April 12, 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 05000266/2007002 AND 05000301/2007002

Dear Mr. Koehl:

On March 31, 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 April 3, 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, two 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 are 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/

Patrick L. Louden, Chief
Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 05000266/2007002; 05000301/2007002
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
Training Manager
Site Assessment Manager
Site Engineering Director
Emergency Planning Manager
J. Rogoff, Vice President, Counsel & Secretary
K. Duveneck, Town Chairman
Town of Two Creeks
Chairperson
Public Service Commission of Wisconsin
J. Kitsembel, Electric Division
Public Service Commission of Wisconsin
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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:	05000266/2007002; 05000301/2007002
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Plant, Units 1 and 2
Location:	Two Rivers, Wisconsin
Dates:	January 1, 2007, through March 31, 2007
Inspectors:	R. Krsek, Senior Resident Inspector G. Gibbs, Resident Inspector D. Schrum, Reactor Engineer D. McNeil, Senior Operations Engineer D. Betancourt, Reactor Engineer J. Robbins, Reactor Engineer K. Barclay, Reactor Engineer
Approved by:	P. Louden, Chief Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000266/2007002 and 05000301/2007002; 01/01/2007 - 03/31/2007; Point Beach Nuclear Plant, Units 1 and 2; Adverse Weather Protection; and Event Followup.

This report covers a 3-month period of inspections by resident inspectors and regional specialists. Two Green findings with associated non-cited violations 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. The inspectors identified a finding and associated non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," having very low safety significance (Green) for the failure to take prompt corrective actions to address a potential cold weather issue initially identified in October 2006 and again in January 2007. The failure to take prompt corrective actions led to the formation of ice on offsite power and plant equipment cable trays and cabling. The sheets of ice were also in close proximity to the Unit 2 Refueling Water Storage Tank level indicators and outlet piping. The licensee initiated condition reports and took immediate corrective actions and had planned additional corrective actions based on a causal evaluation.

  The finding is greater than minor because if left uncorrected the finding would become a more significant safety concern in that the formation of ice in the facade building in this case could have affected safety-related equipment. In addition, the finding is associated with the external factors 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. Because the significant ice buildup in the Unit 2 facade was an external factor and transient initiator contributor, and did not contribute to both the likelihood of both a reactor trip and that mitigation equipment or functions would not be available, the finding is considered to be of very low safety significance (Green). This finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not take appropriate corrective actions in a timely manner, commensurate with their safety significance and complexity. (Section 1R01)

Cornerstone: Barrier Integrity

- Green: A non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, "Design Control," having very low safety significance was self-revealed for the failure to maintain sufficient design margin for the expected running currents of the control room emergency filtration system fans to their thermal overload trip settings. This occurred due to design errors in a modification that replaced the fans in October 2006. Control Room Emergency Filtration System (CREFS) Fan W-1-B tripped on a breaker thermal overload during surveillance testing in February 2007 with low outside ambient air temperature (approximately negative 11°Fahrenheit). Licensee analyses also demonstrated that a trip of fan W-14A could have occurred for the combination of low ambient temperature and degraded grid voltage. The licensee took immediate corrective actions to replace the breaker thermal overloads with thermal overloads of a higher setting as a result of troubleshooting and evaluations performed following the trip of the W-14B fan. The issue was entered into the licensee's corrective action program and a root cause evaluation was subsequently performed.

  The finding is greater than minor because it is associated with the attribute of maintaining radiological barrier functionality of the control room and affected the Barrier Integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Loss of CREFS fans during a release could result in increased dose to the operators in the control room potentially affecting control room habitability. Although the finding involved a potential failure of the CREFS to provide its filtration function, the simultaneous occurrence of low outside air temperature, degraded grid voltage, and a radiological release is of very low probability. The finding for the failure to provide the correct thermal overload trip setting is a design deficiency that has a cross-cutting aspect in the area of human performance in that resources were not effective in maintaining long-term plant safety by maintenance of design margins. (Section 4OA3)

REPORT DETAILS

Summary of Plant Status

Unit 1 was at 100 percent power throughout the inspection period with the exception of brief downpowers during routine auxiliary feedwater and secondary system valve testing and until March 31, 2007, when the unit was shutdown to commence U1R30 refueling outage.

Unit 2 was at 100 percent power throughout the inspection period with the exception of brief downpowers 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 accessible portions of risk-significant equipment and systems susceptible to cold weather freezing during the onset of cold weather. The inspectors walked down all accessible portions of the Units 1 and 2 facade buildings, which enclosed the reactor containments and certain plant equipment inside the protected area. 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 freeze problems. This observation constituted one inspection procedure system sample.

b. Findings

Introduction: The inspectors identified a finding and associated non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” having very low safety significance (Green) for the failure to take prompt corrective actions to address a potential cold weather issue initially identified in October 2006 and again in January 2007. The failure to take prompt corrective actions led to the formation of ice on offsite power and plant equipment cable trays and cabling. The sheets of ice were also in close proximity to the Unit 2 Refueling Water Storage Tank (RWST) level indicators and outlet piping.

Description: On February 6, 2007, plant staff identified that in the Unit 2 containment facade a wall of ice had formed on cable trays and portions of cabling within the trays. The containment facade is an unheated structure surrounding the containment building which contains some plant electrical equipment and other various safety-related equipment, such as the RWST. The cable trays contained the offsite power lines from a Unit 2 transformer to the associated busses, in addition to various equipment cabling. The ice had formed on either side of the cable tray, and had a maximum dimension on one side of the cable tray of approximately 15 feet high by 6 feet wide and a thickness of greater than 1 inch. Operations staff immediately requested walkdowns from system and structural engineers to determine the structural acceptability of the trays current state which were judged satisfactory. The ice wall and cable trays were approximately 5 feet away from the RWST and near the outlet piping for the RWST, the inlet for the RWST minimum flow recirculation piping, and the level transmitters for the RWST. Operations directed maintenance craft to create a tent with a temporary heat source to melt the ice and redirect the source of water.

On February 7, 2007, the inspectors walked down the Unit 2 facade and questioned whether: the melting icicles presented a threat to the equipment below; certain junction boxes covered in ice would be affected by the water as the ice was melted; and certain portions of the facade freeze system were in jeopardy of not functioning due to the melting ice. Operations and engineering determined that no adverse conditions existed and additional operator walkdowns were performed of the facade freeze system (which prevents safety-related lines from freezing during cold weather) and area. The licensee also determined the source of the water was a leaking steam relief valve in the radioactive waste system. The relief valve was located on the 46-foot elevation of the facade and the steam normally relieved into the Primary Auxiliary Building Relief Stack located adjacent to the RWST. The licensee continued with actions to melt the ice on the trays, monitor the area and redirect the water. Finally, the inspectors noted that Condition Report AR0107430, written in January 2007, identified the leaking relief valve and stated that the issue would be a problem when colder weather arrived. The inspectors also identified to the licensee that the condition report appeared to have been inadvertently closed 1 week prior with no actions taken. Finally, the inspectors learned from discussions with plant staff that this area had experienced ice formation in prior years, although no corrective action documentation was found.

On February 8, 2007, the inspectors again walked down the facade and concluded that the relief vent stack, approximately 20 inches in diameter, was most likely filled with a combination of ice and water and noted that the licensee still had not taken any action to address the source of the leak. The inspectors also questioned whether the licensee had considered the potential adverse affects of the relief stack, likely filled with ice and water to a height of approximately 40 feet, near the Unit 2 RWST. The licensee performed additional evaluations to provide reasonable assurance that the vent line would remain intact during a seismic event and initiated actions to thaw the frozen relief stack drain. The thawing of the relief vent stack drain resulted in the sudden removal of a large volume of water from the relief vent stack, along with the melting of ice within the relief stack. In addition, the licensee took immediate corrective actions to replace the leaking relief valve to isolate the source of water.

The licensee subsequently performed an apparent cause evaluation and identified that a second Condition Report, AR01054609, was written in October 2006 identifying the leaking relief valve and the fact this would become a cold weather issue. The condition report was closed to a work request at that time, and neither the condition report, nor the work request were flagged in the licensee's work order and corrective action systems as a cold weather issue. The January 2007 condition report was also not flagged as a cold weather issue and was inadvertently closed due to an inadequate communication. Therefore, multiple barriers, including the corrective action program screening team and work request screening team, failed to identify and track this cold weather issue. The licensee also determined the relief valve vent stack drain froze due to abandonment of the heat trace circuit during the facade freeze heat trace upgrade project in 1999. The licensee subsequently determined as a result of the extent of condition that an additional 40 circuits were abandoned in 1999 with no documentation justifying that action. Finally, as a result of increased attention in this area, the licensee identified an additional 5 facade cold weather issues, which were addressed.

Analysis: The inspectors determined that the licensee's failure to implement prompt corrective actions to correct equipment deficiencies prior to the onset of cold weather, which could have significantly impacted safety-related equipment, was a performance deficiency requiring a significance evaluation. 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 the formation of ice in the facade building in this case could have affected safety-related equipment. In addition, the finding is associated with the external factors 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, "Significance Determination Process," Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," dated November 22, 2005. The transient initiator contributor was the external factor of significant ice buildup in the Unit 2 facade, and did not contribute to both the likelihood of both a reactor trip and 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 a primary cause of this finding is related to the cross-cutting area of problem identification and resolution. Specifically, under the component of corrective action program, the licensee failed to ensure the aspect of taking appropriate corrective actions to address safety issues in a timely manner, which directly contributed to this performance deficiency.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as deficiencies, deviations, and nonconformances, are promptly identified and corrected. Contrary to this, the licensee was made aware of a condition adverse to quality associated with the leaking relief valve on two occasions in October 2006 and January 2007 and did not take prompt corrective actions to correct the condition prior to the onset of cold weather and potential impacts on safety-related systems. The failure to implement prompt corrective actions for these conditions adverse to quality is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (AR01075828), it is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000301/2007002-01).

As stated previously, the licensee took corrective actions to correct the condition upon discovery and performed an apparent cause evaluation. Additional actions taken to address the apparent cause that the corrective action and work request screening teams failed to properly classify work request and condition reports as cold weather issues and subsequently assign timely corrective actions included: actions to both teams to review the event and procedure expectations regarding assigning the proper attributes for any condition to condition reports and work requests; an informational bulletin was sent to plant personnel to ensure the appropriate attributes are contained on work requests; operations staff reviewed the event and reinforced expectations regarding log entries, turnover information, and documentation of condition reports; the 40 abandoned circuits were evaluated to determine which require re-energization; re-energization of the relief valve vent stack facade freeze protection circuit; determination whether the RWST vent lines require freeze protection; and determination if any configuration changes were required for sample and drain locations in the facades.

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 if 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 work orders (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:

- Service Water Trains A, B, and C;
- Service Water Trains D, E, and F; and
- Unit 1 Component Cooling Water (CCW).

b. Findings

No findings of significance were identified.

.2 Complete System Walkdowns

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the Auxiliary Feedwater (AFW) system for Units 1 and 2. This safety-related system was selected based on the risk-significance of the system in the licensee's probabilistic risk assessment. The walkdown of the AFW constituted one semiannual inspection procedure sample.

The inspection consisted of the following activities:

- Review of plant procedures (including selected abnormal and emergency procedures), drawings, and the FSAR to identify proper system alignment;
- Review of outstanding or completed temporary and permanent modifications to the system;
- Review of open corrective action program documents (CAP ARs) and WOs that could impact operability of the system; and
- Walkdown of mechanical and electrical components in the system to assess alignment, component accessibility, availability, and current condition.

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 if 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 FZ596/Fire Area A01-H; Unit 2 Containment Facade;
- Fire Zone 156/Fire Area A06; MCC 1B32 and Fire Zone 154/Fire Area A05; Charging Pump 1P2A;
- Fire Zone 166/Fire Area A15; MCC 2B32 and Fire Zone 155/Fire Area A05; Charging Pump 2P2A;
- Fire Zone 152/Fire Area A03; Charging Pump 1P2C;
- Fire Zone 153/Fire Area A04; Charging Pump 1P2B;

- Fire Zone 163/Fire Area A12; Charging Pump 2P2C;
- Fire Zone 164/Fire Area A13; Charging Pump 1P2B; and
- Fire Zone 771/Fire Area A61; Fuel Oil Pump Room.

b. Findings

No findings of significance were identified.

.2 (Open) Unresolved Item (URI 05000266/2004010-01; 05000301/2004010-01) - Operational Implementation of Alternate Shutdown Capability

The Point Beach Nuclear Plant (PBNP) Fire Protection Program (FPP) described the means by which safe shutdown could be achieved to meet the requirements of 10 CFR Part 50, Appendix R, Sections III.G.3 and III.L. The PBNP safe shutdown analysis identified the minimum number of components and plant systems necessary for achieving Appendix R safe shutdown performance goals. The FPP accomplished safe shutdown by isolating power to most plant equipment and then used an emergency diesel generator or the station blackout turbine generator to power equipment necessary for plant shutdown. Numerous manual actions were required with four operations staff working together to shutdown both Unit 1 and Unit 2.

a. Inspection Scope

The inspectors performed a review of licensee actions regarding URI 05000266/2004010-01; 05000301/2004010-01. This URI was associated with a lack of time for critical manual actions in Abnormal Operating Procedure (AOP) AOP-10A, “Safe Shutdown - Local Control,” Revision 39, and AOP 10C, “Safe Shutdown Following a Fire at PAB [Primary Auxiliary Building] 26 Foot Central,” Revision 0. The fire protection procedures were common to both units and were to be implemented as fire area specific (i.e., alternate shutdown (ASD)) procedures for the Control Room, Cable Spreading Room, and the 26-foot elevation of the PAB. The inspectors reviewed the licensee’s evaluation for these procedures. In addition, the inspectors reviewed the licensee actions regarding Fire Operating Procedure (FOP) 1.2, “Potential Fire Affected Safe Shutdown Equipment,” Revision 15, which had not previously identified time critical manual actions for plant fire scenarios. The inspectors reviewed the licensee’s evaluation for time critical manual actions and changes to this procedure.

Description: The URI issue for procedures AOP-10 and FOP 1.2 regarded the potential for not having adequate water inventory available for reactor makeup during the 72 hours required to reach cold shutdown because of Reactor Coolant Pump (RCP) leakage and the potential drain down of the RWST (i.e., through a spuriously operated valve to the containment sump) from an Appendix R Fire. As a result, the inspectors were concerned that the licensee had the potential for a non-conservative Appendix R manual action response time in their procedures for ensuring that the RWST was isolated from draining to the containment.

The licensee initiated a CAP to formalize a previously unapproved evaluation and prepare a new calculation for the AOP-10 manual actions. The licensee entered this issue into their corrective action program as CAP060624, “Potentially Non-Conservative Appendix R Response Times,” dated November 18, 2004. The licensee completed a review and performed a new calculation to resolve the AOP issue. The licensee performed Calculation WE00010-05, Revision 0, which indicated that adequate time existed to close RWST valves to the containment and maintain an adequate inventory of water in the RWST. The inspectors reviewed this calculation and verified that adequate time existed to isolate the valves to containment and ensure a safe shutdown of the plant. This portion of the URI is considered closed.

The inspectors had identified a similar issue where the licensee did not identify the time critical manual actions in FOP 1.2, “Potential Fire Affected Safe Shutdown Equipment,” Revision 6. However, unlike the AOP-10 procedures, with step by step operator actions to shut down the plant, these procedures were symptom-based, making it difficult to identify a spurious actuation and take a manual action. For example, fires in certain fire areas required time critical manual actions, to ensure that the RWST is isolated to prevent drain down of water to the containment sump. As a result of the inspectors’ concern, the licensee issued CAP 060641, “FOP 1.2 Enhancement Recommendation,” November 19, 2004, to review this issue. During the resolution of this issue, the licensee initiated compensatory actions, such as, assigning hourly fire rounds for all areas where a fire could potential cause a spurious actuation and a subsequent drain down of the RWST. The licensee’s corrective actions included identifying time critical manual actions in this procedure and then prioritizing those manual actions that needed to be performed early during fire scenarios.

However, the inspectors noted that the licensee took credit in its evaluation for a third valve (Valve 1/2SI 851 A/B) with an unknown leakage rate in the RWST piping. In addition, the licensee had not identified the location of the control circuitry, torque, and limit switch wiring for this valve nor evaluated fire scenarios for this wiring. Additionally, the licensee used only a single spurious actuation per fire to resolve this issue. Reliance upon a single spurious actuation was inconsistent with the licensee’s Safe Shutdown Analysis Report (SSAR) and Fire Protection Evaluation Report (FPER). The use of a single spurious actuation per fire area during fire scenarios is an issue that continues to be evaluated by the NRC. As a result, this portion of the URI remains open pending regulatory resolution of the fire protection circuits generic issue.

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 February 15, 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.

.2 Operating Test Results

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the comprehensive annual job performance measure operating tests, the annual simulator operating tests, and the biennial written examinations (required to be given per 10 CFR 55.59(a)(2)). The operating tests were administered by the licensee from August through October 2006. The written examinations were conducted during December 2006. The overall results were compared with the significance determination process in accordance with NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)."

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

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 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 CAPs to determine if 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:

- 125-V DC Safety-Related Battery Chargers, D-07, D-08, D-09, D-107, D-108 and D-109; and
- 125-V DC Safety-Related Batteries D-05, D-06, D-105, D-106, and D-305.

b. Findings

No findings of significance were identified.

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

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 if 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 seven inspection procedure samples:

- Planned and emergent maintenance during the week of January 21, 2007;
- Planned and emergent maintenance during the week of February 4, 2007;
- Planned and emergent maintenance during the week of February 12, 2007;
- Planned and emergent maintenance during the week of February 18,2007;
- Planned and emergent maintenance during the week of March 4, 2007;
- Planned and emergent maintenance during the week of March 11, 2007; and
- Planned and emergent maintenance during the week of March 26, 2007.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed selected operability evaluations (OPRs) associated with issues entered into the licensee's corrective action program. The inspectors reviewed design basis information, the FSAR, Technical Specification (TS) requirements, and licensee procedures to determine the technical adequacy of the operability evaluations. In addition, the inspectors determined if 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 four procedure samples:

- AR1080577, RCS Flow Uncertainty Measurements;
- OBD for AR1043614, Battery D-06 Decreasing Specific Gravity Trends;
- AR1075828, Operability Assessment for Facade Freezing; and
- AR1068922, Issues with CCW Containment Penetrations Unit 1 P-20 & Unit 2 P-19.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

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 if:

- 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:

- Emergency Diesel Generator G-03 Governor testing on January 9, 2007;
- Emergency Diesel Generator G-03 Governor testing on January 11, 2007;
- Unit 1 Containment Spray Inservice Test of Pumps and Valves on January 17,2007;
- Unit 1 Manually Operated Breaker 1D72-DY-04 and 1Y52-DY-04 replacement and testing on January 28 and February 5, 2007;
- Z-39 Air Dryer post-maintenance testing on February 8, 2007;
- 1P-10A Residual Heat Removal pump motor MCE/EMAX and return to service on March 5, 2007; and
- 1P-11A Component Cooling Water pump post-maintenance and inservice test re-baselining on March 5, 2007, following pump inner and outer bearing replacement.

b. <u>Findings</u>

No findings of significance were identified.

1R22 <u>Surveillance Testing</u> (71111.22)

a. <u>Inspection Scope</u>

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

- 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 two inservice tests and three routine tests, for a total of five quarterly inspection procedure samples:

- Inservice testing of Service Water Pumps D, E, and F on February 21, 2007;
- Unit 2 flux mapping the week of February 27, 2007;
- Unit 1 Reactor Protection and Engineered Safety Features Yellow Channel Analog surveillance testing the week of February 28, 2007;
- Unit 1 B Component Cooling Water Pump quarterly inservice test on March 6, 2007; and
- Auxiliary Feedwater Backup Nitrogen Pressure Decay Test, P-38B during the week of March 11, 2007.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors conducted in-plant observations of physical changes to the plant and reviewed the following Temporary Modifications:

- TMOD 9187, Diesel Generator Starting Air Compressor K-4B Pressure Switch PS-305A (Control Air to Lister Diesel).

The review included associated work orders, temporary modification instructions/procedures, and 10 CFR 50.59 screenings. The review of the temporary modification constituted one inspection procedure sample.

b. Findings

No findings of significance were identified

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed one emergency preparedness drill evolution during the inspection period, observing activities in the simulator and attending the critique session. The inspectors evaluated the drill performance and determined that the critique activities appropriately captured weaknesses identified by the inspectors and verified that deficiencies were entered into the corrective action program. These activities constituted one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstones: Mitigating Systems and Barrier Integrity

The inspectors reviewed the licensee's recent Performance Indicator (PI) submittal, using definitions and guidance contained in Nuclear Energy Institute NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4, to assess the accuracy of the PI data. The inspectors independently re-performed calculations where applicable. The inspectors then validated the information required for each PI definition in the guideline, to determine if the licensee reported the data accurately. The following reviewed PIs constituted six inspection procedure samples:

Unit 1

- Unplanned Scrams;
- Unplanned Scrams with Loss of Normal Heat Removal; and
- Unplanned Downpowers.

Unit 2

- Unplanned Scrams;
- Unplanned Scrams with Loss of Normal Heat Removal; and
- Unplanned Downpowers.

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 if 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 CAPs written by licensee personnel during the inspection period. The CAPs 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 Selected Issue Followup: Unit 2 Polar Crane Main Hoist Hook Contacted Unit 2 "B" Steam Generator Vent Line

a. Inspection Scope

This issue followup is to evaluate the licensee's corrective action program response to the issue that was self-revealed on November 14, 2006, when the Unit 2 polar crane main hook contacted the "B" Steam Generator vent line, as discussed in Section 4OA2 of NRC Inspection Report 05000266/2006013; 05000301/2006013.

The inspection criteria for this review include: the completeness and accuracy of identification of the problem, the extent of condition, generic implications of site crane operations, classification and resolution of the issue commensurate with its safety significance, the identification of the causes of the problem, identification of corrective actions, and verification that interim corrective and compensatory actions have been identified and implemented to mitigate the effects of the problem until permanent action can be implemented. The review by the inspectors constituted one inspection procedure sample.

b. <u>Assessments and Observations</u>

The inspectors verified the adequacy of the following aspects of Condition Report AR 1061577, associated with the Unit 2 polar crane main hook contacting the “B” Steam Generator vent line: the completeness and accuracy of identification of the problem, the extent of condition, generic implications of site crane operations, classification and resolution of the issue commensurate with its safety significance, the identification of the causes of the problem, and the identification of corrective actions. In addition, the inspectors reviewed the licensees planned long-term corrective actions and verified their adequacy.

Finally, the inspectors verified that the interim corrective actions were completed prior to the start of the Unit 1 Refueling Outage U1R30. The inspectors noted the interim corrective actions for crane moves with unloaded or bare hooks, which included a memorandum, appeared effective. However, the inspectors also noted that the long-term corrective action to revise the crane usage procedures could have been completed prior to the outage, commensurate with the potential significance of the Unit 2 incident.

4OA3 <u>Event Followup</u>

.1 <u>Event Notification 43149, Control Room Emergency Filtration System Declared Inoperable</u>

a. <u>Inspection Scope</u>

The Control Room Emergency Filtration System (CREFS) was declared inoperable on February 3, 2007, due to W-14B, “F-16 Control Room Charcoal Filter Fan,” being declared inoperable during monthly TS surveillance testing. Upon subsequent investigation of the inoperability of the W-14B fan, the W-14A fan was declared inoperable on February 6, 2007, and the appropriate 7-day TS action condition was entered. Although the subject fans were redundant, CREFS was a single train system, therefore the CREFS inoperability was reported as an event or condition that could have prevented fulfillment of a safety function and also as a degraded condition that significantly affects plant safety. The inspectors reviewed the licensee’s immediate response and actions taken for this event as required by 10 CFR Part 50.72. The review by the inspectors constituted one inspection procedure sample.

b. Findings

Introduction: A non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) was self-revealed on February 6, 2007, during troubleshooting of the CREFS following the breaker trip of Control Room Filter Fan W-14B on thermal overload (TOL) during surveillance testing February 3, 2007. The fan motors and breaker TOL trips were replaced in October 2006 via Modification 05-016, "VNCR Emergency Fan Motor Modification." The breaker thermal overload trips were not sized appropriately for the potential range of operating conditions, nor were the breaker TOL settings tested prior to installation. The purpose of the modification was to add margin to the measured makeup air flow rate for the emergency fans as the CREFS had only marginally passed a previous TS Surveillance 3.7.9.6.

Description: The modification to install the new fan motors was completed and tested on October 9, 2006. The temperature during testing was approximately 50 °Fahrenheit (F). During routine monthly testing of CREFS on December 8, 2006, the W-14A fan tripped. The temperature was approximately 23 °F when the fan tripped on thermal overload. The overload was replaced and the system retested successfully. Apparent Cause Evaluation ACE01066583-01 was performed and incorrectly concluded that the cause of the trip was due to infant mortality as the thermal overload relay was tripping at a value much less than design. The relay was replaced and a surveillance was subsequently performed with no failures occurring for Fan W-14A.

The cause of the thermal overload trip of the W-14B fan on February 3, 2007, was evaluated by the licensee. Troubleshooting determined that the fan tripped from pumping high density cold air (approximately -11 °F) resulting in a motor current that was greater than the existing thermal overload trip settings could accommodate. (The outside air temperature was within the design parameters for the system, -15 °F to 95 °F). Applying subsequent troubleshooting results, the licensee determined that under a degraded grid condition with cold outside air temperatures, the redundant fan, W-14A could also trip with the existing TOL setting for its breaker. The change in air density from 55 °F to approximately -11 °F resulted in a calculated motor current of about 9 amperes (amps). The motor actually tripped on thermal overload at 9.2 amps. Given the accuracy of the temperature measurement, current measurement and motor cleanliness changes, the calculated trip value for the low temperature condition was close to the actual trip current measured. However, the breaker thermal overload trip at 9.2 amps was well below the 10.4 amps where the licensee anticipated breaker TOL to trip. The 10.4 amps was based on the thermal overload selected and installed for the fan motor modification.

The reason for the lower trip value for the selected thermal overload was investigated by the licensee. The licensee determined that the relay was not tested to determine what the actual thermal overload trip point was prior to releasing the fan for use. Point Beach Nuclear Plant selects breaker TOL heaters using sizing criteria for single pole applications, but uses the heaters in a different configuration. This configuration was previously justified as satisfactory in an unreviewed and unapproved "white paper" that does not meet the requirements for an engineering design product. Absent a verification of the as-left trip point, PBNP has relied only on the manufacturers quality assurance program for an overload heater that operates in accordance with the published specifications.

The licensee determined that the actual breaker TOL trip setpoint was required in the range of 11.5 to 12.3 amps to address the combined effects of low temperature, degraded grid conditions, and a maximum value of motor nameplate current. An engineering change, EC 10091, was issued to adjust the trip setpoint to this range. The minimum setting allows for fan operation under the worst condition of low air temperature and degraded grid and allows for a maximum of 140 percent of motor nameplate current.

Analysis: The inspectors determined that the failure to address the combined affects of low ambient air temperature and degraded grid voltage for the motor thermal overload trip settings of the CREFS fan motors, in combination with an apparent inadequate methodology for determining breaker TOL settings and failure to test the breaker TOL settings, was a performance deficiency that resulted in loss of functionality of the single train CREFS and 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 containment barrier cornerstone of reactor safety. The inspectors evaluated the finding using IMC 0609, "Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." The finding screens as (Green) very low safety significance, as it only represents a degradation of the radiological barrier function provided for the control room. The loss of CREFS fans during a release could result in increased dose to the operator in the control room potentially affecting control room habitability. Although the finding involved a potential failure of CREFS to provide its filtration function, the simultaneous occurrence of low outside temperature, degraded grid voltage and a radiological release is of very low probability. The finding for the failure to provide the correct thermal overload trip settings is a design deficiency that has a cross-cutting aspect in the area of human performance in that resources were not effective in maintaining long-term plant safety by maintenance of design margins.

Enforcement: 10 CFR, Part 50, Appendix B, Criterion III, "Design Control," requires that the design bases are correctly translated into specifications, drawings, procedures and instructions. Contrary to these requirements, the licensee did not establish the correct thermal overload trip settings for the CREFS fan motors for MR 05-016, "VNCR Emergency Fan Motor Modification," resulting in a failure to maintain sufficient design margin from the maximum expected running current to the thermal overload trip setting. As a result, CREFS Fan W-14 B tripped on thermal overload during testing with low outside ambient air temperature (approximately -11 °F) and it has been shown by analysis that a trip of fan W-14A would also have been likely for the combination of low temperature and degraded grid voltage. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program (AR0107542), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2007002-02; 0500301/2007002-02).

The licensee issued an engineering change, ECN 1601, to revise the trip setpoint to a range of 11.5 to 12.3 amps and both fans subsequently passed post maintenance and surveillance testing, restoring system operability. The licensee performed Root Cause Analysis RCE 01075472-01, “Root Cause Analysis of Control Room Emergency Filter Fan System (CREFS) Motor Tripping,” for these issues. The licensee’s planned corrective actions included the creation and implementation of a Design Guide for engineers to select new motor overload relays consistent with the National Electric Code to ensure that the design and licensing bases and the motor protection requirements were appropriately considered for modifications. The licensee also planned corrective actions to generically ensure that the initial installation testing requirements for breaker TOL trip settings are established and verified periodically. Additionally, planned corrective actions to change maintenance procedures will provide methods for setting and testing motor overload relays, consistent with the issued Design Guide.

.2 Licensee Response to Increased Frequency of Draining of the Unit 2 “B” Steam Generator Vent Line

a. Inspection Scope

On November 14, 2006, the Unit 2 polar crane main hook contacted the “B” Steam Generator vent line, as discussed in Section 40A3 of NRC Inspection Report 05000266/2006013; 05000301/2006013. The licensee concluded the line was operable but degraded, isolated the line with the first-off valve, 2MS-212. The licensee also implemented compensatory measures, which included periodic ultrasonic testing (UT) measurements of water level in the vent line and subsequent draining of that line, as required. Valve 2MS-212 started leaking which admitted steam/condensate into the pipe. The vent line piping downstream of the valve was to be maintained less than full to address a potential Generic Letter 96-06 overpressure concern, with 2MS-212 closed, a valve which was normally open during power operations. To maintain the line less than full, more frequent UT of the piping outside containment was required, as well as more frequent draining of the piping. In February 2007, the inspectors evaluated the station’s Operational Decision-Making Issue Evaluation Document (ODMI) that addressed options to obviate the need for the frequent UT and draining of the line. The inspectors observed performance of the UT measurements, attended the licensee’s working meeting to discuss the ODMI and reviewed related documentation. The inspectors concluded the licensee’s evaluation of options and subsequent decision to reposition the first-off isolation valve, 2MS-212, to the normal open configuration (so the entire length of the line was open back to the Steam Generator) was consistent with the station approved ODMI process and appropriate for the identified issues. Finally, the inspectors observed portions of the operations evolutions associated with the re-opening of the 2MS-212 valve. The review by the inspectors constituted one inspection procedure sample.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting

On April 3, 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:

- Biennial Operator Requalification Program Inspection with Mr. R. Amundson, General Supervisor Operations Training, Point Beach Nuclear Plant on January 17, 2007; and
- The (Open) Unresolved Item (URI 05000266/2004010-01; 05000301/2004010-01) Operational Implementation of Alternate Shutdown Capability on January 17, 2007.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Amundson, General Supervisor Operations Training
C. Butcher, Site Engineering Director
G. Casadonte, Fire Protection Coordinator
G. Corell, Radiation Protection General Supervisor
F. Flentje, Licensing Supervisor
T. Gemskie, Emergency Preparedness Supervisor
R. Harrsch, Operations Manager
J. Schleif, Assistant Operations Manager
C. Jilek, Maintenance Rule Coordinator
R. Johnson, Senior Emergency Preparedness Coordinator
T. Kendall, Engineering Senior Technical Advisor
D. Koehl, Site Vice-President
G. LeClair, Radiation Protection Supervisor
J. McCarthy, Director of Site Operations
J. McNamara, Engineering Supervisor
C. Monarch, Fire Protection Engineer
R. Mrozinsky, Appendix R Engineer
P. Nicholson, Fire Protection Engineer
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
P. Wild, Design Engineering Projects Supervisor

Nuclear Regulatory Commission

P. Milano, Point Beach Project Manager, NRR
P. Louden, Chief, Reactor Projects, Branch 5

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000301/2007002-01	NCV	Failure to Take Corrective Actions for Cold Weather Issues Prior to the Onset of Cold Weather (Section 1R01)
05000266/2007002-02; 05000301/2007002-02	NCV	Failure to Maintain Design Margin for Control Room Emergency Filtration Fan Thermal Overload Trips (Section 4OA3.1)

Discussed

05000266/2004010-01; 05000301/2004010-01	URI	Operational Implementation of Alternate Shutdown Capability (Section 1R05.2)

LIST OF DOCUMENTS REVIEWED

1R01: Adverse Weather Protection

PC 49, Part 5; Cold Weather Checklist: Outside Areas and Miscellaneous; Revision 20
PC 49, Part 4; Auxiliary Building Miscellaneous and Facades; Revision 19
OI 106; Facade Freeze Protection; Revision 21
AR01075828; PAB Exhaust Stack Frozen

1R04: Equipment Alignment

CL 10B; Service Water Safeguards Lineup, Rev 60, March 2, 2006
PB01 MWSK00000368; Sheet 1; P&ID Service Water System
PB01 MWSK00001026; Sheet 1A; P&ID Service Water System
PB01 MWSK00000145; Sheet 2; P&ID Service Water System
PB02 MWSK00000258; Sheet 1; P&ID Service Water System
PB01 MWSK00000511; Sheet 2; P&ID Service Water System
CL 13E Part 1; Auxiliary Feedwater Valve Lineup Turbine-Driven Unit 1; Rev 36, July 29, 2004
CL 13E Part 2; Auxiliary Feedwater Valve Lineup Motor-Driven; Rev 42, August 22, 2005
CL 13E Part 1; Auxiliary Feedwater Valve Lineup Turbine-Driven Unit 2; Rev 20, July 29, 2004
Ar01079090; AF system P&ID Missing IA/N2 Pressure Regulators
AR01077659; 2FI-4002, Steam Driven AFP Flow Indicator Failed
AR01049806; 1P-29 AFW Pump shutdown due to low oil level in bubbler
AR01041992; 1P-29 Steam Driven AFP turbine inbd/obd elevated temperatures
AR01041992; Apparent Cause Evaluation for knowledge weakness associated with isolating steam lines and associated steam traps for Turbine Driven Auxiliary Feed Water pumps
CL CC 001; Component Cooling Unit 1, Rev II, May 1, 2006
AOP-9B Unit ; Component Cooling System Malfunction, Rev 19; February 27, 2006
PB01 MSFK00000264; Sheet 1; P&ID Auxiliary Coolant System
PB01 MCCK00000421; Sheet 2; P&ID Auxiliary Coolant System
PB01 MCCK00000141; Sheet 3; P&ID Auxiliary Coolant System

1R05: Fire Protection

Fire Hazards Analysis Report for Applicable Fire Areas Reviewed; December 2005

Triennial Fire Protection

CALC-WE0005-10; Determination of Time Available to Isolate RWST; Revision 1
FPTE-012; TE of PBNP's Manual Action Report; Revision 3
CAP060581; Potentially Incorrect Information in LER 2000-008-00; dated November 17, 2004
CAP060583; Large Number of Manual Actions; dated November 17, 2004
CAP060624; Potentially Non-Conservative Appendix R Response Times; dated November 18, 2004
CAP060641; FOP 1.2 Enhancement Recommendation; dated November 19, 2004
OPR00088; Appendix R Fires May Challenge Operator Response – Revision of Safe Shutdown Response Procedure (FOP 1.2) to Ensure Feasibility; Revision 2

OPR000116; RWST Inventory Available for RCS Makeup Following an Appendix R Fire; dated November 20, 2004
CAP012654; Contractor Spurious Opening of the 851 Safety Injection (SI) Valves and Potential Sump Flooding; dated September 22, 2000
AOP-10A; Safe SD – Local Control; Revision 38 and 39
AOP-10C; Safe SD Following Fire at PAB 26 Foot Central; Revision 0
AOP-10E; Safe SD Following Fire at PAB 46 Foot CCW HX/BAST Room; Revision 0
PBNP Safe Shutdown Analysis Report; Revision 2
Fire Affecting Safe Shutdown Components; Revision 0
DBD-T-40; Fire Protection/Appendix R Design Basis Document; June 25, 2004
FHAR; Fire Hazards Analysis Report; Revision 2
FOP 1.2; Potential Fire Affected Safe SD Components; Revision 6 and 15
FPER; Fire Protection Evaluation Report; Revision 3
LER 2000-008-00; Inadequate Procedural Guidance for Spurious Operation of Valves; Revision 0
During Appendix R Alternate Shutdown; dated October 19, 2000
Report; Duke E and S Cable Routing 2SI-00851B Containment Sump Isolation MOV; Revision 0
SSAR; Safe SD Analysis Report; Revision 2
WCAP-15603-NP; Transmittal of WOG2000 RCP Seal Leakage Model for Westinghouse PWR; Revision 0

1R11: Licensed Operator Requalifications

Point Beach Licensed Operator Requalification Program Results

1R12: Maintenance Effectiveness

Passport Work order Search; 1/18/2007; All PMs Not Completed 125v dc
Passport Work order Search; 1/18/2007; Not PM Completed for 125v dc
Action Tracking -Parent Search; ½0/07; D-107 Battery Charger from 01/01/05
Action Tracking -Parent Search; ½0/07; D-07 Battery Charger from 01/01/05
Action Tracking-Parent Search; 1/18/2007; CAP for 125V dc
Action Tracking - Assignment (Child) Search; 1/24/07; AR for 125V dc for 2 years
125VDC System Maintenance Rule Information; 1/22/07
AR00838407; D-08 Battery Charger exceeds 80 percent unavailability threshold
AR00773306; D-09 Battery Charger Low Voltage Relay Found Out of Tolerance
AR01058436; D02/D-04 125 V DC Bus Under/Over Voltage Alarm (D-09 Battery Charger)
AR00837333; D-09 125 VDC Battery Charger Exceeded MR Unavailability Criteria in 1QTR 2005
AR00838401; D-07 Unavailability Exceeds 80% of the Maintenance Rule Performance Criteria
AR00858675; Potential to Exceed Design Basis of SR Battery Chargers During RMP 93969-7/8
AR00861029; Station Battery Recovery Instructions are Missing from RMP 9359-7
AR01069350; 125VDC System MR Reliability Criteria exceeded
CEO15364; Complications with D-09 Station Battery Charger Restoration
Emails from System Engineering to Maintenance Planning circa February and March 2006 identifying electrolytic capacitors requiring replacement during Battery Charger Maintenance to address obsolescence concerns
Troubleshooting Log for Work Order 311930; 1/19/07; D-09 Battery Charger Malfunction
Point Beach Nuclear Plant (PBNP) Top Equipment Issues List

System Health Report; 125VDC System; First Quarter 2007
Maintenance Rule 125 VDC Functional List; 12/7/2004
Maintenance Rule Performance Criteria for 125VDC System; Rev. 3, October 13, 2004
Maintenance Rule Performance Assessment for 125 VDC System; through 4th Quarter 2006
PB 00262107; PM for D-09, Capacitor and Printed Circuit Board Replacement; not completed; 3/12/06
Plant Health Equipment Issue Presentation Summary Sheet; D-07, D-08, D-09 Station Battery Charger obsolete; June 6, 2006
Plant Health Equipment Issue Presentation Summary sheet; D-07, D-08, D-09 Station Battery Charger Replacement; February 8, 2007
0-=SOP-DC-001; Point Beach Nuclear Plant system Operating Procedures, 125 VDC System, Rev. 11, January 18, 2007
Battery Replacement Schedule; January 12, 2007
Action Tracking Parent Searches for Batteries D-05, D-06, D-305, D-105, and D-106; Jan. 05 - January 07
AR01080947; Trend of Battery D-06 Specific Gravity Performance and Sharepoint Folder Assessment
AR01043614; ACE for D-06 battery declining Specific Gravity (SG) trend

1R13: Maintenance Risk Assessment and Emergent Work Evaluation

Safety Monitor Calculation Reports for Units 1 and 2 for applicable work weeks
Work Week Execution Schedules for the applicable weeks
Operator Logs for the applicable work weeks

1R15: Operability Evaluations

AR01043614; Operability Determination for Station Battery D-06 declining specific gravity and associated ARs
AR01075828; Initial Operability Assessment for Facade Freezing
AR01080577; RCS Flow Uncertainty Measurements
AR01068922; Issues with CCW Containment Penetrations Unit 1 P-20 & Unit 2 P-19

1R19: Post-Maintenance Testing

PBTP -46; Emergency Diesel Generator G03 governor Testing
WO 254354; Replacement of G03 Governor
RMP 9374-2; Molded Case Circuit Breaker (MOB/Panel) Maintenance
WO262074; D72-DY-04 CAT-1 MCCB Replacement and Testing
WO262039; 1Y52-DY-04 CAT-1 MCCB Replacement and Testing
IT 05 Containment Spray Pumps and Valves (Quarterly) U1
0-SOP-IA-002; Operation of Instrument Air Dryers and Filters
RMP 9333-4; Instrument Air Dryer (Z-39) Annual Maintenance, Rev. 4, May 4 2006
RMP 9404; Motor E-Max Testing, Rev. 0, January 17, 2007
RMP 9387; AC Induction Motor MCE Testing Procedure, Rev. 4, February 21, 2006
IT 12; CCW Pumps and Valves, Rev. 33, May 18, 2006

1R22: Surveillance Testing

RESP 6.1; Core Power and Nuclear Power Range Detector Calibration
REI 6.0; Normal Flux Mapping Using the Incore Movable Detector System
2ICP 02.001YL; Reactor Protection and Engineering Safety Features Yellow
Channel Analog 92 Day Surveillance Test
IT 12; CCW Pumps and Valves, Rev. 33, May 18, 2006
IT 07D/E/F; Service Water Pump (Quarterly), Rev. 17/19/18;
Dec. 8, 2005/July 7, 2006/Dec. 8, 2005
P 38B; Auxiliary Feedwater Water Backup Nitrogen Pressure Decay Test

1R23: Temporary Plant Modification

TMOD 9187/ K-4B Pressure Switch PS-305A (Control Air to Lister Diesel)
10 CFR 50.59 Screening 2006-0190, Rev. 0, October 13, 2006
EC 9187/ K-4B Pressure Switch (Control air to Lister Diesel)
Point Beach Nuclear Plant (PBNP) TSs
Point Beach Nuclear Plant (PBNP) TSs Basis
Point Beach Nuclear Plant (PBNP) Final Safety Evaluation Report (FSAR)
PB31 MDGK00000119; Sheet 12 P&ID Emergency Diesel Air Starting System

4OA1: Performance Indicator Verification

Performance Indicator Data for items reviewed

4OA2: Identification and Resolution of Problems

RCE AR01061577; Root Cause Analysis of Unit 2 Polar Crane Main Hoist Hook Contacted
the Unit 2 “B” Steam Generator Vent Line during Crane Modification Process
IWP 04-022; Unit 1 Polar Crane Modernization

4OA3: Event Followup

MR 05-016; VNCR Emergency Fan Motor Modification, Rev. 0, April 13, 2006
AR01066583; Apparent Cause Evaluation for W-014A multiple fan trips on thermal overload
RCE01075472-01; Root Cause Analysis fo Control Room Emergency Filter Fan System Motor
Tripping
AR01075472; W-14B Charcoal Filter Fan Tripped during TS-9
ODMI for AR1061577

LIST OF ACRONYMS USED

amps	Amperes
AOP	Abnormal Operating Procedure
APP R	Appendix R
AR	Action Request
BAST	Boric Acid Storage Tank
CAP	Corrective Action Program Document
CFR	Code of Federal Regulations
CREFS	Control Room Emergency Filtration System
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EN	Event Notification
F	Fahrenheit
FHA	Fire Hazards Analysis
FPER	Fire Protection Evaluation Report
FPP	Fire Protection Program
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
IR	Inspection Report
LER	Licensee Event Report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OPR	Operability Recommendation
PAB	Primary Auxiliary Building
PBNP	Point Beach Nuclear Plant
PI	Performance Indicator
PI&R	Problem Identification and Resolution
RCP	Reactor Coolant Pump
RWST	Refueling Water Storage Tank
SD	Shutdown
SDP	Significance Determination Process
SER	Safety Evaluation Report
SSA	Safe Shutdown Analysis
SSAR	Safe Shutdown Analysis Report
TOL	Thermal Overload
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
UT	Ultrasonic Testing
WO	Work Order