April 17, 2006

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 05000266/2006002;
05000301/2006002

Dear Mr. Koehl:

On March 31, 2006, 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 4, 2006, 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, four findings of very low safety significance were identified. Three of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these three 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 addition to the routine NRC inspection and assessment activities, Point Beach performance is being evaluated quarterly as described in the Annual Assessment Letter - Point Beach Nuclear Plant, dated March 2, 2006. Consistent with Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," plants in the multiple/repetitive degraded cornerstone column of the Action Matrix are given consideration at each quarterly performance assessment review for (1) declaring plant performance to be unacceptable in accordance with the guidance in IMC 0305; (2) transferring to the IMC 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems," process; and (3) taking additional regulatory actions, as appropriate. During this inspection period, the NRC reviewed Point Beach operational performance, inspection findings, and performance indicators. Based on this review, we concluded that Point Beach is operating safely. We determined that no additional regulatory actions, beyond the already increased inspection activities and management oversight, are currently warranted.

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

Sincerely,

/RA/

Mark A. Satorius, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000266/2006002; 05000301/2006002
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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:	05000266/2006002; 05000301/2006002
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Plant, Units 1 and 2
Location:	Two Rivers, Wisconsin
Dates:	January 1, 2006, through March 31, 2006
Inspectors:	R. Krsek, Senior Resident Inspector G. Gibbs, Resident Inspector G. O'Dwyer, Reactor Engineer T. Ploski, Senior Emergency Preparedness Analyst L. Haeg, Reactor Engineer
Approved by:	P. Louden, Chief Projects Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000266/2006002, 05000301/2006002; 01/01/2006 - 03/31/2006; Point Beach Nuclear Plant, Units 1 and 2; Operability Evaluations, Event Followup and Other.

This report covers a 3-month period of inspection by resident inspectors and announced inspections by regional specialists. A Green finding and three 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 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 finding of very low safety significance was self-revealed when the failure of circulating water (CW) pump 1P-30B and subsequent reactor trip occurred on December 13, 2005. This Green finding with no associated violation was identified for the licensee's failure to provide an adequate maintenance procedure for CW pump 1P-30B. Lack of appropriate maintenance to maintain required clearances, due to inadequate procedures, resulted in excessive clearances within the pump and the lower shaft sleeve failing directly above the flange where the shaft sleeve attached to the guide vane. The failure of the shaft sleeve caused increased vibration which resulted in low stress, high cycle fatigue of the coupling bolts. When the coupling bolts sheared, a rapid loss of condenser vacuum occurred and the operators initiated a manual reactor trip in anticipation of a total loss of vacuum.

  The intermediate term corrective action was to perform a root cause evaluation for the failure mechanism and repair CW pump 1P-30B. Repair included replacement of the coupling and coupling bolts. The licensee completed the root cause evaluation and identified several actions to prevent recurrence.

  The inspectors concluded the finding is greater than minor because it is 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. (Section 4AO3.1)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for the failure to maintain the design basis and configuration control for the detection of recirculation system leakage from the containment sump isolation valve cylinders (valves SI-850A and SI-850B for Units 1 and 2). This issue was initially identified by the inspectors during walkdowns and reviews of the containment sump recirculation piping in November/December 2005; however, at that time, the issue was not recognized by the licensee as part of the design basis of the facility. During a review of a request for additional information from the Office of Nuclear Reactor Regulation regarding a November 8, 2005, 10 CFR 50.72 report, the licensee subsequently determined that, in fact, leakage detection of the containment sump isolation valve cylinders through the pipe sleeve into the auxiliary building was part of the system's design and licensing basis.

  At the end of the inspection, the licensee had not completed a causal evaluation; however, several interim actions were in place to address the operable, but non-conforming condition. The licensee had established a corrective action to determine how to resolve this non-conforming issue.

  The inspectors concluded that this finding is greater than minor because it was associated with the design control and the equipment performance attributes of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the finding is a design or qualification deficiency confirmed to not result in loss of function per NRC Generic Letter 91-18. Therefore, the inspectors determined that this finding is a licensee performance deficiency of very low risk significance (Green). (Section 1R15.1)

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for the failure to ensure the safety function of the containment sump isolation valves was maintained and tested in accordance with the design and licensing basis. This issue was initially identified by the inspectors during walkdowns and reviews of the containment sump recirculation piping in November/December 2005; however, at that time, the issue was not recognized by the licensee as part of the design and licensing basis of the facility. The licensee subsequently determined that the design and licensing basis for the closed safety function of these valves was not properly implemented in accordance with the facility's license and required codes or standards.

  The licensee performed a causal evaluation and developed several interim and long-term corrective actions. Those corrective actions included: revision of the inservice testing program documents for testing the valves; revision of the design basis document (DBD) for the residual heat removal system; reinforcement of the expectations with engineering staff on the use of DBDs and inservice testing background documents; and development of a project plan to update the inservice test background document.

The inspectors concluded that this finding is greater than minor because it was associated with the design control, equipment performance and maintenance and testing procedure quality attributes of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the finding is a design or qualification deficiency confirmed to not result in a loss of function per NRC Generic Letter 91-18. Therefore, the inspectors determined that this finding is a licensee performance deficiency of very low risk significance. (Section 1R15.2)

- Green. The inspectors identified a non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) when the licensee failed to consider the effects of elevated control room temperatures on instrument inaccuracies following a design basis loss-of-coolant accident, which could potentially affect mitigation of the event. During the Problem Identification and Resolution Inspection documented in NRC Inspection Report 2005012, the inspectors identified an unresolved item (URI) related to the effects of elevated control room temperatures on instrument accuracies and accident mitigation during a design basis loss of coolant accident. Subsequent review and root cause evaluation determined that the licensee had failed to consider the effects of elevated control room temperatures on instrument inaccuracies for a calculation associated with the reconstitution project.

  The licensee entered the issue in its corrective action system and performed a root cause analysis. Corrective actions to prevent recurrence included strengthening review requirements for the 30 percent, 60 percent and Owner Acceptance Review of vendor-supplied calculations for the calculation reconstitution project.

  The inspectors concluded that the finding was greater than minor, as the finding represented a programmatic deficiency associated with the calculation reconstitution project that, if left uncorrected, would become a more significant concern due to calculation errors. The design deficiency did not result in a loss of function per Generic Letter 91-18 as sufficient emergency diesel generators remained available through administrative controls to provide electrical power for operators to promptly restart the control room ventilation system, hence the finding screened as very low safety significance (Green). (Section 4OA5.1)

B. Licensee-Identified Violations

A violation of very low significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

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.

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.

2. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

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 or not 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 for both units constituted three inspection procedure samples:

- Auxiliary Feedwater System Turbine- and Motor-Driven;
- Component Cooling (CC) System safety-related portions; and
- Emergency Core Cooling System.

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 service water (SW) system. 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 SW system 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 (CAPs) 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.

The inspectors also reviewed selected documented issues to determine if the issues were properly addressed in the licensee's corrective action program.

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 to impact equipment which 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 FZ-304S/304N; Auxiliary Feedwater Pump Room North Section;
- Fire Zone FZ-306/307; Battery Room-D06 and Battery Room-D05;
- Fire Zone FZ-305; 4160-Volt Vital Switchgear Room;
- Fire Zone FZ-308/309; Diesel Room-G01 and Diesel Room-G02;
- Fire Zone FZ-151; Containment Spray and Safety Injection Pump Room;
- Fire Zone FZ-318; Cable Spreading Room;
- Fire Zone FZ-237; Boric Acid Tank and Component Cooling Water (CCW) Heat Exchanger Area; and
- Fire Zone FZ-310; Air Compressor Room.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

.1 Biennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed documents associated with inspection, cleaning, and performance trending of heat exchangers primarily focusing on the 12D CCW heat exchanger and the instrument air compressor aftercooler heat exchanger HX-49A. These heat exchangers were chosen based upon their importance in supporting required safety functions, as well as their relatively high risk achievement worth in the plant-specific risk assessment. Also, these heat exchangers were not previously selected for a biennial heat sink review. The CC heat exchanger was also selected to evaluate the licensee's thermal performance testing methods. During the inspection, the inspectors reviewed calculations that indicated proper heat transfer. The inspectors reviewed the documentation to confirm that the inspection methodology was consistent with accepted industry and scientific practices, based on review of heat transfer texts and Electrical Power Research Institute standards. Specifically, the inspectors reviewed the licensee's heat transfer related calculations and/or maintenance activities to confirm that the minimum design heat transfer capability was maintained for these heat exchangers, in accordance with licensee commitments to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," and limiting design performance values identified in the FSAR.

The inspectors' review of licensee activities and documents regarding the 12D CCW heat exchanger and the instrument air compressor aftercooler HX-49A was in accordance with the biennial review sections of Inspection Procedure 71111.07, "Heat Sink Performance."

The inspectors reviewed documents associated with licensee controls for the ultimate heat sink (UHS) to ensure functionality during adverse weather conditions, (e.g., icing or high temperatures). The inspectors also reviewed recent inspection results documentation for intake structures. Review of these documents met the procedure requirements for verifying two attributes of the UHS.

The inspectors reviewed CAPs concerning heat exchanger and UHS performance issues to verify that the licensee had an appropriate threshold for identifying issues and entering them in the corrective action program. The inspectors also evaluated the effectiveness of the corrective actions for identified issues, including the engineering justification for operability.

The documents that were reviewed are listed in the Attachment to this report. The review of the CCW and the instrument air heat exchangers constituted two inspection procedure samples.

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 March 14, 2006, the inspectors observed the operating crew performance during a simulator as-found requalification examination. The inspectors also reviewed some of the changes to the simulator model against modifications made in the plant. 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) NP-2.1.1, "Conduct of Operations."

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:

- Unit 1 and Unit 2 SW System; and
- Unit 1 and Unit 2 Safety Injection Accumulators.

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 the following maintenance activities, completing risk assessment and emergent work control inspection procedure samples. 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 was 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 five inspection procedure samples:

- Planned and emergent maintenance during the week of February 20, 2006;
- Planned and emergent maintenance during the week of February 27, 2006;
- Planned and emergent maintenance during the week of March 6, 2006;
- Planned and emergent maintenance during the week of March 13, 2006; and
- Planned and emergent maintenance during the week of March 27, 2006.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Licensee Defeated Design Basis Leakage Detection Capability

a. Inspection Scope

The inspectors reviewed selected operability evaluations (operability recommendations (OPRs)) associated with issues entered into the licensee's corrective action system. 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. Review of OPR000170, "Design Basis Leakage Detection Capability May Have Been Defeated," constituted one inspection procedure sample.

b. Findings

Introduction: The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for the failure to maintain the design basis and configuration control for the detection of recirculation system leakage from the containment sump isolation valve cylinders (valves SI-850A and SI-850B for Units 1 and 2). This issue was initially identified by the inspectors during walkdowns and reviews of the containment sump recirculation piping in November/December 2005; however, at that time, the issue was not recognized by the licensee as part of the design basis of the facility.

Description: In November and December 2005, the inspectors conducted an in-depth review of the long-term emergency core cooling system, in response to a 10 CFR 50.72 report made on November 8, 2005, (NRC Inspection Report 2005013, Section 4OA2). Following a walkdown of the recirculation piping and review of the design basis, the inspectors questioned the ability of plant operators to detect containment recirculation sump leakage from the containment sump isolation valve cylinders located in the tendon gallery.

Unit 1 and Unit 2 have two containment sump recirculation lines each, with a remotely operated valve (SI-850) at the end of each line in the containment. The purpose of the SI-850 valves was to ensure the recirculation pipe inside the containment could be isolated in the event of a passive failure of the recirculation pipe or SI-850 valve cylinder assembly, both located in the tendon gallery. The inspectors noted that the recirculation piping in the gallery was contained in a pipe sleeve that passed through the tendon gallery wall into the auxiliary building; however, the sleeve was sealed on the auxiliary building side of the penetration. The licensee had asserted that direct detection of leakage from the valve cylinders was not part of the licensing and design basis of the facility.

While developing a response to a January 10, 2006, request for additional information (ADAMS ML060030437) from the Office of Nuclear Reactor Regulation regarding the November 8, 2005, 10 CFR 50.72 report, the licensee discovered additional information. Specifically, the licensee determined that, in fact, detection of cylinder leakage through the pipe sleeve into the auxiliary building was part of the systems' design and licensing basis. Section 6.2 in the Point Beach FSAR, stated, in part, that the containment sump recirculation piping passes through a set of sleeves between the tendon gallery and auxiliary building, and that leakage detection exterior to containment was achieved through the use of the auxiliary building sump level indication. In addition, original licensing correspondence regarding the unique containment sump isolation valve design at the plant credited the pipe sleeves passing leakage into the auxiliary building, as the method of detection of a leak post-accident.

The licensee initiated a CAP for this condition adverse to quality and performed an operability evaluation that concluded the condition was non-conforming, in accordance with NRC Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability." At the conclusion of the inspection period, the licensee had not determined the cause or determined why the pipe sleeves were previously sealed. In addition, the licensee initiated a CAP to determine why the design and licensing basis of this particular system had not been not well understood by plant staff.

Analysis: The inspectors determined that the licensee's failure to maintain the design basis and configuration for the leakage detection of this system is a performance deficiency warranting a significance evaluation. The inspectors concluded that this finding is greater than minor because it was associated with the affected design control and the equipment performance attributes of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the reliability and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors evaluated this finding using the guidance provided in Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined the finding is a design or qualification deficiency confirmed to not result in loss of function per NRC Generic Letter 91-18. Therefore, the inspectors determined that this finding is a licensee performance deficiency of very low risk significance (Green).

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to this, the licensee failed to assure that the applicable regulatory requirements and design basis were incorporated into design changes which modified four auxiliary building penetrations for the safety-related recirculation line sleeves. Pipe sleeves which were installed around the recirculation piping for licensed operators to detect post-accident leakage from the Unit 1 and 2, SI-850A and SI-850B valve cylinders (located in the tendon gallery) were sealed by a plant modification which defeated the design basis leakage detection capability for this system. Failure to maintain adequate design control for these systems is a violation of 10 CFR 50, Appendix B, Criterion III. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program as CAP069723, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2006002-01; 05000301/2006002-01).

The licensee had not completed a causal evaluation by the end of the inspection period; however, the licensee had created a corrective action to evaluate resolution of this non-conforming condition. In addition, the licensee also identified existing methods which would be used to identify potential leakage of the valve cylinders.

.2 Safety Function Determination for Containment Accident Sump Isolation Valves

a. Inspection Scope

The inspectors reviewed selected OPRs associated with issues entered into the licensee's corrective action system. 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 if compensatory measures were implemented, as required. The inspectors assessed whether system operability was properly justified and that the system remained available. This review of OPR000171, "Safety Function for Containment Sump B Isolation Valves," constituted one inspection procedure sample.

b. Findings

Introduction: The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for the failure to ensure the safety function of the containment sump isolation valves was maintained and tested in accordance with the design and licensing basis. This issue was initially identified by the inspectors during walkdowns and reviews of the containment sump recirculation piping in November/December 2005; however, at that time, the issue was not recognized by the licensee as part of the design and licensing basis of the facility.

Description: In November and December 2005, the inspectors conducted an in-depth review of the long-term emergency core cooling system, in response to a 10 CFR 50.72 report made on November 8, 2005, (NRC Inspection Report 2005013, Section 4OA2). Following a walkdown of the Unit 1 SI-850A/B containment sump recirculation valves and review of the design and licensing basis, the inspectors questioned why the licensee did not consider the Unit 1 or Unit 2 SI-850A/B valves to have a safety function in the closed direction.

Specifically, FSAR Section 6.2, stated, in part, that the recirculation sump line had two remotely operated valves with the first valve located at the end of the pipe in the containment such that the line inside containment could be isolated in the event of a passive failure. Section 6.2 further stated that the passive failure of one suction line (presumably excessive packing or weld leakage) would not impair the operation of the redundant valve. However, the inspectors noted that in the current plant procedures for quarterly and refueling inservice testing of the valves, the plant DBDs, and in discussions with licensee personnel, the SI-850A/B valves were only credited as having an open safety function.

Subsequent to this, the licensee initiated CAP069116, “Apparent Discrepancy in the Defined Safety Function for SI-850 Valves.” During the review of this issue, the licensee determined that the design and licensing basis for the closed safety function of these valves was not properly implemented in accordance with the facility’s license, and required codes or standards. In addition, the licensee discovered information regarding the license and design basis from original plant licensing in 1970, which further corroborated the FSAR statements that the SI-850A/B valves had a closed safety function.

On January 18, 2006, the licensee initiated CAP069881, “Safety Function for Containment Sump B Isolation Valves,” which described the condition adverse to quality and initiated an operability evaluation. Operability evaluation OPR000171 addressed the following with respect to the SI-850A/B valves for Units 1 and 2: ability of the SI-850A/B valves to close; ability of the SI-850A/B valves to limit leakage from containment when closed; compliance with control room habitability and 10 CFR Part 100 dose limits; ability to detect and isolate a passive leak; and compliance with environmental qualification requirements. The licensee’s evaluation concluded the valves were operable, but in a nonconforming condition. In addition, the licensee initiated a condition report to determine why the design and licensing basis of this particular system was not well understood by plant staff previously.

<u>Analysis</u>: The inspectors determined that the licensee’s failure to maintain the design basis and license basis for the closed safety function of the SI-850A/B valves is a performance deficiency warranting a significance evaluation. The inspectors concluded that this finding is greater than minor because it is associated with the design control, equipment performance and maintenance and testing procedure quality attributes of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the reliability and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors evaluated this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined the finding is a design or qualification deficiency confirmed to not result in loss of function per NRC Generic Letter 91-18. Therefore, the inspectors determined that this finding is a licensee performance deficiency of very low risk significance (Green).

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions. Contrary to this, the licensee failed to assure that the applicable regulatory requirements and design basis were correctly incorporated into the specifications and procedures concerning the closed safety function of the SI-850A/B valves. Failure to maintain adequate design control for the safety-related closed function of these components is a violation of 10 CFR 50, Appendix B, Criterion III. Because of the very low safety significance of this finding and because the issue was

entered into the licensee's corrective action program as CAP069891, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2006002-02; 05000301/2006002-02).

The licensee conducted Apparent Cause Evaluation ACE002003 and concluded that the cause of the inservice testing program not considering the safety function of the SI-850A/B valves in the closed direction was inappropriate clarification in the background documentation when this issue was first identified by the NRC in Inspection Report 92-008, dated May 28, 1992. More recent inservice testing program and code changes continued to dilute the original background document statements for the SI-850A/B valves until the program no longer reflected the design and licensing basis functions for these valves in the closed direction. The licensee also completed and planned several corrective actions to address this issue, which included, but were not limited to: revision of the inservice testing program documents; revision of the quarterly inservice test procedures for the valves; revision of the shutdown inservice test procedures for the valves; revision of the leakage reduction and preventive maintenance program testing of the valves; revision of the DBD for the residual heat removal system; reinforcement of the expectations with engineering staff on the use of DBDs and inservice testing background documents; and development of a project plan to update the inservice test background document which was last updated during the third inservice testing interval prior to 2002.

.3 Additional Operability Evaluations Reviewed

a. Inspection Scope

The inspectors reviewed selected OPRs associated with issues entered into the licensee's corrective action system. 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 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:

- OPR000159; FSAR Does Not Match Plant Configuration for Emergency Diesel Generator (EDG) Protection (CAP067946);
- OPR000167; FSAR Statement Concerning Post Loss of Coolant Accident (LOCA) Hydrogen Generation May Not be Valid (CAP069267);
- OPR000168; Mis-coordination With 1(2)B-30 and Q-List Discrepancy (CAP069465); and
- OPR000179; SI-850 Solenoid Valves Fail Minimum Voltage Criteria in Calc. 2005-008 (CAP071048).

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 commencement of 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.

During this inspection period, the inspectors completed the following inspection procedure samples, which constituted five quarterly inspection procedure samples:

- Reviewed the documentation for and observed the conduct of alternating current induction motor circuit evaluation testing for the electric fire pump and replacement and retest of the discharge check valve during the week of February 20, 2006;
- Reviewed the documentation for and observed the conduct of alternating current induction motor circuit evaluation testing and the results for the CC pump 2P-11A motor on March 6, 2006;
- Reviewed the documentation for and observed racking out of breaker 2B52-27C for SW pump P-032E and replacement, and operation of the replacement breaker on March 13, 2006;
- Reviewed the calibration of pressure instrument, PIC-639 channel 1, for indication and auto-start of the CCW pump and confirmed the functionality of the auto-start feature on low discharge pressure on March 20, 2006; and
- Reviewed the documentation for and observed racking-out and replacement of breaker 2B52-38A for Unit 2 containment spray pump 2P-14A and replacement, and operation of the replacement breaker on March 1, 2006.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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, properly documented, as-left setpoints were within required ranges, and the calibration frequency was 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 declared inoperable;
- Where applicable for safety-related instrument control surveillance tests, reference setting data was 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 quarterly inspection procedure samples:

- IT-60, Containment Isolation Valve Quarterly Test of WL-1721, Reactor Coolant Drain Tank Suction Containment Isolation Valve;
- IT-07, Service Water Quarterly Test of P-32 A/B/C Service Water Pumps and associated discharge check valves ;
- IT-06, Containment Spray Pumps and Valves Quarterly Test for Unit 2 and vibration and lube oil sample results for the last 2 years; and
- TS-6, Unit 2 Rod Exercise Test.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed a screening review of mid-December 2005 revisions to the following portions of the Point Beach Nuclear Plant Emergency Plan to determine whether any changes made in these revisions may have decreased the effectiveness of the licensee's emergency planning: Section 1, Revision 27; Section 5, Revision 49; Section 7, Revision 49; Section 8, Revision 47; Section 9, Revision 38; Appendix A, Revision 25; Appendix D, Revision 25; and Appendix M, Revision 0. The screening review of these revisions did not constitute an approval of the changes and, as such, the changes are subject to future NRC inspection to ensure that the emergency plan continues to meet NRC regulations.

These activities completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed an Emergency Preparedness Quarterly Drill/Training evolution on March 29, 2006, completing one drill sample. The inspectors observed activities in the Technical Support Center and attended 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.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Initiating Events

The inspectors reviewed the licensee's recent Performance Indicator submittal. The inspectors used performance indicator definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, to assess the accuracy of the PI data. The inspectors reviewed selected applicable conditions and data from logs, Licensee Event Reports, and CAPs from July 2002 through July 2004. 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 four inspection procedure samples:

Unit 1

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

Unit 2

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

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 system 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 quarter. 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.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000266/2005008-00, Manual Reactor Trip and Auxiliary Feedwater Actuation Due to Circulating Water (CW) Pump Failure

a. Inspection Scope

A manual reactor trip occurred on December 13, 2005, due to a loss of condenser vacuum caused by a mechanical failure of the running CW pump 1P-30B. At the end of the previous inspection period (NRC Inspection Report 2005013), the licensee was performing a root cause evaluation of this failure. During the current inspection period, the inspectors reviewed the completed evaluation.

b. Findings

Introduction: A Green finding with no associated violation was self-revealed for the licensee's failure to provide an adequate maintenance procedure for CW pump 1P-30B. Lack of appropriate maintenance to maintain required clearances, due to inadequate procedures, resulted in the lower shaft sleeve failing directly above the flange where the shaft sleeve attached to the guide vane. The failure of the shaft sleeve caused increased vibration which resulted in low stress, high cycle fatigue of the coupling bolts. When the coupling bolts sheared, the pump failed, causing a rapid loss of condenser vacuum.

Description: On December 13, 2005, at 3:38 a.m., Unit 1 control room operators received multiple secondary system alarms and indication that condenser vacuum was lowering rapidly. At 3:39 a.m., operators manually tripped the reactor in anticipation of a total loss of condenser vacuum. The cause of the alarms and lowering condenser vacuum was the failure of CW pump 1P-30B, the only operating pump for Unit 1 due to colder lake temperatures. The pump coupling bolts failed, separating the pump shaft from the motor. The instantaneous loss of pump 1P-30B increased pressure in the condenser which caused a rapid loss of condenser vacuum.

Subsequent inspection of the pump revealed that the lower shaft sleeve failed directly above the flange, where the shaft sleeve attached to the guide vane. The failure of the shaft sleeve caused increased vibrations, which resulted in a low stress, high cycle fatigue of the coupling bolts. Licensee analysis revealed that the cause of the lower shaft sleeve failure was due to a progressive fracture mechanism, such as fatigue. The cause of the fatigue at the lower shaft sleeve was excessive clearance at the pump

Cutless® bearings and between the pump impeller, guide vanes and inlet casing. The excessive clearances caused increased vibration which ultimately led to failure of the lower shaft sleeve which subsequently led to failure of the coupling bolts and loss of pump 1P-30B.

The licensee's root cause evaluation determined that the cause of the excessive clearances was an inadequate Routine Maintenance Procedure (RMP), RMP-2112, "Circulating Water Pump Rotating Assembly, Overhaul, and Installation." Neither the current procedure nor the previous revisions identified refurbishment specifications or tolerances for critical dimensions within the CW pumps. In addition, the procedures did not have a specific step to check for shaft sleeve cracks using nondestructive examination techniques. A review of past site operating experience regarding these pumps, revealed that in 1994, CW pump 2P-30A had experienced a similar failure.

The licensee's evaluation also identified that the micro-hardness test results of the coupling bolts showed decarburization of the thread surfaces in excess of that allowed per American Society for Testing and Materials Standard A-547. The decarburization was considered a contributing factor in the failure of the coupling bolts. Pump 1P-30B was last disassembled and inspected in April 1993. The CW pumps had a 10.5-year inspection and preventive maintenance frequency and pump 1P-30B had not been inspected for 12.5 years; however, this was within the licensee's administrative grace period of 125 percent.

<u>Analysis</u>: The inspectors determined that failure to have an adequate maintenance procedure for specification and maintenance of acceptable tolerances for critical dimensions within the pump is a performance deficiency warranting a significance evaluation. The inspectors concluded the finding is greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on September 30 2005, 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).

<u>Enforcement</u>: The failure to establish and implement an adequate maintenance procedure for the CW pumps was not an activity affecting quality subject to 10 CFR Part 50, Appendix B, nor a procedure required by license conditions or TSs. Therefore, while a performance deficiency existed, no violation of regulatory requirements occurred. This was considered a finding of very low safety significance (FIN 05000266/2006002-03). In addition, LER 05000266/2005008-00 is considered closed.

The licensee entered the event into their corrective action program as CAP069331 and took immediate corrective actions to evaluate the physical failure mechanism and repair the 1P-30B pump. Repair included the lower shaft sleeve, replacement of the coupling, coupling bolts, the Cutless® bearings, and Belzona® repair of worn areas on the guide vanes and pump impeller.

The licensee performed a root cause evaluation, identifying the cause cited above and several actions to prevent recurrence. These actions included but were not limited to: developing appropriate replacement and refurbishment specifications and tolerances within RMP-9112, "Circulating Water Pump Rotating Assembly Removal, Overhaul, and Installation"; restoration or refurbishment of CW pumps to within manufacturer's tolerances; and replacement of all normally inaccessible bolting during refurbishment.

.2 (Closed) LER 05000266/2005007-00, Control Rod Movement with Refueling Cavity Water Level Below T.S. 3.9.6 Limit

Technical Specification 3.9.6, "Refueling Cavity Water Level," requires, in part, that refueling cavity water level remain greater than 23 feet for core alterations, except during control rod latching and unlatching activities. However, reactor operators identified in November 2005, that procedure RP-4A, "Full-Length Control Rod Drive Shaft Unlatching and Latching," also contained actions to perform control rod drag testing with water in the refueling cavity at a height of less than 23 feet. This testing consisted of latching a control rod, raising the control rod 10 feet while monitoring a load cell for weight changes and then setting the control rod back in place and unlatching. While latching and unlatching were permitted at this water level height, movement of the control rod was not allowed by the TS. Therefore, contrary to the TS, this activity had taken place with the water level in the refueling cavity less than the 23 feet required by TSs. This licensee-identified violation is also discussed in Section 4OA7 of this report. This LER is considered closed.

4OA5 Other

.1 (Closed) Unresolved Item URI 05000266/2005012-02; 0500301/2005012-02, Effects of Elevated Temperatures on Control Room Instrumentation

Introduction: The inspectors identified a Green finding associated with a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," when the licensee failed to consider the effects of elevated control room temperatures on instrument inaccuracies following a design basis LOCA which could potentially affect mitigation of the event.

Description: During the Problem Identification and Resolution inspection conducted from September 12 through October 6, 2005, the team identified an unresolved item related to the effects of elevated control room temperatures on instrument accuracies and accident mitigation during a design basis LOCA. FSAR Section 9.8, "Control Room Ventilation System," stated that during a design basis LOCA concurrent with a loss of offsite power and a single failure, the control room ventilation fans would not be automatically loaded onto an EDG. Further, the fans may not be manually started by operators for as long as 2 hours due to the need to limit EDG loading. During this time, control room temperature could increase to 112 degrees Fahrenheit (EF). The FSAR further stated that, because the instrumentation and associated circuitry located in the control room was generally rated for an ambient temperature range of 40EF to 120EF, it could be concluded that this equipment would perform the intended function during a 2-hour loss of control room ventilation.

Because elevated temperatures could affect the accuracy of control room instruments, the team reviewed the following licensee calculations to verify that they included the effects of elevated control room temperatures on instrument accuracies:

- PBNP-IC-08, “Pressurizer Level Instrument Uncertainty/Setpoint,” Revision 2;
- PBNP-IC-12, “Low and High Pressurizer Pressure Reactor Trip Instrument, Revision 2; and
- PBNP-IC-17, “Low Range Containment Pressure Instrument Loop Uncertainty/Setpoint Calculation,” Revision 0.

The team found that assumptions in licensee instrument loop uncertainty calculations for selected control room instruments that could be used during a LOCA (reactor coolant system pressure, containment pressure, and pressurizer level) included control room temperature at 75EF ± 10EF, with negligible effect on instrument inaccuracies. The calculations did not evaluate the effects of elevated control room temperatures up to 112EF on instrument accuracies. Increased instrument inaccuracies during a design basis LOCA could potentially affect mitigation of the event.

The licensee entered this issue into the corrective action program as CAP067405 and CAP067700. A licensee root cause evaluation determined that the subject calculations did not appropriately address control room temperature effects in instruments under accident conditions. This occurred due to less than adequate program management of the current calculation reconstitution project which led to inadequate management of emerging issues. This allowed an ongoing problem with the licensee’s calculation comment resolution process to continue throughout the development, completion, and final acceptance of the vendor-supplied calculation, until the issue was identified by the inspectors. Specifically, the 30 percent review of the calculation did not address design basis accident conditions as was requested by a licensee comment in the calculation development process, nor did a licensee 60 percent review question to the vendor concerning whether the control room temperature calculation of record should be used, get adequately addressed in the calculation supplied by the vendor.

<u>Analysis</u>: The inspectors determined that the licensee’s failure to consider the effects of elevated control room temperatures on instrument inaccuracies following a design basis LOCA is a performance deficiency that could potentially affect mitigation of the event and warranted a significance determination. The inspectors concluded that the finding is greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues and Cross-Cutting Aspects,” issued on September 30, 2005, as the finding represented a programmatic deficiency associated with the calculation reconstitution project that, if left uncorrected, would become a more significant concern to calculation errors.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." The design control deficiency did not result in a loss of operability as sufficient EDGs remained available to provide electrical power for operators to promptly restart the control room ventilation system, hence the finding screened as very low safety significance (Green).

Enforcement: 10 CFR, Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for the identification and control of design interfaces and for coordination among participating design organizations; also, the design control measures shall provide for verifying or checking the adequacy of design, such as by performance of design reviews or by the use of alternate calculational methods. Contrary to these requirements, the licensee's design reviews of vendor provided calculations did not appropriately address elevated control room temperature effects in instruments under accident conditions. Because this violation was of very low safety significance and has been entered into the licensee's corrective action program (as CAP067405 and CAP067700), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 0500266/2006002-04; 05000301/2006002-04).

The licensee planned to revise the subject calculations to address instrument uncertainty for elevated control room ambient temperature caused by a temporary loss of heating ventilation and air conditioning, and to revise the FSAR as necessary. Prompt operability was addressed by a calculation check of the review of elevated temperature effects on control room instruments and existing operational procedural guidance to restore control room heating ventilation and air conditioning per emergency operating procedures.

Corrective actions to prevent recurrence included providing more detailed information on the conduct of the 30 percent, 60 percent and Owner Acceptance Review of vendor-supplied calculations; strengthening the comment coordination process by requiring the use of comment review forms at the 60 percent review; having the 30 and 60 percent comment review forms present during the Owner Acceptance Review; requiring the use of a calculation checklist; and including discussion on performance issues at the weekly project meetings with the vendor. The vendor was also required to perform an internal review of the event to identify why its independent review and approval did not address accident conditions for control room temperatures.

4OA6 Meetings

.1 Exit Meeting

On April 4, 2006, 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 in nature.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Heat Sink Performance biennial inspection with Mr. D. Koehl, Site Vice-President and G. Packard, Operations Manager, on February 10, 2006; and
- Emergency Preparedness inspection with Ms. M. Ray on February 3, 2006.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Technical Specification 3.9.6, “Refueling Cavity Water Level,” requires, in part, that refueling cavity water level height remain greater than 23 feet for core alterations, except during control rod latching and unlatching activities. However, reactor operators identified in November 2005 that procedure RP-4A, “Full-Length Control Rod Drive Shaft Unlatching and Latching,” also contained actions to perform control rod drag testing with water level in the refueling cavity less than 23 feet. This testing consists of latching a control rod, raising the control rod 10 feet while monitoring a load cell for weight changes and then setting the control rod back in place and unlatching. While latching and unlatching were permitted at this water level height, movement of the control rod was not allowed by the TS. Therefore contrary to the TSs, this activity had taken place with less than the 23 feet of water (in this case, 11 feet and 8 inches of water) in the refueling cavity required by TSs. This licensee-identified finding was entered into the corrective action program and is of very low significance because while the movement of irradiated fuel assemblies is an accident initiator, the dropping of a control rod within the guide tube is a mitigating activity, not an accident initiator. The licensee initiated corrective actions to revise the procedure and submit a license amendment request.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

C. Butcher, Site Engineering Director
G. Casadonte, Fire Protection Coordinator
F. Flentje, Senior Regulatory Compliance Engineer
T. Gemskie, Emergency Preparedness Supervisor
B. Grazio, Regulatory Affairs Manager
C. Hill, 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
R. Ladd, Fire Protection Engineer
M. Lorek, Plant Manager
J. McCarthy, Director of Site Operations
J. McNamara, Engineering Supervisor
G. Packard, Operations Manager
L. Peterson, Design Engineer Manager
M. Ray, Emergency Planning Manager
D. Schuelke, Radiation Protection Manager
J. Schweitzer, Site Engineering Director
G. Sherwood, Engineering Programs Manager
C. Sizemore, Training Manager
N. Stuart, Maintenance Manager
P. Wild, Design Engineering Projects Supervisor
R. Womack, Fleet Program Engineering Manager

Nuclear Regulatory Commission

C. F. Lyon, Point Beach Project Manager, NRR
P. Louden, Chief, Reactor Projects, Branch 5

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000266/2006002-01 05000301/2006002-01	NCV	Failure to Adequately Maintain Leak Detection Capability (Section 1R15.1)
05000266/2006002-02 05000301/2006002-02	NCV	Failure to Adequately Maintain Safety Function for SI-850 Valves in the Closed Direction (Section 1R15.2)
05000266/2006002-03	FIN	Self-Revealed Failure of Unit 1 Circulating Water Pump 1P-30B Due to Inadequate Maintenance (Section 4OA3.1)
05000266/2006002-04 05000301/2006002-04	NCV	Failure to Address Effects of Elevated Temperatures on Control Room Instruments (Section 4OA5.1)

Closed

05000266/2005012-02 05000301/2005012-02	URI	Effects of Elevated Temperatures on Control Room Instruments (Section 4OA5.1)
05000266/2005007-00	LER	Control Rod Movement with Refueling Cavity Water Level Below T.S. 3.9.6 Limit (Section 4OA3.2)
05000266/2005008-00	LER	Manual Reactor Trip and Auxiliary Feedwater Actuation Due to Circulating Water Pump Failure (Section 4OA3.1)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

CAP069757; Buried Piping ISI Requirements Not Being Met
CAP052827; Safety Concern, Plugged Drains in Service Water System
CAP054125; Excessive Service Water Silting
CAP061286; 0SW-396A Check Valve Found Stuck Open
WO 0509767 Inspect Service Water Piping HB-19
CL 10J; Safeguards Service Water System Checklist Unit 1
CL 10B; Service Water Safeguards Lineup
System Health Report Service Water; January 13, 2006
CL 13E Part 1; Auxiliary Feedwater Valve Lineup Turbine-Driven Unit 2
CL 13E Part 1; Auxiliary Feedwater Valve Lineup Turbine-Driven Unit 1
CL 13E Part 2; Auxiliary Feedwater Valve Lineup Motor-Driven
CL-CC-001; Component Cooling Unit 2 Valve Checklist
CL 7A; Safety Injection System Checklist, Unit 1; Revision 23
1-TS-ECCS-001; Safeguard Systems Valve and Lock Checklist (Monthly), Unit 1; Revision 5
1-TS-ECCS-002; Safeguards System Venting (Monthly), Unit 1; completed March 1, 2006

Section 1R05: Fire Protection

Fire Hazard Analysis Report (FHAR) for applicable Fire Areas Reviewed; December 2005
Fire Area Analysis Summary Report for applicable Fire Areas Reviewed; August 8, 2005
CAP070837; Calculation Not Revised to Reflect Modification Changes; March 3, 2006
Plant Modification 00-016; Appendix R Upgrade of 8' PAB Sprinkler System; May 24, 2001
Calculation 00087.01.00012.02-TR01; Evaluation of Fire Suppression Systems Covering Fire Zones 142, 151, 156, and 166; Revision 0

Section 1R07: Heat Sink Performance

OPR000137; Revision 0 (CAP064141); Increased Valve Weights in SW Return Piping from CCW Heat Exchangers; May 3, 2005
NPM 2004-0183; Focused Self-Assessment Report of PBNP Heat Exchanger Condition Assessment Program - PBSA-ENG-03-11; March 18,2004
OI 151; HX-012C and D Component Cooling System Heat Exchanger Data Collection; completed October 15, 2000
OI 151; HX-012C and D Component Cooling System Heat Exchanger Data Collection; completed April 14, 2002
Heat Exchanger Specification Sheet; Component Cooling Heat Exchanger; February 24, 1992
TIN 2000-1382; CCW HXs HX-12C & 12D Thermal Performance Test Data Evaluation and Uncertainty Analysis; Revision 1
PGT 2002-1270; CCW HXs HX-12C & 12D Thermal Performance Test Data Evaluation and Uncertainty Analysis; Revision 0
Completed Work Order Number 206375; Inspect & Clean Component Cooling Heat Exchanger HX-012D; March 17, 2003
Completed Work Order Number 407564; Inspect & Clean Component Cooling Heat Exchanger HX-012D; November 12, 2004

HX-01; Heat Exchanger Condition Assessment Program; Appendix E; Annual Cycle Inspection Schedule; Revision 1; February 25, 2004
GL-89-13 Program Document; Revision 5; September 9, 2005
EE 2001-0036; CCW HX Testing & Acceptance Criteria, Revision 0; November 27, 2001
EE 2003-0007; CCW HX Tubing Plugging & Stabilization; Revision 2
EE 2003-0008; CCW HX Plugging Limit; Revision 3
Atlas to WEPCO Letter; January 9, 1989, CCW Hxs Vibration Analyses Atlas Job #5887
Work Order History; HX-49A; February 3, 2006
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CAP059372; CCW HX Baffle Degradation, Tube Vibration, and End of Life Issues; September 21, 2004
CAP060468; HX-12D Epoxy Damage; November 10, 2004
CAP059743; Contingency WO Planning not Done in Time, Jeopardizes GL 89-13 CCW project; October 6, 2004
CAP064944; Fish Intrusion: Significant Number of Fish in Forebay; June 4, 2005
CAP069844; GL 89-13 Test Vendor Going Out of Business - No Apparent Replacement; January 16, 2006
CAP068336; Service Water (GL 89-13) Program (FSA); October 26, 2005
CAP069446; Known Failure Rate of 1SW-00322 and 1SW-00360; December 19, 2005
CAP069447; Work Around Created Which Must Be Dealt With by the Operations Department; December 19, 2005
CAP069527; CCW HX Pipe Supports With Improper Cotter Pins; December 22, 2005
CAP068651; CCW Transferring from Unit 2 to Unit 1 CCW Systems; November 6, 2005
CAP068674; 1HX-12A Component Cooling Water Heat Exchanger Needs to be Returned to Service; November 7, 2005
CAP069897; GL 89-13, Action V, Implementation Question; January 18, 2006
CAP070055; Potential Conditions Adverse to Quality Closed to PHC Priority List; May 12, 2006
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*CAP070354; Apparent Discrepancy in FSAR 9.1 and 14.3.4; February 7, 2006

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Section 1R11: Licensed Operator Qualifications

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Section 1R12: Maintenance Effectiveness

CAP063969; Service Water Leakage; G02 Emergency Diesel Generator Room
CAP070893; Review Service Water Maintenance Rule function for EDG
CAP070743; Review EOP scoping criteria for the Maintenance Rule
CAP068857; Service Water Pump 0P-32E Showing Signs of Degradation
CAP067043; Test Results from P-32D Service Water Pump Place it near Alert Limit
ACE1869; Service Water Leakage; G02 Emergency Diesel Generator Room
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System Health Report, Diesel Generator System (DG); October 26, 2005
System Health Report, Component Cooling Water System (CCW); December 22, 2005
Performance Criteria Assessments for DG since January 1, 2006
WO0501655; SW-55 Investigate and Repair Leak
WO0512399; SW-360 Replace Stem
WO0406909; 2Piping HB-19 Replace Piping
WO0501443; 2SW-850A Clean off Bolting and Flange
Documentation of Maintenance Rule Performance Criteria, File T7.2.6
Safety Injection System Health Report; December 22, 2005
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MRE000403; 2LI-934 ‘B’ SI Accumulator Level Failed Low; May 17, 2005
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MRE000479; Unit 2 “B” Accumulator Level Channel LI-935 Failed Low; October 1, 2005
CAP068686; 1LI-934, SI Accumulator Level, Fails Channel Check; November 8, 2005
CAP069429; LT-935A Called OOS Due to Reading 10% Below LT-934 (‘B’ SI accumulator); December 17, 2005
CAP069810; LT-935A (U-2 ‘B’ SI Accumulator Level Indicator) Appears to be Failing Low; January 14, 2006
CAP069975; LT-935 Called OOS Due to Reading 7% Below LT-934 (‘B’ SI accumulator); January 21, 2006
CAP068439; 1P-15A SI Pump Seal Leak During IT-530C LRMP Piggyback Test Train A; October 30, 2005

CAP070575; NSAL 06-2 - Potential to Draw Air Into ECCS Pumps When RWST Is Exhausted; February 21, 2006

Section 1R13: Maintenance Risk Assessment and Emergency Work Evaluation

PNP Schedule J07A2 Workweek for March 7, 2006
CAP070863; Schedule Activity Deleted From E-1 Risk Evaluation for J06
CAP070897; IT-07 Service Water Pump Schedule Change
CAP070708; Work Week J05 Real Time Safety Monitor Missing Some Components
CAP070904; Unplanned Yellow CDF in E0 Week
Safety Monitor Calculation Reports Units 1/2 for Weeks of February 20, February 27, March 6, March 13, and March 27, 2006

Section 1R15: Operability Evaluations

OPR000159; FSAR Does Not Match Plant Configuration for EDG Protection
OPR000168; Mis-coordination with 1(2)B-30 and Q-List Discrepancy
OPR000179; SI-850 Solenoid Valves Fail Minimum Voltage Criteria in Calc. 2005-0008
CAP067946; FSAR Does Not Match Plant Configuration for EDG Protection
CAP069465; Mis-coordination with 1(2)B-30 and Q-List Discrepancy
CAP071048; SI-850 Solenoid Valves Fail Minimum Voltage Criteria in Calc. 2005-0008

Section 1R19: Post-Maintenance Testing

RMP 9387; AC Induction Motor MCE Testing Procedure
ICP 6.15; Auxiliary Coolant System (Non-Outage)
WO 512941; Calibration of 1PIC-639
Work Order 400377; Breaker Maintenance per RMP 9303; February 28, 2006
Operating Instruction (OI) 35; 480V Electrical Equipment Operation; Revision 5
OI 163; SI, RHR, and CS Pump Runs and Venting SI Pump Casings; partial Revision 6; March 1, 2006

Section 1R22: Surveillance Testing

IT-06; Containment Spray Pumps and Valves (Quarterly) Unit 2
IT-07A/B/C; Service Water Pump Quarterly for pumps A, B, and C
IT-60; Containment Isolation Valves (Quarterly) Unit 1
CAP065874; 1WL-1721 Stroked Faster Than Expected During Inservice Testing
WO 0510586; Overhaul Valve Actuator

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Point Beach Nuclear Power Plant Emergency Plan; Section 1, Revision 27; Section 5, Revision 49; Section 7, Revision 49; Section 8, Revision 47; Section 9, Revision 38; Appendix A, Revision 25; Appendix D, Revision 25; and Appendix M, Revision 0.

Section 1EP6: Drill Evaluation

Emergency Planning Tabletop Drill, March through June 2006 Tabletop Drills, Revision 0

NRC-Identified CAPS

CAP069650; Radio Configuration Does Not Allow Console Override to Transmit in an Emergency; January 5, 2006
CAP069739; Security Testing Equipment; January 10,2006
CAP069757; Buried Piping ISI Requirements Not Being Met; January 11, 2006
CAP069767; Security Drill Program Enhancement Opportunity; January 12, 2006
CAP069781; Enhancement For Security Warehouse Search Process; January 12, 2006
CAP069891; Safety Function For Containment Sump B Isolation Valves; January 18, 2006
CAP070216; Over Due Corrective Actions In Maintenance; February 2, 2006
CAP070218; SAMG SCG-2 Data May Be Outdated; February 2, 2006
CAP070224; NRC Issues Generic Letter 2006-02: Grid Reliability; February 2, 2006
CAP070306; Generic Letter 89-13 Calculation Issues PGT-2002-1270; February 7, 2006
CAP070354; Apparent Discrepancy in FSAR 9.1 and 14.3.4; February 9, 2006
CAP070363; Closeout of CAP001537, Water Hammer in SW System, Less Than Adequate; February 9, 2006
CAP070439; Hose Reel HR-61 Incorrectly Identified on Drawing PBC-218, Sheet 2; February 14, 2006
CAP070449; NRC Request for Information: Generic Letter 2004-02, ECCS Sump Blockage; February 15, 2006
CAP070464; Possible RCS Leakage Tracking Improvement; February 15, 2006
CAP070665; NRC Identified Enhancement Opportunity with ISFSI Blast Calculation; February 24, 2006
CAP070667; NRC Identified Enhancement Opportunity with AOP-29; February 24, 2006
CAP070669; NRC Identified Enhancement Opportunity with Security LLEA Integration Plan; February 24, 2006
CAP070837; Calculation Not Revised to Reflect Modification Changes; March 3, 2006
CAP070893; Review Service Water Maintenance Rule Function For EDG; March 8, 2006
CAP070980; PBNP ENS Phone System Failure; March 13, 2006
CAP071030; Specified Requirements Unclear For Exemption 18, FPER Table 5.2.6-1; March 15, 2006
CAP071045; PAB Superstructure Calc Requires Some Improvements; March 16, 2006
CAP01020023; Potential Adverse Trend 50.59 Screen and OPRs; March 23, 2006
CAP01019916; OPR 179 Compensatory Actions May Not Meet NP 5.3.7; March 23, 2006

LIST OF ACRONYMS USED

CAP	Corrective Action Program Document
CC	Component Cooling
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CW	Circulating Water
DBD	Design Basis Document
DG	Diesel Generator
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
HX	Heat Exchanger
IMC	Inspection Manual Chapter
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NP	Nuclear Plant Procedure
NRC	Nuclear Regulatory Commission
OPR	Operability Recommendation
RMP	Routine Maintenance Procedure
SDP	Significance Determination Process
SW	Service Water
TS	Technical Specification
UHS	Ultimate Heat Sink
URI	Unresolved Item
WO	Work Order