operations.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Failure to Establish Appropriate Test Condition for Leak-Rate Testing Outside Containment

a. Inspection Scope

The inspectors observed in-plant activities and reviewed the performance of Test Procedure IT-530C, “Leakage Reduction and Preventive Maintenance Program,” which constituted one routine test quarterly inspection procedure sample. The inspectors verified the following aspects of this surveillance activity to determine whether:

- Preconditioning occurred;
- Effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- Acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- Plant equipment calibration was correct, accurate, and properly documented; as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the FSAR, procedures, and applicable commitments;
- Measuring and test equipment calibration was current;
- Test equipment was used within the required range and accuracy;
- Applicable prerequisites described in the test procedures were satisfied;
- Test frequencies met TS requirements to demonstrate operability and reliability;
- Tests were performed in accordance with the test procedures and other applicable procedures;
- Where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- Equipment was returned to a position or status required to support the performance of its safety functions; and
- All problems identified during the testing were appropriately documented and dispositioned in the corrective action program.

b. Findings

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," having very low safety significance (Green) for the failure to have procedures appropriate to the circumstances, which established the appropriate test conditions for primary coolant sources testing outside containment. Specifically, testing procedures, which satisfied TS 5.5.2, "Primary Coolant Sources Outside Containment," did not ensure that residual deposits of boric acid on the containment spray, high head and low head safety injection systems were removed, so that, active system fluid leaks could be identified, as required during the tests.

Description: The inspectors observed the licensee perform IT-530C, "Leakage Reduction and Preventive Maintenance Program Train "A" High Head Safety Injection and Residual Heat Removal 'Piggyback Test' Unit 1." The current licensing basis for total Emergency Core Cooling System leakage, as stated in the Containment Leakage Rate Testing Basis Document, is less than 400 cubic centimeters per minute. Plant Operations personnel performed the test in accordance with the procedure, and identified three visible active leaks associated with the systems during the test.

However, the inspectors noted that the initial conditions for the test did not require that large amounts of boric acid deposits, located primarily on pump seals and valve packing, were removed prior to the performance of the test. The inspectors questioned why this was not required, since there were deposits of boric acid on some equipment, which would have masked potential active leakage from the system and affected the licensee's ability to accurately determine the system leakage. The licensee subsequently initiated a condition report to address the issue.

Following the performance of the IT-530C test, one of the leaks the licensee identified was repaired. As part of the repair for cleanliness and contamination control, the licensee decontaminated the "A" Safety Injection Pump inboard seal area. This area had significant deposits of boric acid in the mechanical seal catch basin and did not exhibit signs of active leakage during the IT-530C test. Although the mechanical seal itself was not worked on as part of the maintenance, during the post maintenance test, the inboard seal developed an active leak following removal of the dried boric acid.

Analysis: The inspectors determined that the failure to ensure that the test procedures associated with the primary coolant sources outside containment established appropriate test criteria to ensure identification of all active leaks in the containment spray, high head and low head safety injection systems was a performance deficiency and a finding. The inspectors concluded that the finding is greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," November 2, 2006, because the finding is associated with the containment barrier cornerstone of reactor safety. In addition, the finding is associated with the procedure quality attribute of the barrier integrity cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events.

The inspectors evaluated the finding using IMC 0609, "Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." The finding screened as very low safety significance (Green) because the finding did not: represent the degradation of the radiological barrier function provided for the auxiliary building; represent a degradation of the barrier function of the control room; and did not represent an actual open pathway in the physical integrity of reactor containment.

The inspectors also determined that the primary cause of this finding is related to the cross-cutting area of human performance. Specifically, under the component of resources, the licensee failed to ensure that procedures were adequate and accurate to assure nuclear safety (H.2(c)).

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality be prescribed by documented instructions or procedures of a type appropriate to the circumstances. Contrary to this, station procedures for implementing the program required by TS 5.5.2, "Primary Coolant Sources Outside Containment," did not ensure that the appropriate test conditions were established prior to the commencement of testing. Specifically, the procedures did not ensure that residual deposits of boric acid on the containment spray, high head and low head safety injection systems were removed, so that all active system fluid leaks could be identified, as required, during the tests. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (CAP 01090269), the violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2007003-05).

The licensee took immediate corrective actions to address the issue, and at the end of the inspection period the licensee continued to evaluate the causes associated with this finding.

.2 Additional Surveillance Testing Samples

a. Inspection Scope

During completion of the inspection procedure samples, the inspectors observed in-plant activities and reviewed procedures and associated records to determine whether:

- Preconditioning occurred;
- Effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- Acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- Plant equipment calibration was correct, accurate, and properly documented; as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the FSAR, procedures, and applicable commitments;
- Measuring and test equipment calibration was current;
- Test equipment was used within the required range and accuracy;
- Applicable prerequisites described in the test procedures were satisfied;
- Test frequencies met TS requirements to demonstrate operability and reliability;
- Tests were performed in accordance with the test procedures and other applicable procedures;
- Jumpers and lifted leads were controlled and restored where used;
- Test data and results were accurate, complete, within limits, and valid;
- Test equipment was removed after testing;
- Where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers Code, and reference values were consistent with the system design basis;
- Where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- Where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- Where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- Prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- Equipment was returned to a position or status required to support the performance of its safety functions; and
- All problems identified during the testing were appropriately documented and dispositioned in the corrective action program.

During this inspection period, the inspectors completed the following inspection procedure samples, which constituted four inservice tests and four routine tests, for a total of eight quarterly inspection procedure samples:

- Unit 1 SI actuation with loss of engineered safeguards alternating current for train "A," Procedure ORT-3A, conducted the week of April 1, 2007;
- Unit 1 SI actuation with loss of engineered safeguards alternating current for train "B" Procedure ORT-3B, conducted the week of April 1, 2007;
- Unit 1 High Head Safety Injection Full Flow Inservice Test, Procedure IT-760, conducted the week of April 8, 2007;
- Unit 1 Residual Heat Removal Pumps and Valve Inservice Test in Decay heat Removal Mode, Procedure IT-03A, during the week of April 15, 2007;
- Unit 1 Auxiliary Feedwater AMSAC Testing, Procedure ORT-3C, conducted the week of April 23, 3007;
- Unit 1 Containment Spray and Sequence Testing, Procedure ORT-6C, conducted the week of April 23, 3007;
- Inservice Testing of Service Water Pump P-32A, Procedure IT-7A, conducted the week of May 27, 2007; and
- Inservice Testing of the Unit 1 Turbine Driven Auxiliary Feedwater Pump 1P-29, IT-8A, conducted the week of June 4, 2007.

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 Plant Walkdowns/Boundary Verifications and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors identified work being performed within high and locked high radiation areas of Unit 1 Containment and other potentially exposure significant work activities and reviewed radiation work permit (RWP) packages and radiation surveys for these areas. The inspectors evaluated the radiological controls to determine whether these controls including postings and access control barriers were adequate. These work activities included:

- Reactor Head Lift;
- Fuel Moves;
- SG Eddy Current Testing; and
- SG Hot Leg Ultrasonic Testing of Indication.

The inspectors reviewed RWPs and work packages which governed activities in radiologically significant areas to identify the work control instructions and control barriers that had been specified. For these activities, electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications.

The inspectors walked down and surveyed radiologically significant area boundaries in the Unit 1 Containment, the Primary Auxiliary Building and in the Unit 1 Facade to determine whether the prescribed radiological access controls were in place, licensee postings were complete and accurate, and physical barricades/barriers were adequate. During the walkdowns, the inspectors challenged access control boundaries to determine whether high radiation area (HRA) and locked high radiation area (LHRA) access was controlled in compliance with TS and the requirements of 10 CFR 20.1601, and was consistent with Regulatory Guide 8.38, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants.”

The inspectors reviewed job planning records and interviewed licensee representatives to determine whether there were airborne radioactivity areas in the plant with a potential for individual worker internal exposure to exceed 50 millirem committed effective dose equivalent. Engineering control effectiveness such as the use of high efficiency particulate air ventilation systems and the use of respiratory protection were evaluated for worker protection. Radiological surveys for work areas having a potential for transuranic isotopes were reviewed to determine whether the licensee had assessed that potential and provided appropriate worker protection as applicable. The licensee’s process and procedure for internal dose assessment was reviewed to determine whether it was technically sound, satisfied 10 CFR 20.1204, and included assessment of the impact of hard to detect radionuclides such as pure beta and alpha emitters, as applicable. The inspectors reviewed internal dose assessment results for any workers that had intakes during the current Unit 1 outage. No worker internal exposures greater than 50 millirem committed effective dose equivalent occurred for the period reviewed by the inspectors.

These reviews represented five inspection samples.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the results of a radiation protection department self-assessment related to the radiological access control program and the corrective action program database along with individual condition reports related to the radiological access and exposure control programs to determine whether identified problems were entered into the corrective action program for resolution. In particular, the inspectors reviewed radiological issues which occurred over the four-month period that preceded the inspection including the review of any HRA radiological incidents (non-performance indicator (PI) occurrences identified by the licensee in high and locked high radiation areas) to determine whether follow-up activities were 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;
- Resolution of NCVs tracked in the corrective action system;
- Identification of contributing causes; and
- Identification and implementation of corrective actions.

The inspectors reviewed the licensee's process for problem identification, characterization, and prioritization and determined whether problems were entered into the corrective action program and were being resolved in a timely manner. For potential repetitive deficiencies or possible trends, the inspectors determined whether the licensee's self-assessment activities were capable of identifying and addressing these deficiencies, when applicable.

The inspectors reviewed the licensee's documentation for all potential PI events occurring since the last radiological access control inspection performed in December 2006 to determine whether any of these events involved dose rates greater than 25 Rem/hour at 30 centimeters or greater than 500 Rem/hour at 1 meter or involved unintended exposures greater than 100 millirem total effective dose equivalent (or greater than 5-Rem shallow dose equivalent or greater than 1.5 Rem lens dose equivalent). None were identified.

Additionally, the inspectors reviewed the circumstances surrounding two HRA/LHRA access control issues identified by the licensee during the two weeks preceding the inspection. These issues are summarized in Section 4OA7.

These reviews represented four inspection samples. Specifically, the samples pertained to the licensee's self-assessment capabilities, its problem identification and resolution program for radiological incidents, a review of the licensee's ability to identify and address repetitive deficiencies, and a review of those radiological incidents and potential PI occurrences of greatest radiological risk.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed selected jobs being performed in HRAs, LHRAs and potential airborne radioactivity areas to assess those work activities that presented the greatest radiological risk to workers. The work included SG inspections and related activities, reactor vessel head lift, and various other work activities in the Unit 1 Containment Building. Radiation survey information to support these work activities was reviewed by the inspectors. The radiological job requirements were assessed for adequacy, and field observations were made to determine whether ALARA measures were implemented as necessary to reduce dose. The inspectors also attended the pre-job briefing for one of these activities to assess the adequacy of the information exchanged.

Job performance was observed to determine whether 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, air sampling, contamination controls, and the overall work oversight provided by the radiation protection technicians.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.4 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance for conformity with radiation protection work requirements and to determine whether workers were aware of the radiological conditions, the RWP controls and limits in place, and whether their performance had accounted for the level of radiological hazards present.

The inspectors also reviewed radiological problem reports, which found the cause of the event was due to radiation worker errors, to determine whether there was an observable pattern traceable to a similar cause and to determine whether this matched the corrective action approach taken by the licensee to resolve the identified problems.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.5 Radiation Protection Technician Proficiency

a. Inspection Scope

During job observations and general plant walkdowns, the inspectors evaluated RPT performance with respect to radiation protection work requirements, conformance with requirements specified in the RWP, and assessed overall proficiency with respect to radiation protection requirements and health physics practices.

The inspectors reviewed selected radiological problem reports generated since December 2006 to determine the extent of any specific problems or trends that may have been caused by deficiencies with RPT work control and to determine whether the corrective action approach taken by the licensee to resolve the reported problems, when applicable, was adequate.

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 refueling outage exposure history, current exposure trends for U1R30 and ongoing outage activities in order to assess current dose performance and exposure challenges. Additionally, the inspectors reviewed the exposure results from the Fall 2006 Unit 2 Cycle 28 Refueling Outage (U2R28) and compared the results with the projected values. This included determining the plant's current three-year rolling average for collective exposure in order to provide a perspective of significance for any resulting inspection finding assessment.

The inspectors reviewed U1R30 work and the associated exposure (dose) projections, including time/labor estimates and historical dose data for the following work activities which were likely to result in the highest personnel collective exposures:

- SG Nozzle Dam, Manway and Hand Hole Installation/Removal;
- Reactor Vessel Head Lift/Set;
- SG Eddy Current Testing and Sludge Lancing;
- General Maintenance in Containment; and
- Containment Scaffolding and Insulation Activities.

The inspectors determined site specific trends in collective dose based on plant historical exposure for similar work activities. The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and evaluated those processes used for U1R30 to develop dose projections, including time/labor estimates, and to track work activity specific exposures.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors obtained the licensee's list of U1R30 refueling outage work ranked by estimated exposure and reviewed the following work activities that were projected to expend radiation dose of 1 rem or greater or were otherwise potentially radiologically significant activities:

- SG Eddy Current Testing;
- Reactor Vessel Head Removal/Reinstallation;
- Reactor Coolant Pump Inspection and Maintenance;
- SG Sludge Lancing;
- Sump "B" Modification;
- SG Manway/Diaphragm Removal and Installation;
- SG Nozzle Dam Installation and Removal; and
- Ultrasonic Testing of SG Hot Leg Channel Head.

For each of the activities listed above, the inspectors reviewed the RWP and the work package which consisted of various radiological work assessment forms and ALARA planning information including total effective dose equivalent ALARA evaluations (i.e., respirator evaluations), as applicable. The reviews were performed in order to determine whether the licensee had established radiological engineering controls and dose mitigation criteria that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into activities that were based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the exposure results achieved through approximately one-third of the 35-day refueling outage including the person-rem expended with the doses projected in the licensee's ALARA planning for the above listed work activities and for other selected outage activities. The projected versus actual dose expenditures for the fall 2006 U2R28 refueling outage were likewise reviewed by the inspectors. Reasons for inconsistencies between intended (projected) and actual work activity doses as well as time/labor differences were examined to determine whether the activities were planned adequately and to determine whether the licensee was cognizant of work planning deficiencies.

These reviews represented three 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 licensee's assumptions and basis for its collective refueling outage exposure estimate and for individual outage job estimates, and evaluated the methodology and practices for projecting work activity specific exposures. This included evaluating both dose rate and time/labor estimates for adequacy compared to historical station specific data.

The inspectors reviewed the licensee's process for adjusting outage exposure estimates when unexpected changes in scope, emergent work or other unanticipated problems were encountered which could significantly impact worker exposures. This included determining whether adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to effectively plan or control the work. No jobs completed during the previous refueling outage (U2R28) or taking place during the current outage (U1R30) had exceeded or were likely to exceed the NRC's significance determination process collective dose thresholds (5-person rem criterion and greater than 50 percent of the planned dose).

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Controls

a. Inspection Scope

The inspectors observed a variety of ongoing outage work activities including fuel moves, reactor head set and various SG related activities to assess the adequacy of the ALARA initiatives and the job specific radiological controls.

The licensee's use of ALARA controls for these work activities was evaluated to determine whether:

- The licensee developed and effectively used engineering controls to achieve dose reductions and to verify that the controls were consistent with the licensee's ALARA work packages.
- Workers were cognizant of work area radiological conditions, utilized low dose waiting areas and that radiological oversight of work was adequate.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.5 Radiation Worker and Radiation Protection Technician Performance

a. Inspection Scope

Radiation worker and RPT performance was assessed by the inspectors through direct observation focusing on outage activities performed in the Unit 1 containment building. The inspectors determined whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and the tools to be used for the job, by utilizing low dose waiting areas, and assessing whether workers had knowledge of the radiological conditions and adhered to the ALARA requirements for the work activity. Job support and the communications provided by the radiation protection staff were also evaluated by the inspectors.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a Inspection Scope

The inspectors reviewed radiation protection program related ARs generated during the initial 10 days of the refueling outage and for the four months that preceded the outage and licensee staff members were interviewed to assess whether 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 and contributing causes;
- Resolution of NCVs tracked in the corrective action system; and
- Identification and implementation of effective corrective actions.

These reviews represented one inspection samples.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Resident Inspector Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to determine whether issues were entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors also reviewed all condition reports written by licensee personnel during the inspection period. The condition reports written by the licensee as a result of inspectors' observations are included in the list of documents in the Attachment to this report.

b. Findings

No findings of significance were identified.

.2 Resident Inspector Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of licensee trending activities to determine whether emerging adverse trends might indicate the existence of a more significant safety issue not previously identified. The inspectors also determined whether the trends were entered into the licensee's corrective action system at an appropriate threshold, and timely corrective actions were planned or implemented by the licensee. The effectiveness of licensee trending activities was assessed by comparing trends identified by the licensee with those trends identified by the NRC during the daily reviews of condition reports, as discussed in Section 4OA2.1 of this report.

The inspectors' review considered the six-month period of October 2006 to March 2007, although some examples extended beyond those dates when the scope of the trend warranted. The inspectors also reviewed the Department Roll-Up Meeting Reports and Quarterly Department Roll-Up Meeting Summary from October 2006 to March 2007. Finally, the inspectors reviewed the 4rd quarter 2006 and 1st quarter 2007 human performance trend reports. The inspectors' review was focused on licensee human performance errors, but also considered the results of daily inspector corrective action program item screening, licensee trending efforts, and licensee human performance results. This inspection effort constituted one semi-annual trending inspection procedure sample.

b. Findings

No findings of significance were identified.

.3 Selected Issue Followup: Confirmatory Order Implementation

a. Inspection Scope

This issue followup is to evaluate the Nuclear Management Company's (NMC's) corrective actions in response to the January 3, 2007, Confirmatory Order issued to the licensee. The modifications to the license as a result of the Confirmatory Order contained the following four items:

1) NMC shall review, revise, and communicate to NMC employees and managers its policy relating to the writing of corrective action program reports, and provide training to NMC employees and managers to clarify management's expectation regarding the use of the program with the goal to ensure employees are not discouraged, or otherwise retaliated or perceived to be retaliated against, for using the corrective action program;

2) NMC shall communicate its safety culture policy (including safety conscious work environment) to NMC employees, providing employees with the opportunity to ask questions in a live forum;

3) NMC shall train its employees holding supervisory positions and higher who have not had formal training on Safety Conscious Work Environment Principles within the previous two years of the issuance of the Confirmatory Order; and

4) NMC agrees to develop action plans to address significant issues identified as needing management attention in the NMC 2004 and 2006 Comprehensive Cultural Assessments at Point Beach; to conduct focus group interviews with Priority 1 & 2 organizations to understand the cause of the survey results; and to review and, as appropriate, reflect nuclear industry best practices in its conduct of focus groups and action plans to address the issues at Point Beach.

The effectiveness of the licensee's action plans, Confirmatory Order Item 4, will be evaluated at a later date.

As part of the review, the inspectors interviewed site personnel, observed training conducted in response to the Confirmatory Order, observed meetings held by the licensee in response to the Confirmatory Order, and reviewed some of the applicable corrective actions the licensee had taken in response to the Confirmatory Order. The review by the inspectors included an Office of Enforcement Specialist, and constituted one inspection selected issue followup sample.

b. Assessments and Observations

Willingness to Raise Safety Issues and the Corrective Action Program

Based on the interviews with employees, the inspectors noted similar results to those obtained and documented in the Fall 2006 NRC Problem Identification and Resolution Report, NRC Inspection Report 05000266/05000301-2006015. All the individuals interviewed indicated they did not have any hesitancy in raising nuclear safety issues.

Typically, they raised issues and concerns through their supervisors which, in some cases, involved the supervisor entering the issue into the corrective action program. None of the individuals interviewed expressed any negative experiences for bringing issues to their supervisors. They were also cognizant of the other avenues available to them for raising concerns and of their right to raise a concern to the NRC.

Regarding the corrective action program, many of the interviewees stated that the April 2006 change in the software system had increased the difficulty of and time required for entering issues into the corrective action program, even though everyone had received training. In addition, although all individuals who were interviewed expressed a willingness to raise concerns using the corrective action program, many of the interviewees did not hold a high degree of confidence that timely corrective actions would be implemented. The inspectors noted that this may be attributable to the fact that approximately 50-percent of condition reports written are either “closed to actions taken” or “closed to trend with no action taken,” without any feedback from the screen team to the initiator of the concern. While feedback to the initiator of a condition report is a management expectation, the inspectors noted from the interviews that this was not consistently accomplished.

<u>Observations on Actions taken in Response to the Order</u>

The inspectors reviewed the first three action items of the Order and noted the following observations:

- In response to Confirmatory Order Item 1, the licensee revised Procedure FP-PA-ARP-01, “CAP Action Request Process,” in the form of Attachment 13, “Expectations for Use of the Corrective Action Program,” and communicated its policy, and changes therein, to managers and employees. The inspectors observed several sessions of CAP training for employees and CAP training for managers. During the training, the inspectors observed that managers were guided to encourage employees to initiate condition reports, and employees were encouraged to initiate condition reports. However, the inspectors noted that the written guidance in Attachment 13 was, in some cases, ambiguous, and could be perceived by some employees and managers as a restriction to the free flow of information, potentially inhibiting an employee’s use of the corrective action program. For example, some supervisors and managers who were interviewed following the training, did not have a consistent understanding of whether an employee could be subject to discipline for non-compliance with any of the articulated expectations. The inspectors noted that those expectations in Attachment 13, which had the potential to restrict the content of a condition report, could be perceived as contrary to the recognized benefits of having a CAP with a low threshold for initiation.

- In response to Confirmatory Order Item 2, the licensee conducted meetings with site employees, as required. The inspectors noted that some of the employee meetings had a large number of plant employees, and that the large groups may not have been conducive to employees asking questions in a live forum, as evidenced by some of the employee sessions with a low number of questions asked. In addition, the inspectors noted that the licensee's presentation placed a disproportionate emphasis on the employees' responsibility to create a Safety Conscious Work Environment with marginal reference to licensee management's primary role in creating and fostering a Safety Conscious Work Environment.

- In response to Confirmatory Order Item 3, the licensee trained employees holding supervisory positions and higher who have not had formal training on Safety Conscious Work Environment principles. The inspectors attended a supervisory training session and had no observations. However, the inspectors noted, that this training, conducted after the employee sessions for Item 2, did clearly highlight that the responsibility for maintaining a Safety Conscious Work Environment was a primary responsibility of licensee management.

The licensee initiated condition report (CAP 01096862), to evaluate and address the inspectors' observations for Confirmatory Order Items 1 and 2. At the end of the inspection period, the licensee continued to evaluate and develop corrective actions to address those observations made concerning the Confirmatory Order.

4OA3 Event Followup

.1 Event Notification 43407, June 5, 2007, Unit 1 Reactor Trip

a. Inspection Scope

Inadequate Maintenance Procedures Result in Manual Reactor Trip Due to Feedwater Valve Failure

On June 5, 2007, upon observing that the Unit 1 main feedwater regulating valve (1CS-00476B) went from full open to full shut, operators entered abnormal operating procedure, AOP 2B, "Feedwater System Malfunction." An immediate inspection of the valve determined that the valve positioner arm was disconnected, with the positioner arm locknut found on the floor adjacent to the valve. Operators manually tripped the Unit 1 reactor in response to the loss of "B" train main feedwater control. The inspection scope included a review of the conditions that led to the reactor trip.

b. Findings

Introduction: A finding and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance (Green), was identified for failure to have procedures appropriate to the circumstances for maintenance on air-operated valve positioners. Specifically, there were no maintenance procedures that ensured the valve positioner arm hardware was properly secured. The finding was self-revealed when the hardware attaching the connecting link between the "B" feedwater regulating valve positioner and actuator became disconnected resulting in loss of control of the valve. The licensee performed an extent-of-condition review for similar valve positioners.

Description: On June 5, 2007, operators observed the Unit 1 “B” feedwater regulating valve cycling from full open to full shut and determined that automatic or manual control of the valve was not possible and manually tripped the reactor. The cause of the inability to control the valve was that the connecting link between the positioner and the actuator was no longer connected at the positioner end. The connecting link did not have a locknut on the positioner end, the nut backed off, the screw fell out, and valve control was lost. Westinghouse Technical Bulletin NSR-TB-92-09-R0 was issued on July 31, 1992, and discussed the use of locknuts on connecting links for pressurizer spray valves. The licensee response to the technical bulletin was to install locknuts on the pressurizer spray valves. However, the response was not adequate, in that it did not address other plant valves with similar positioner connecting linkage design, such as the feedwater regulating valves. This failure was classified by the licensee as a maintenance preventable functional failure since industry OE was not properly implemented.

The licensee performed an extent-of-condition review to determine whether other positioners might not have their connecting link hardware positively locked such that the screws could back out over time as a result of positioner motion. Seventeen valves were identified in each unit which had the Bailey positioner linkage configurations. These included valves such as residual heat removal (RHR) to letdown isolation valves, feedwater regulating bypass valves, RHR heat exchanger outlet control valves, non-regenerative heat exchanger temperature control valves, letdown pressure control valves, and pressurizer spray valves, among others. The inspectors challenged the licensee as to why only Bailey positioners were reviewed, as other manufacturers' valves have similar positioning linkage configurations. The licensee subsequently identified five additional valves that needed to be addressed.

Analysis: The inspectors determined that failure to have adequate maintenance procedures for positively capturing connecting hardware for air-operated valve positioner linkages was a performance deficiency and a finding. The inspectors concluded the finding is greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” issued on November 2, 2006, because the finding was associated with the equipment performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

The inspectors evaluated the finding using IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.” The transient initiator contributor was a reactor trip that did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Consequently, the finding is considered to be of very low safety significance (Green).

The inspectors also determined that the primary cause of this finding is related to the cross-cutting area of human performance. Specifically, under the component of resources, the licensee failed to ensure complete, accurate and up-to-date, procedures and work packages for work on air-operated valve positioners (H.2(c)).

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions procedure or drawings. Contrary to this, multiple licensee procedures and instructions for work on air-operated valves that affects the positioner linkage do not have adequate instructions to ensure the positioner hardware connections have adequate locking (nylock nuts, locktite, or double nuts) to ensure that the linkage cannot become disconnected due to its inservice motion. Because of the very low safety significance of this finding and because the issue was entered into the licensee’s corrective action program (CAP 01095358, CAP 01096094, and CAP 01095598), the violation is being treated as a NCV, consistent with Section VI.A.1 of NRC Enforcement Policy (NCV 05000266/2007003-06; 05000301/2007003-06).

The licensee entered the event into their corrective action program and took immediate corrective actions to correct the most safety and risk significant valve positioners. Some valves identified for review as part of the extent-of-condition did not need repair. For others, repairs included installing new nylock nuts, double nuts or other positive means of ensuring that the attachment screws would not back out. The licensee is performing a root cause evaluation.

.2 Event Notification Number 43424, Unit 1 TS Required Shutdown

a. Inspection Scope

The inspectors observed portions of the Unit 1 TS required shutdown which began at approximately 18:20 on June 14, 2007. Following a quarterly test run of the Unit 1 Turbine Driven Auxiliary Feedwater Pump 1P-29 on June 12, 2007, in which the Terry Turbine outboard bearing temperature rose to 249.5° Fahrenheit (F), plant operators declared Pump 1P-29 inoperable. The licensee attempted to troubleshoot and repair the cause of the high outboard bearing temperature and were unable to satisfactorily resolve the problem; therefore, a Unit 1 shutdown commenced, in accordance with TS.

Based on the probabilistic risk and deterministic criteria specified in Management Directive 8.3, "NRC Incident Investigation Program," and Inspection Procedure 71153, "Event Followup," and due to the equipment performance problems which occurred, a Special Inspection was initiated in accordance with Inspection Procedure 93812, "Special Inspection." Therefore, the results of this Special Inspection will be documented in NRC Inspection Report Number 05000266/2007008.

b. Findings

No findings of significance were identified.

.3 <u>(Closed) Licensee Event Report (LER) 05000266/2005001-00; 05000301/2005001-00:</u> Unanalyzed Condition Due to Inadvertent Omission of Safe Shutdown Equipment From Fire Organizational Plan.

On August 5, 2005, the licensee notified the NRC via LER 05000266/301/2005001-00 that Revisions 6 through 9 of the Point Beach Fire Organizational Plan (FOP) 1.2 omitted some safe shutdown equipment. This issue is discussed in detail in Section 4AO7.1 of this report.

.4 <u>(Closed) LER 05000266/2005002-00; 05000301/2005002-00:</u> Unanalyzed Condition Due to Deficiency in Appendix R Safe Shutdown Strategy for Charging Pump Capability.

On August 8, 2005, the licensee notified the NRC via LER 05000266/301/2005002-00 that control cables for charging pumps in Units 1 and 2 would be damaged and prevent remote operation of the charging pumps from the control room should a fire damage the cables. This issue is discussed in detail in Section 4AO7.2 of this report.

4OA5 <u>Other Activities</u>

.1 <u>Temporary Instruction (TI) 2515/166 - Pressurized Water Reactor Containment Sump Blockage (Partial Completion)</u>

a. <u>Inspection Scope</u>

The scope of this TI included verifying the implementation of the plant modifications and procedure changes required to support the modification. The inspectors reviewed the installation of the strainers as specified in plant modification MR 05-017, "Install New ECCS Sump (Sump B) Screen - Unit 1," and installation work plan IWP 05-017, "Install New ECCS Sump (Sump B) Screen - Unit 1." Additionally, as allowed by the Temporary Instruction, the changes to Unit 1 procedures (bulleted below) were sampled as a comprehensive review of procedures, was previously accomplished for Unit 2. The TI was not completed, as numerous commitments remain outstanding pending the results of continued analysis of coatings and debris both by the industry and the NRC. For tracking purposes, the procedures that remain to be reviewed are NP-7.2.28, Containment Debris Control Program, Revision 2 and NP-8.4.15, Protective Coating Program, Revision 5. These procedure changes should be reviewed after the NRC completes its testing and analysis to support resolution of chemical issues as part of closure of Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance." Also the following quarantined procedure should be reviewed when it is revised: IT-536, "Leakage Reduction and Preventive Maintenance Program Test of Containment Sump B Suction Line Mode 5, 6, or Defueled Unit 2," Revision 22.

- RP-1A, 'Preparation for Refueling,' Revision 74,
- RP-1B, "Recovery from Refueling," Revision 63,
- EOP-1.3, "Transfer to Containment Sump Recirculation - Low Head," Revision 38,
- BG-EOP-1.3, "Transfer to Containment Sump Recirculation - Low Head,' Revision 31,

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>(Closed) Unresolved Item (URI) 05000266/2006013-02; 05000301/2006013-02:</u> Basis for selection of the Ultrasonic transducer angles and scews for the SG A Primary Inlet and Outlet Nozzle Inner Radius exams.

During the Point Beach Nuclear Plant Unit 2, Fall 2006 refueling outage, the inspectors questioned the adequacy of the ultrasonic examination techniques utilized on the SG inlet and outlet nozzle inside radius sections. Specifically, the unresolved issue was generated based on the inability of the licensee to provide a basis that the Code required volume was inspected and that the transducer angles and scews produced orientation angles that would provide reliable detection of flaws.

In response to this issue, the licensee performed "forward" modeling using the Electric Power Research Institute's three-dimensional nozzle modeling toolkit, Version 1, Revision 1. This model was able to simulate the sound propagation and reflection of sound from flaws based on a nozzle geometry input from licensee nozzle dimensions. Based on the three-dimensional model developed for the specific Point Beach nozzles, the ultrasonic examination coverage was verified to include 100 percent of the required Code volume and mis-orientation angles were small enough to facilitate reliable detection of flaws.

This unresolved item is closed.

.3 <u>(Closed) Unresolved Item (URI) 05000266/2006006-10; 05000301/2006006-10:</u> EDG Endurance Test not Being Performed

During the Point Beach Component Design Basis Inspection, the inspectors determined that the licensee's TS did not contain requirements for performing an endurance run on the EDGs. An endurance run tests the ability of the EDG to remain operationally intact for a potentially long period of time. Its primary purpose is to demonstrate that each EDG is in operational readiness to assume the design basis accident loads. A standard time period for such an endurance test would be 24 hours. The licensee's longest EDG TS surveillance (Surveillance Requirements (SR) 3.8.1.3) was a one-hour test that did not bound predicted accident condition loads.

The inspectors were concerned that without an endurance run requirement the present TS surveillances did not adequately test the EDGs to ensure that they could perform their design basis accident function. The endurance run gives confidence in the readiness of the EDG to deliver its design basis loads for an extended period by challenging the EDGs mechanical systems, electrical systems, and control systems. Without this test, the inspectors were not confident in the ability of the EDG to perform its design function.

However, based on the facility's licensing basis, there were no apparent regulatory requirements in place for an EDG endurance run test. Therefore, potential enforcement actions, in regard to this issue were not identified. After the component design inspection exit meeting, the licensee voluntarily performed endurance tests for three of their four safety-related EDGs. All of the tests were performed satisfactorily per the licensee's test procedures. Since only two EDGs (one for each division) were required for accident mitigation, the completion of endurance testing for three EDGs tested functionality of an EDG for each division. Additionally, the licensee intended to continue performing this testing on a frequency consistent with their refueling outages.

This unresolved item is closed.

4OA6 Meetings

.1 Exit Meeting

On July 12, 2007, the resident inspectors presented the inspection results to Mr. D. Koehl and members of his staff, who acknowledged the findings. The licensee did not identify any information, provided to or reviewed by the inspectors, as proprietary.

.2 Interim Exit

Interim exits were conducted for:

- ISI inspections with Mr. D. Koehl on April 13, 2007. The inspectors returned proprietary information reviewed during the inspection and the licensee confirmed that none of the potential report input discussed was considered proprietary.

- Radiation Protection ALARA and radiological access control inspection with Mr. D. Koehl and other licensee staff on April 13, 2007.

4OA7 Licensee-Identified Violations

The following violations of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as a Non-Cited Violation (NCV).

.1 10 CFR Part 50, Appendix B, Criterion V requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. Contrary to the above as described in CAP 00854754, dated June 8, 2005, the licensee omitted safe shutdown equipment during Revisions 6 through 9 of the FOP. As a result, these omissions could have resulted in not completing operator actions that were credited in the Safe Shutdown Analysis. The licensee reported this event to the NRC on August 5, 2005, pursuant to 10 CFR Part 50.73(a)(2)(ii)(B) via LER 05000266/2005001-00; 05000301/2005001-00.

The violation was more than minor because the failure to include all necessary safe shutdown steps in the FOP could have affected the mitigating systems cornerstone. Specifically, the licensee's failure to include all necessary safe shutdown steps in the FOP may have resulted in the omission of safety significant operator actions had a fire occurred. This issue is circuit-related and the licensee is in transition to 10 CFR 50.48(c), National Fire Protection Association (NFPA) 805, therefore the licensee completed a quantitative risk assessment evaluation for this issue using the methodology contained in IMC 0609 Appendix F. The licensee's evaluation concluded that the risk associated with this issue was not of high safety significance. The inspectors reviewed the evaluation and concluded it was appropriate.

The inspectors evaluated the violation in accordance with the four criteria established by Section A of the NRC's Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR Part 50.48) for a licensee in NFPA 805 transition. This was performed because the licensee-identified violation was a circuit-related finding that was not associated with a finding of high safety significance. The inspectors determined that for this violation: (1) the licensee would have identified the violation during the scheduled transition to 10 CFR Part 50.48(c); (2) the licensee had established adequate compensatory measures within a reasonable time frame following identification and would correct the violation as a result of completing the NFPA 805 transition; (3) the violation was not likely to have been previously identified by routine licensee efforts; and (4) the violation was not willful. As a result, the inspectors concluded that the violation met all four criteria established by Section A and the NRC is exercising enforcement discretion to not cite this violation in accordance with the NRC's Enforcement Policy.

.2 10 CFR Part 50, Appendix R, Section III.G.2 requires that one redundant train of systems necessary to achieve and maintain hot shutdown conditions be free of fire damage. Contrary to the above as described in CAP 00830359 dated April 8, 2005, the licensee failed to protect the Unit 1 Charging Pump 2C, 1P-2C, control cables located in Fire Area A06 from damage. Additionally, the licensee failed to protect the Unit 2 Charging Pumps 2P-2A and 2P-2B control cables located in Fire Area A15 from damage. Those control cables would be damaged and prevent remote operation of the Unit 2 Charging Pumps, 2P-2A and 2P-2B, from the control room following postulated fire damage to those cables. The licensee reported this event to the NRC on August 8, 2005, pursuant to 10 CFR Part 50.73(a)(2)(ii)(B) via LER 050002662005002-00; 05000301/2005002-00).

The violation was more than minor because this failure could have affected the mitigating systems cornerstone objective. Specifically, the licensee's failure to physically protect the control cables for charging pumps 1P-2C, 2P-2A, and 2P-2B in the event of a fire in the 10 CFR Part 50, Appendix R, Section III.G.2 fire areas, left the charging pumps 1P-2C, 2P-2A, and 2P-2B control cables vulnerable to fire damage. This issue was circuit-related and the licensee was in transition to NFPA 805. Therefore the licensee completed a quantitative risk assessment evaluation for this issue using the methodology contained in IMC 0609 Appendix F. The licensee's evaluation concluded that the risk associated with this issue was not of high safety significance based on the number of the ignition sources and the effectiveness of the automatic/manual suppression in the area. The inspectors reviewed the evaluation and concluded it was appropriate.

The inspectors evaluated the violation in accordance with the four criteria established by Section A of the NRC's Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR Part 50.48) for a licensee in NFPA 805 transition. This was performed because the licensee identified violation was a circuit-related finding that was not associated with a finding of high safety significance. The inspectors determined that for this violation: (1) the licensee would have identified the violation during the scheduled transition to 10 CFR Part 50.48©; (2) the licensee had established adequate compensatory measures within a reasonable time frame following identification and would correct the violation as a result of completing the NFPA 805 transition; (3) the violation was not likely to have been previously identified by routine licensee efforts; and (4) the violation was not willful. As a result, the inspectors concluded that the violation met all four criteria established by Section A and the NRC is exercising enforcement discretion to not cite this violation in accordance with the NRC's Enforcement Policy.

.3 Technical Specification 5.7.1 and 5.7.2 governing access into high radiation areas with dose rates not exceeding 1.0 rem/hour and those exceeding 1.0 rem/hour, respectively, require, in part, that: (1) access into these areas be controlled by means of a RWP or equivalent that includes appropriate radiation protection measures; and (2) entry personnel are knowledgeable of the dose rates in the area.

Contrary to these requirements, there were two occasions when high and locked high radiation area access control requirements were not met:

- On March 28, 2007, a worker entered a posted high radiation area (dose rates in accessible areas did not exceed 1.0 rem/hour) without being assigned onto the proper RWP and without knowledge of the dose rates in the area.

- On April 7, 2007, a worker entered a posted locked high radiation area (dose rates in accessible areas exceeded 1.0 rem/hour) without being assigned onto the proper RWP and without satisfying the requirements of that RWP. The proper RWP required continuous radiation protection coverage plus the establishment of stay times based on work area dose rates neither of which were met while the worker remained in the area.

These incidents are documented in the licensee's corrective action program as CAP 01084526 and CAP 01086559. These high radiation area access control problems each represent a finding of very low safety significance because they did not involve ALARA Planning, no overexposure occurred, and a substantial potential for an overexposure did not exist given the radiological conditions and the use of electronic dosimetry. Also, the licensee's ability to assess worker dose was not compromised for these incidents.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
C. Butcher, Site Engineering Director
R. Harrsch, Operations Manager
D. Koehl, Site Vice-President
J. McCarthy, Director of Site Operations
M. Miller, Plant and System Engineering Manager
G. Packard, Plant Manager
L. Peterson, Design Engineer Manager
M. Ray, Regulatory Affairs Manager
D. Schuelke, Radiation Protection and Chemistry Manager
L. Schofield, Employee Concerns Program Manager
J. Schweitzer, Manager of Projects
G. Sherwood, Engineering Programs Manager
C. Sizemore, Training Manager
B. Vandervelde, Maintenance Manager
S. Tulley, Emergency Preparedness Manager

Nuclear Regulatory Commission
P. Milano, Point Beach Project Manager, NRR
J. Cameron, Chief, Reactor Projects, Branch 5, Region III

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000266/2007003-01; 05000301/2007003-01	NCV	Failure to Implement Work Instructions for Preventive Maintenance on Safety-Related Battery Chargers (Section 1R12.1)
05000266/2007003-02	NCV	Failure to Appropriately Manage an Orange Risk Condition (Section 1R13.1)
05000266/2007003-04; 05000301/2007003-04	NCV	Inadequate Program for Preventive Maintenance of Breaker Mechanism Operated Control Switches (Section 1R19.1)
05000266/2007003-05; 05000301/2007003-05	NCV	Failure to Establish Appropriate Test Conditions for Leak-Rate Testing Outside Containment (Section 1R22.1)
05000266/2007003-06; 05000301/2007003-06	NCV	Failure to Perform Appropriate Maintenance on Air-Operated Valve Positioner Linkage (Section 4OA3.1)

Closed

05000266/2006013-02; 05000301/2006013-02	URI	Basis for selection of the Ultrasonic transducer angles and scews for the SG A Primary Inlet and Outlet Nozzle Inner Radius exams (Section 4OA5.2)
05000266/2005001-00; 05000301/2005001-00	LER	Unanalyzed Condition Due to Inadvertent Omission of Safe Shutdown Equipment From Fire Organizational Plan (Section 4OA3.3)
05000266/2005002-00; 05000301/2005002-00	LER	Unanalyzed Condition Due to Deficiency in Appendix R Safe Shutdown Strategy for Charging Pump Capability (Section 4OA3.4)
05000266/2006006-10; 05000301/2006006-10	URI	EDG Endurance Test not Being Performed (Section 4OA5.3)

Opened

05000266/2007003-03; 05000301/2007003-03	URI	Failure to Submit Reactor Coolant System Pressure and Temperature Limits Report (Section 1R15.1)

LIST OF DOCUMENTS REVIEWED

1R01: Adverse Weather Protection

CAP 01094188; 2006 PC 49, Cold Weather Preps not Completed as Scheduled
CAP 01089220; Potential Tornado Missile Hazards
CAP 01085209; PC 99 Tornado Hazards Inspection CK - Potential Inadequacies
CAP 01093671; Some Warm Weather Issues Missing from Site Summer Readiness
PC 99; Tornado Hazards Inspection Checklist, Revision 0

1R04: Equipment Alignment

CAP 01097166; 2P-11B CCW Pump leaking 1.1 gal/day Out of Inboard Seal
CAP 01098674; 2P-11A, CCW Pump Excessive Oil Leakage
CL 11A G-02; G-02 Diesel Generator Checklist; Revision 26
CL 5C; Spent Fuel Pool Cooling and Refueling Water Circulating Pump Normal Operation Valve Lineup; Revision 12

1R05: Fire Protection

Fire Hazards Analysis Report; January 2007
NP1.9.9; Transient Combustible Control; Revision 12
CAP 01080091; Sprinklers installed too far from ceiling
CAP 01080098; Sprinklers not positioned per requirements
FPEE 1999-003; Diesel Generators G03 & G04 Building Boundaries

1R08: Inservice Inspection Activities

Engineering Evaluation EE 2007-0002; Revision 0; Unit 1 Reactor Vessel Head Effective Degradation Year (EDY) Determination; dated April 12, 2007
Ultrasonic Data Sheet w.o.324583; “B” Steam Generator Hot Leg Manway; dated April10, 2007
Ultrasonic Data Sheet 2007UT-019; SIS -06-CS-1004-09; Pipe to Elbow; dated April 10, 2007
Ultrasonic Data Sheet 2007UT-021; SIS -06-CS-1004-12; Elbow to Pipe; dated April 10, 2007
Visual, VT-3; Examination Data Sheet 2007VT-034; SI-301R-1-H8; dated April 10, 2007
Dye Penetrant Data Sheet 2007PT-010; SIS -06-CS-1004-09; Pipe to Elbow Weld; dated April 10, 2007
Dye Penetrant Data Sheet 2007PT-011, SIS -06-CS-1004-12; Elbow to Pipe Weld; dated April 10, 2007
NMC Memo NPM 2007-0132; from W.A. Jensen to L.E. Hawki; Documentation of Methodology for Determining the Adequacy fo Point Beach Nuclear Plant; Unit 2 Steam Generator Inlet/Outlet Nozzle Inner Radius Examinations; dated April 11, 2007

Personnel Certifications

T. Blechinger, LMT, PT/UT/VT-3; dated March 8, 2007
A. Stevermer, LMT, PT/UT/VT-3; dated March 8, 2007

Documents Related to Code Pressure Boundary Welding

Work Order 0216335; SI-00845A valve replacement; 2 inch valve from P-15A Safety Injection (SI) Pump to RC Loop A Cold Leg SI check valve; dated March 22, 2004
Repair/Replacement Form; 2004-008
PT Data sheet 450828 for Welds SW-1; SW-2; dated April 4, 2004
PT Data sheet 450832 for Welds FW-1; dated April 8, 2004

PT Data sheet 450831 for Welds FW-1; dated April 8, 2004
Work Order 9937801; RH-00715C valve replacement; 2 inch valve Residual Heat Removal (RHR) to Letdown cross-connect; dated September 25, 2005
Repair/Replacement Form, 2005-0039; dated August 1, 2005
PT Data sheet 451496 for Weld FW-3; dated October 10, 2005
PT Data sheet 451495 for Weld FW-2; dated October 10, 2005
PT Data sheet 451549 for Weld SW-1; dated September 27, 2005
Welding Procedure WPM 2.P8-GT, Revision 1; Welding Procedure for Austenitic Stainless Steels ASME Group P-8 GTAW -Pipe Diameter over 1"; dated April 23, 2004
Weld Procedure Specification (WPS) WP-7, Revision 2, Austenitic Stainless Steels ASME Group P-8 GTAW Pipe Diameters over 1."
Welder Qualification J. J. Blazer; dated September 26, 2001
Welder Qualification J. Fessler; dated February 10, 2000
Welder Qualification D.S. Tomman; dated September 19, 2005
Welder Qualification D.O. Pederson; dated September 19, 2005

Documents Associated with Boric Acid Corrosion Program
BALCM Program; Revision 3; Boric Acid Leakage and Corrosion Monitoring Program; dated June 12, 2006
Procedure NP 7.4.14, Revision 3, Boric Acid Leakage and Corrosion Monitoring; dated June 14, 2006
BALCM Program Appendix A; Revision 5; Reactor Coolant System Leak Test Boundary Document; dated June 12, 2006
BALCM Program Appendix B; Revision 2; Boric Acid Examination Guidelines; dated June 12, 2006
BALCM Program Appendix C; Revision 3; Boric Acid Indication Evaluation; dated June 12, 2006
Boric Acid Indication and Evaluation 07-0120; 1SI-V-24 Containment Spray Test Line Common Vent; dated April 10, 2007
Boric Acid Indication and Evaluation 07-0119; 2SI-829C Safety Injection Test Line Flow Control; dated April 10, 2007
Boric Acid Indication and Evaluation 07-0056; 1SI-853A Low Head SI Core Deluge Check Valve; dated April 01, 2007
Boric Acid Indication and Evaluation 07-0053; 1RH-720 RHR return to RCS; dated April 6, 2007
Boric Acid Indication and Evaluation 07-0103; 1CV-371B Letdown Line Containment Manual Isolation; dated April 1, 2007
Boric Acid Indication and Evaluation 07-0070; 1SI-841B T-34B SI Accumulator Outlet; dated April 6, 2007
Boric Acid Indication and Evaluation 07-0072; 1SI-878B P-15A SI Pump Loop B Injection; dated April 6, 2007
U1R30 Pressure Test As-Found Indication Disposition Summary; dated April 2, 2007

Documents Associated with Non-Destructive Testing Procedures
NDE-173 Revision 9; PDI Generic Procedure for the Ultrasonic Examination of Austenitic Piping Welds; dated May 26, 2006
NDE-175 Revision 3; PDI Generic Procedure for the Manual Ultrasonic Through Wall and Length Sizing of Ultrasonic Indications in Reactor Pressure Vessel Welds; dated May 26, 2006
NDE-451 Revision 23; Visible Dye Penetrant Examination Temperature Applications 45°F to 125°F; dated May 26, 2006

NDE-754 Revision 15; Visual Examination (VT-3) of Nuclear Plant Components; dated January 12, 2007

Documents Associated with Disposition of Relevant Indications

CAP 01086603; Discoloration inside the SG B Hot Leg Primary Channel Head; dated April 8, 2007

Indication Disposition Report 2005-026; AC-601R-2-H2-IWA -Linear Indication on Welded Attachment; dated June 13, 2005

Indication Disposition Report 2005-027; 1SI-878B-3 Missing U-Bolt on Support; dated September 28, 2005

Indication Disposition Report 2005-031; RC-04-PR-1001-05 PT Indication after Weld Preparation on Elbow to Elbow Weld; Dated October 22, 2005

Indication Disposition Report 2005-032; RC-04-PS-1001-04 PT Indication after Weld Preparation on Elbow to Elbow Weld; dated October 26, 2005

Indication Disposition Report 2005-033, RC-04-PS-1001-03 PT Indication after Weld Preparation on Pipe to Elbow Weld; dated October 22, 2005

CAP 00067738; Indications Found on Pressurizer Spray Line Exceed Acceptance Criteria; dated October 6, 2005

Indication Disposition Report 2005-034, RC-03-PSF-1002-03 PT Indication after Weld Preparation on Tee to Pipe Weld; dated October 22, 2005

Indication Disposition Report 2005-035, SG-B-4 Transition Cone to shell weld Unacceptable UT indications; dated October 28, 2005

Vendor Calculation PBCH-14Q-302, Steam Generator B Flaw Evaluation, dated October 28, 2005

Indication Disposition Report 2005-036; PZR-C weld-1 Upper Head to Shell Weld, Unacceptable UT indications; dated October 24, 2005

CAP 00068177; Snubber HS-2 Cold Piston Setting less than Procedure 1-PT-SS-1 Tolerance; dated October 21, 2005

Indication Disposition Report 2005-035; SG-A-4 Transition Cone to Shell Weld, Unacceptable Ut Indications; dated October 28, 2005

Indication Disposition Report 2006-004; 1FE-00601- 1HX-3A/B Non-Regen HX Shell Side Outlet Flow Element, Through Wall Leak; dated August 8, 2006

Documents Associated with Steam Generator Examinations

Procedure NP 7.7.16; Steam Generator Program, Revision 8; dated March 28. 2007

Procedure NP 7.7.17; Requirements for Steam Generator Primary Side Activities; Revision 6; dated March 28, 2007

Procedure NP 7.7.18; Requirements for Steam Generator Secondary Side Activities; Revision 4; dated March 21, 2007

Procedure SEM 7.11.20; Revision 0; Eddy Current Testing of the Unit 1 Steam Generators; Revision 0; dated August 5, 2006

Westinghouse Steam Generator Condition Monitoring Assessment of Spring 2004 Inspection Results and Operational Assessment of Operating Cycle 29 and 30, Point Beach Unit 1; Revision 0; dated July 22, 2004

Westinghouse Steam Generator Degradation Assessment for Point Beach Unit 1 SG, Revision 0; dated March 2007

Corrective Action Documents As A Result of NRC Inspection
CAP 01086031; Incorrectly Recorded Parameter on Welder Qualification Records; dated April 4, 2007
CAP 01086750; Did not meet expectations for timely response to NRC questions; dated April 9, 2007
CAP 01086763; NRC Observation regarding IDR 2006-004; dated April 9, 2007
CAP 01087319; Issues with Unit 1 RHR Pipe Support; dated April 11, 2007
CAP 01087482; Improvement opportunities in Boric Acid Evaluations; dated April 12, 2007
CAP 01087570; Timeliness of completing Boric Acid Evaluations associated with quarterly walkdown; dated April 13, 2007
CAP 01087502; Improvement Opportunities in OE Program; dated April 12, 2007

1R11: Licensed Operator Requalifications
NP 2.1.1; Conduct of Operations, Revision 2

1R12: Maintenance Effectiveness
CAP 01078095; Unexpected Alarm D-02/D-04 Bus under/overvoltage
CAP 01083861; Unexpected Alarm D-02/D-04 Bus under/overvoltage
CAP 01085065; Unexpected Alarm D-02/D-04 Bus under/overvoltage
CAP 01089918; D-108 Current Sensing Card A3 Failure
CAP 01071742; D-09 Battery Charger Failure while Aligned to D-01 Bus
CAP 01069350; 125Vdc System has Accumulated Two Battery Charger MPFF
CAP 01076589; D-107 Battery Charger Current Limiter Found Failed High
CAP 058939; Review of CAP 014161 for Continued Aging Effects Since Disposition
ACE 001855; Apparent Cause Evaluation for Problems with Restoration of D-09
ACE 01071742-06; D-09 Battery Charger Failure
CE 015364; Complications with D-09 Station Battery Charger Restoration
Maintenance Rule Function List for 125Vdc Electrical System; December 7, 2004
Pant Health Equip Issue Presentation; Replace D-07, D-08, D-09 Station Battery Chargers; February 28, 2007
MRE01089918-01; D-108 Current Sensing Card A3 Failure
WO 9505080; Replace Output Filter Capacitors for D108
WO 0415746; D-108 Battery Charger Maintenance
WO 0415753; Replace capacitors and printed circuit boards
WO 0262079; Replace Capacitors and Printed Circuit Boards
WO 00262107; D-09 Capacitor and Printed Circuit Board Replacement
RMP 9359-7C; DC Station Battery Charger D-09 Maintenance Procedure; Revision 3
RMP 9359-8B; DC Station Battery Charger D-108 Maintenance Procedure; Revision 0
Maintenance Rule (a)(1) 125Vdc System Action Plan, January 24, 2007
Point Beach Nuclear Plant Top Equipment Issues
Documentation Identified in Point Beach Nuclear Power Plant NRC Special Inspection Report 05000266/2007008 for 1P-29 and 2P-29 Turbine Driven Auxiliary Feedwater Pumps

1R13: Maintenance Risk Assessment and Emergent Work Evaluation
NP 10.3.6; Shutdown Safety Review and Safety Assessment; Revision 19
U1R30 Reduced Inventory Orange Path Contingency Plan; Revision 0
U1R30 Reduced Inventory Orange Path Contingency Plan; Revision 1
U1R30 Reduced Inventory Orange Path Contingency Plan; Revision 2

SEP2.3 Unit 1; Cold Shutdown Loss of Coolant Accident; Revision 13
BG SEP2.3 Unit 1; Background Document Cold Shutdown Loss of Coolant Accident; Revision 10
CAP 01086452; Orange Path Contingency Plan
CAP 01087625; Safety Monitor Issue with Diving Activities
Safety Monitor Calculation Reports for Units 1 and 2 for Applicable Work Weeks
Work Week Execution Schedules for the Applicable Work Weeks
Operator Logs for the Applicable Work Weeks

1R15: Operability Evaluations

CAP 01085701; Degraded Voltage Unanalyzed Condition for a Unit Trip
Operational Decision Making Issue Evaluation for CAP 01085701
CAP 01090876; 1-PT-430 Instrument Root Valve Configuration Discrepancies
CAP 01085701; 1X03 Degraded Voltage Concerns
CAP 01098358-01; OPR for Moisture in 2P-29 TDAFW Pump
OPR175; Evaluation of Exceeding Neutron Fluence Limits of TRM 2.2 on Unit 1 Material Locations
NP 7.7.14; Reactor Vessel Integrity Program; Revision 3
NP 7.7.14; Reactor Vessel Integrity Program; Revision 3

1R19: Post-Maintenance Testing

Work Order 00323846; Containment Crane Z-013
RMP 9118-1; Containment Building Crane Operability Inspections; Revision 6
RMP 9118-2; Containment Building Crane Inspections; Revision 3
RMP 9044-1; Auxiliary Feedwater Pump Terry Turbine Overhaul; Revision 11
RMP 9201; Control and Documentation for Troubleshooting and Repair Activities; Revision 3
IT-290B; Overspeed Test Turbine-Driven Auxiliary Feedwater Pump, Refueling Interval Unit 1 Completed on May 1 and 3, 2007
RMP 9314; 1-SI-850-B Maintenance, Static Test, and Adjustment Completed on April 28, 2007
RMP 9314; 1-SI-850-A Maintenance, Static Test, and Adjustment Completed on April 28, 2007
RMP 9370; Bus Inspection and Cleaning; Revision 5
Westinghouse 499B466 SH.265A; Elementary Wiring Diagram, 4160Vac Switchgear 1A-05 Cubicle 57 Bus Tie Breaker 1A52-57
RMP 9376-2; Limitorque MOV Static/Dp Testing for Gate and Globe Valves; Revision 12
IT-40B; Safety Injection Valves Shutdown Unit 1 Completed on April 27, 2007
IT-03D; RHR Valve Exercise Test for Operation of Shutdown; Revision 8
IT-03F; 1P-10A LHSI Pump Profile Test Mode 6 High Cavity Level Unit 1; Revision 2
IT-320; CVCS Valves (Cold Shutdown) Unit 1; Revision 17
0-SOP-G02-001; Maintenance Operation for EDG G-02; Revision 6
CAP 01091966; 4160V Breaker MOC Switch Preventive Maintenance
CAP 01087821; 13.8 KV Bus H02 Feeder Breaker H52-20 from 1X-03 Transformer does not Indicate Correctly at ATC
CAP 01088588; Issue with Methodology of Setting Up Cell Switch Linkages
CAP 01091966-01; Condition Evaluation
CAP 01087816; TS-82 Emergency Diesel Generator Routine Testing Abort
CAP 00034339; G02 EDG Governor Floating During TS-82
CAP 00050780; Circuit Breaker 2A-76 Causes ORT-3 Problems
CAP 00050758; G02 Load Control not as Expected During ORT-3A

MRE 000148; Maintenance Rule Evaluation to Address Failure of Breaker
CAP 00055394; MOC Switch Alignment
CAP 01091966; 4160Vac Breaker MOC Switch Preventive Maintenance
CAP 01088580; Instructions were not Adequate to Identify an Improperly Adjusted Cell Switch
ACE 001378; G02 Governor Floating During TS-82 - Apparent Cause for CAP 034339
WO 325458-08; Work Plan for 1A52-66 MOC Switch
WO 325458-01; Work Plan for 1A52-57 TB5-3 and TB5-4 Voltage Measurement

<u>Section 1R20: Refueling and Other Outage Activities</u>
ROD 11; Core Layout information Unit 1 Cycle 31; Revision 17
RP-1A; Preparation for Refueling; Revision 74
RP-1B; Recovery from Refueling; Revision 63
OP-4D Part 1; Draining the Reactor Coolant System; Revision 71
OP-4F; Reactor Coolant System Reduced Inventory Requirements - Unit 1; Revision 8
NP 1.2.6; Infrequently Performed Tests or Evolutions; Revision 10
NP 2.1.8; Protected Equipment; Revision 7
1RMP 9096-1; Reactor Vessel Head Removal and Installation Using Biach Tensioning System; Revision 0
U1R30 Reduced-Inventory Orange Path Contingency Plan; Revisions 1, 2, and 3; April 6, 19, 23, 2007 Respectively;
CL 2A; Defueled to Mode 6 Checklist; Revision 10; April 6, 2007;
CL 2B; Mode 6 to Mode 5 Checklist; Revision 9; September 6, 2005;
CL 2C; Mode 5 to Mode 4 Checklist; Revision 11; October 18, 2006
CL 2D; Mode 4 to Mode 3 Checklist; November 12, 2006
CL 20; Post Outage Containment Closeout Inspection; Unit 1; Revision 9; January 22, 2007;
OP 1B; Reactor Startup; Revision 54; July 17, 2006
OP 1C; Startup To Power Operation, Unit 1; Revision 12; March 15, 2007
OP 4D Part 1; Draining the Reactor Coolant System; Revision 71; February 5, 2007;
Focused Self Assessment Report Template; U1R30 Shutdown Safety Review
Daily Shutdown Risk Assessments
CAP 00901950; NRC Sites Potential Criterion XVI Violation
CAP 01067005; PI&R Inadequate Extent of Condition
CAP 01064780; PI&R Inadequate ACE for Criterion XVI Violation
CAP 00069111; Inadequate Corrective Actions for Boric Acid indications
CAP 00901950-02; Apparent Cause Evaluation of CAP 0164780
CAP 01090876; Evaluation of As-found Conditions for Instrument Tubing
CAP 01088061; Unit 1 foreign Material on Lower Core Plate
CAP 01086603; Discoloration Inside the Steam Generator B Hot Leg Primary Manway
CAP 01083177; Resolve HX-18A/B HELB and Flooding Concern
CAP 01086213; Westinghouse Fuel Calculation Issues
CAP 01087188; Wood in Containment
CAP 01087752; Discoloration on Fuel Assemblies
CAP 01088588; Methodology of Cell Switch Linkage Setup
CAP 01088785; Boric Acid Indications on B Reactor Coolant Pump
CAP 01088919; Constant Load Support Will Require Replacement
CAP 01089007; Corrosion on Flange Next to Reactor Coolant Pump 1P-1B
CAP 01088041; State of Core Exit Thermocouples
CAP 01088588; Verification of Cell Switch Contacts During Breaker Swaps

CAP 01088785; RCP Main Flange Joint Integrity
CAP 01086573; Steam Generator Blowdown Heat Exchanger Crane Issue
CAP 01086573; RWST Insulation Damaged from Steam Generator Blowdown Heat Exchanger
CAP 01088042; Containment Spray Ring Support Potentially Missing or Broken

1R22: Surveillance Testing
Inservice Inspection Basis Document; Revision 3
Inservice Testing Program Document - 4th Interval; Revision 1
IT-530C; Leakage Reduction and Preventive Maintenance Program Train A High Head Safety Injection and Residual Heat Removal Piggyback Test (Refueling); Unit 1; Revision 12
IT-760; Flow Test of High Head Safety Injection Check Valves (Refueling); Unit 1; Revision 10
Containment leak Rate Testing Program; Revision 8
IT-8A; Cold Start of Turbine-Driven Auxiliary Feed Pump and Valve Test (Quarterly); Unit 1; Revision 44
IT-07A; P-32A Service Water Pump (Quarterly); Revision 18
IT-03A RHR Pumps and Valves in DHR Mode Cold Shutdown; Unit 1; Train B
ICP 6.57; Service Water Header Flow Calibration; Revision 3
ORT-3A; Safety Injection Actuation with Loss of Engineered Safeguards AC (Train A); Revision 41
ORT-3B; Safety Injection Actuation with Loss of Engineered Safeguards AC (Train B); Revision 38
ORT-3C; Auxiliary Feedwater System and AMSAC Actuation; Unit 1; Revision 9
ORT-06; Containment Spray Sequence Test; Revision 24; Completed April 28, 2007
CAP 01090269; Large Amount of Boric Acid Around the Pump Seals
CAP 01090268; Active Boric Acid Leak at Pump Seal
CAP 01090943; As-Found Reactor Coolant System Leak Check Failed to Identify a Significant Boric Acid Leak
CAP 01088785; Boric Acid Indications on ‘B’ Rector Coolant Pump
CAP 01089007; Corrosion on Flange Next to P-1B RCP Labrynth Seal Isolation 1CV-308B
CAP 01094513; P-32A Service Water Pump Fails Inservice Testing

2OS1: Access Control to Radiologically Significant Areas
HPIP 1.57.1; Evaluation of Whole body Count Results; Revision 15
RWP 00000517; Nozzle Dams Installation/Removal - Airborne; Revision 03
RWP 00000475; High Radiation Area/Class 2G; Revision 04
RWP 00000731; Remove and Reinstall Reactor Vessel Head; Revision 00
Self-Assessment Report No. SAR 01024831; Control of High Radiation Areas, Locked High and Very High Radiation Areas at Point Beach; Assessment Dates October 23- 27, 2007
CAP 01078987; Containment Entry Made on Incorrect RWP; dated February 23, 2007
CAP 01085712; Unit 1 Cavity LHRA Gate Needs Enhancement; dated April 3, 2007

2OS2: ALARA Planning and Controls
FP-RP-JPP-01; Radiation Protection Job Planning; Revision 3
NP 4.2.1; ALARA Program; Revision 17
U1R30 Work Activity Exposure Estimates and Dose Reports for April 10 - 13, 2007
Work in Progress Review for Work Order 00278897; Steam Generator Eddy Current Testing; dated April 10, 2007
Work in Progress Review for Work Order 00278898; Steam Generator Tube Sheet

Cleaning/Sludge Lancing; dated April 11, 2007
Work in Progress Review for Work Order 00290892; Steam Generator Nozzle Dam Installation/Removal; dated April 9, 2007
Work in Progress Review for Work Order 00278895; Open/Close Steam Generator Handholes; dated April 10, 2007
Work in Progress Review for Work Order 00278882; Open/Close Steam Generator Primary Manways; dated April 9, 2007
Package for Work Orders 00278887/00278897 and Associated Radiological Assessment Forms; Steam Generator Eddy Current Testing; dated various periods in 2007
Package for Work Order 00279192 and Associated Radiological Assessment Forms including TEDE ALARA Evaluation; Remove and Reinstall Reactor Vessel Head; dated various periods in 2007
Package for Work Order 00279398 and Associated Radiological Assessment Forms; Inspect and Maintain Reactor Coolant Pump; dated various periods in 2007
Package for Work Orders 00278890/00278898 and Associated Radiological Assessment Forms; Steam Generator Tube Sheet Cleaning/Sludge Lancing; dated various periods in 2007
Package for Work Orders 00278882/00278894 and Associated Radiological Assessment Forms including TEDE ALARA Evaluation; Removal/Installation of Primary Steam Generator Manways and Diaphragms; dated various periods in 2007
Package for Work Order 00290892 and Associated Radiological Assessment Forms including TEDE ALARA Evaluation; Installation/Removal of Nozzle Dams; dated various periods in 2007
Package for Work Order 00222369 and Associated Radiological Assessment Forms; Sump “B” Modification; dated various periods in 2007
Package for Work Order 00324583 and Associated Radiological Assessment Forms including TEDE ALARA Evaluation; Ultrasonic Test of Indication in Steam Generator Hot-Leg Bowl; dated April 9 - 11, 2007
CAP 01086620; Breathing Air Connections Tampered; dated April 7, 2007
CAP 01074280; Nozzle Dam Issues Requiring Resolution and Communication; dated January 26, 2007
CAP 01086029; Poor Radworker Practices; dated April 4, 2007

4OA2: Identification and Resolution of Problems
Documentation Associated with the Response to the Confirmatory Order of January 3, 2007
CAP 01096862; Potential Ineffective Confirmatory Order Corrective Actions

4OA3: Event Followup
11CP 05.011; Feedwater Control Valve Outage Calibration; Revision 10; August 30, 2005
CAP 00830359; Problem Identified with Appendix R Safe Shutdown Strategy for Fire Area A06; dated April 8, 2005
CAP 00839145; Problem with the Use of 2P-2A and 2P-2B for Appendix R in Fire Area A15; dated April 28, 2005
CAP 00839181; Problem with Use of 2P-29 for a Fire in Fire Area A15; dated April 28, 2005
CAP 00850920; Appendix R Separation Issue Requires Further Analysis; dated May 27, 2005
CAP 00854754; Guidance for Manual Actions Due to a Fire Does Not Match Safe Shutdown Analysis; dated June 8, 2005
CAP 01095354; Equipment Malfunctions During Unit 1 Reactor Trip
CAP 01095361; Following a Manually Initiated Reactor/Turbine Trip, 345 KV Bus Section 2 (BS-2) was Automatically Locked Out (de-energized)

CAP 01095598; Positioner Linkage for the ICS-00476, 1B Feed Reg Valve, Disconnected Due to a Nut Backing Off of the Connection Bolt
CAP 01096094; Review of Non-Bailey positioner feedback linkage connections
CAP 01099675; AOV Positioner Feedback Linkage Fastener Configuration
MRE 01095454-01; ICS-00476, Feedwater Regulating Valve Failure
NP 5.3.3; Incident Investigation and Post-Trip Review; Revision 5; September 27, 2004
NSR-TB 92-09-RO; Failure of Pressurizer Spray Valve Linkage
RMP 9141; Air-Operated Valve Testing and Adjustment; Revision 6; April 4, 2007
WO 0413524; Perform ICP 5.11 on Feedwater Control Valves
WO 00278954 01; Perform ICP 5.11 on Feedwater Control Valves
WO 00279399 01; Rebuild Positioner
WO 00333145 01; Install New Positioner Linkage Nut
WO 00333161-04; Perform Lock-Out Testing of the 1-86-TG-01 and 1-86-X-01

40A5: Other
MR 05-017 (EC 1602); Install New ECCS Sump (Sump B) Screen - Unit 1
0-PT-EDG-021; G02 EDG Endurance and Margin Testing; September 2, 2006; dated September 3, 2006
0-PT-EDG-031; G-03 Emergency Diesel Generator Endurance and Margin Testing; dated January 13, 2007
Procedures as listed under Section 4OA5
SFS-PB-GA-02; Drawing - Point Beach Unit 1 Sure-Flow Strainer (B Strainer)
SFS-PB-GA-00; Drawing - Point Beach Unit 1 Sure-Flow Strainer (Recirc Sump System)
SFS-PB-GA-03; Drawing - Point Beach Unit 1 Sure-Flow Strainer (A Strainer)
WO022369; U1 Containment Sump B Screen

4OA7: Licensee Identified Violations
CAP 01084526; HRA Entered Without Being on a HRA RWP (and Associated Area Survey Data); dated March 28, 2007
CAP 01086559; Individual Enters LHRA Without Correct Authorized RWP (and Associated Area Survey Data); dated April 7, 2007
RWP 00000512; Locked High Radiation Area; Revision 04 CAP 00830359; Problem Identified with Appendix R Safe Shutdown Strategy for Fire Area A06; dated April 8, 2005
CAP 00830359; Problem Identified with Appendix R Safe Shutdown Strategy for Fire Area A06; dated April 8, 2005
CAP 00839145; Problem with the Use of 2P-2A and 2P-2B for Appendix R in Fire Area A15; dated April 28, 2005
CAP 00839181; Problem with Use of 2P-29 for a Fire in Fire Area A15; dated April 28, 2005
CAP 00850920; Appendix R Separation Issue Requires Further Analysis; dated May 27, 2005
CAP 00854754; Guidance for Manual Actions Due to a Fire Does Not Match Safe Shutdown Analysis; dated June 8, 2005

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
EDG	Emergency Diesel Generator
ET	Eddy Current
FOP	Fire Organizational Plan
FSAR	Final Safety Analysis Report
HRA	High Radiation Area
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
LHRA	Locked High Radiation Area
MOC	Mechanism Operated Control
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PI	Performance Indicator
PM	Preventive Maintenance
PTLR	Pressure and Temperature Limits Report
RC	Reactor Coolant
SDP	Significance Determination Process
SG	Steam Generator
SI	Safety Injection
TS	Technical Specification
U1R30	Unit 1 Cycle 30 Refueling Outage
U2R28	Unit 2 Cycle 28 Refueling Outage
URI	Unresolved Item
WO	Work Order