UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

April 25, 2008

Mr. James McCarthy
Site Vice-President
FPL Energy Point Beach, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, NRC INTEGRATED INSPECTION REPORT 05000266/2008002 AND 05000301/2008002

Dear Mr. McCarthy:

On March 31, 2008, 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 with you and members of your staff on April 9, 2008.

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 NRC-identified and self-revealed findings of very low safety significance (Green) were identified. These four 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 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 a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the 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), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael A. Kunowski, Chief
Branch 5
Division of Reactor Projects

Docket Nos. 50-266; 50-301; 72-005
License Nos. DPR-24; DPR-27

Enclosure: Inspection Report 05000266/2008002; 05000301/2008002
w/Attachment: Supplemental Information

cc w/encl: M. Nazar, Senior Vice President and Nuclear
Chief Operating Officer
J. Stall, Senior Vice President and
Chief Nuclear Officer
R. Kundalkar, Vice President, Nuclear Technical Services
Licensing Manager, Point Beach Nuclear Plant
M. Ross, Managing Attorney
A. Fernandez, Senior Attorney
K. Duveneck, Town Chairman
Town of Two Creeks
Chairperson
Public Service Commission of Wisconsin
J. Kitsembel, Electric Division
Public Service Commission of Wisconsin
P. Schmidt, State Liaison Officer

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).

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Letter to J. McCarthy from M. Kunowski dated April 25, 2008

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, NRC INTEGRATED INSPECTION REPORT 05000266/2008002 AND 05000301/2008002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-266; 50-301; 72-005
License Nos:	DPR-24; DPR-27
Report No:	05000266/2008002; 05000301/2008002
Licensee:	FPL Energy Point Beach, LLC
Facility:	Point Beach Nuclear Plant, Units 1 and 2
Location:	Two Rivers, Wisconsin
Dates:	January 1, 2008, through March 31, 2008
Inspectors:	R. Krsek, Senior Resident Inspector R. Ruiz, Resident Inspector J. Neurauter, Senior Reactor Inspector K. Barclay, Reactor Engineer D. Betancourt, Reactor Engineer
Approved by:	Michael Kunowski, Chief Branch 5 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000266/2008002, 05000301/20080002; 01/01/2008-03/31/2008; Point Beach Nuclear Plant, Units 1 & 2; Adverse Weather Protection; Post-Maintenance Testing; Follow-up of Events; Other Activities.

This report covers a three-month period of inspections by resident inspectors and regional specialists. Four Green findings with associated Non-Cited Violations (NCVs) 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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding and associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," having very low safety significance (Green) for the license's failure to take prompt corrective actions to address recurring cold weather issues in the facade building which again occurred in January 2008. The failure to take prompt corrective actions led to the formation of ice on offsite power and plant equipment cable trays and cabling, which supplied offsite power to both Units' busses. The sheets of ice were also in proximity to the Unit 2 refueling water storage tank level indicators and outlet piping. The licensee initiated condition reports, took immediate corrective actions, and was performing a causal evaluation at the end of the inspection period.

  The finding is more 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. Because the ice buildup in the Unit 2 facade was an external factor and transient initiator contributor that did not contribute to both the likelihood of a reactor trip and the likelihood 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 (P.1(d)). (Section 1R01)

Cornerstone: Mitigating Systems

- Green. A self-revealed finding and an associated Non-Cited Violation of Technical Specification 5.4.1, "Procedures," having very low safety significance (Green), was identified for the license's failure to implement procedures associated with conduct of operations for plant systems. Specifically, on January 4, 2008, control room operators responded to a Unit 1 'A' Safety Injection Accumulator Level High Alarm and initiated actions to drain the accumulator, without utilizing the redundant or backup indication for the draining evolution required by plant procedure. This resulted in the inadvertent draining and inoperability of the accumulator with respect to the minimum Technical

Specification required accumulator pressure, because the level accumulator channel used to drain the accumulator had failed in the "as-is" position, causing the initial alarm. The licensee took immediate corrective actions which included restoration of the Unit 1 Safety Injection (SI) accumulator to an operable status, repair of the level indicator, and establishment of a new conduct of operations procedure. In addition, the licensee completed an apparent cause evaluation and developed additional corrective actions to correct this performance deficiency.

The finding is more than minor because it is associated with the human performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding is of very low safety significance (Green) because it did not involve a design or qualification deficiency, there was no actual loss of safety function, no single train loss of safety function for greater than the Technical Specification allowed outage time, and no risk due to external events. The inspectors also determined that the finding has a cross-cutting aspect in the area of human performance. Specifically, human error prevention techniques were not utilized following the receipt of the accumulator level alarm and during the draindown evolution (H.4(a)). (Section 4OA3.1)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding of very low safety significance (Green) and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the license's failure to implement prompt corrective actions for the degraded conditions initially identified with the single failure proof primary auxiliary building crane by maintenance personnel on January 17, 2008. As a result, on March 4, while a new fuel storage canister was being lowered in a laydown area after traversing the width of the spent fuel pool, the crane failed to the safe position with the load suspended approximately one foot off the floor. In a review of work order and corrective action history, the inspectors determined that all of the degraded conditions from January were not corrected during maintenance on February 21. The licensee entered the issue into its corrective action program and took immediate corrective actions, including repair of the crane. The licensee continued to evaluate the causes and corrective actions to address this finding at the end of the inspection period.

  The finding is more than minor because it could reasonably be viewed as a precursor to a significant event. Specifically, the failure to correct the degraded condition of the primary auxiliary building crane resulted in the failure of the single failure proof crane while in use to move loads over the spent fuel pool. The finding affected the Barrier Integrity Cornerstone and is of very low safety significance (Green) because this spent fuel pool issue did not result in the loss of spent fuel pool cooling, did not result in damage to fuel clad integrity in the spent fuel pool, and did not result in a loss of spent fuel pool inventory. 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 (P.1(d)). (Section 1R19.1)

- Green. The inspectors identified a finding of very low safety significance (Green) and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to evaluate service water piping to pipe anchor integral welded attachments in conformance with the design requirements of the design basis American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The licensee entered this issue into its corrective action program.

This finding is more than minor because it's associated with the design control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to maintain the structural integrity of the service water system, structures, and components and the operational capability of the containment fan coolers. The finding was of very low safety significance (Green) based on a Phase 1 screening in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and Appendix H, "Containment Integrity Significance Determination Process," because pressurized water reactor containment fan coolers impact late containment failure and source terms, but not large early release frequency. There was not a cross-cutting aspect to this finding. (Section 4OA5.1)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 100 percent power. On January 16, 2008, a Technical Specification (TS) required shutdown was initiated, due to the failure of electrical cabling on January 15. Electrical cables between the Unit 1 X-04 transformer and the associated 4160-Volt busses failed and caused a loss of normally supplied offsite power to Unit 1 safety busses. On February 5, following repair of the electrical cables, Unit 1 was returned to power operations, where the unit remained until the end of the inspection period.

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

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness For Impending Adverse Weather Condition – Cold Weather

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 safety-related plant equipment inside the protected area. The inspectors reviewed the corrective action documents and work orders (WOs) written for 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 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 prior winters, and which occurred again in January 2008. The failure to take prompt corrective actions led to the formation of ice on offsite power and plant equipment cable trays and cabling, which at the time was supplying offsite power to both units due to a failure of the 1X-04 transformer 4160-Volt cabling on January 16. The ice was also in proximity to the Unit 2 refueling water storage tank (RWST) level indicators and outlet piping.

Description: On January 18, 2008, plant staff identified that in the Unit 2 containment facade, 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 various other 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. Due to an unrelated failure affecting the Unit 1 electrical distribution system (discussed in Section 4OA3.2 of this report), the Unit 2 busses were cross-tied to those of Unit 1 so that the Unit 2 electrical distribution system provided offsite power to both units. Operations staff immediately requested walkdowns from system and structural engineers to determine the structural acceptability of the trays' current state, and was deemed satisfactory. The ice formation and cable trays were approximately five feet away from the Unit 2 RWST and near the RWST outlet piping, minimum flow recirculation inlet piping, and level transmitters. The icing of the cable trays was caused by the repeat freezing of a relief vent stack drain in the facade. Operations directed maintenance craft to create a tent with a temporary heat source to melt the ice and redirect the source of water. A temporary heat source was also applied to the relief vent stack.

The licensee continued with actions to melt the ice on the trays, monitor the area, and redirect the water. On January 20, 2008, the inspectors verified that the licensee's actions, including increased operator rounds and the building of the tent with a temporary heat source, were still in effect. The inspectors walked down the facade building and discovered that the licensee's corrective actions were not effective, as evidenced by the relief stack drain again being found frozen over, and the stack filled with a combination of ice and water. The licensee again took immediate actions to thaw the frozen relief stack drain and assess the potential adverse effects of the relief stack being filled with ice and water in proximity to the Unit 2 RWST. The licensee evaluations provided reasonable assurance that the relief stack would remain intact during a seismic event.

A similar performance deficiency was identified in January 2007 and was documented as a Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI in NRC integrated inspection report 05000266/2007002 and 05000301/2007002. The licensee performed an apparent cause evaluation of the associated Corrective Action Program Document-condition report (CAP) 01075828 and 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. However, since January 2007, the licensee failed to implement timely corrective actions to address this issue; although the licensee had taken action in the fall of 2007 to address some leaking relief valves in the system. Additionally, as previously discussed, once the issue reappeared on January 18, 2008, the licensee failed to take timely corrective actions to ensure the relief stack drain did not refreeze.

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 Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 20, 2007, 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.

The inspectors evaluated the finding using IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” dated January 10, 2008, for the Initiating Events Cornerstone. The transient initiator contributor was the external factor of ice buildup in the Unit 2 facade on the offsite power cable trays and did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Consequently, the finding is considered to be of very low safety significance (Green). The inspectors also determined that this finding has a cross-cutting aspect in the cross-cutting area of problem identification and resolution. Specifically, the licensee failed to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity (P.1(d)).

Enforcement: Title 10 CFR 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 non-conformances, are promptly identified and corrected. Contrary to this, the licensee had identified conditions adverse to quality associated with the formation of ice on quality structures and components in the facade during periods of cold weather, including January 2007, and failed to promptly correct those conditions prior to the next onset of cold weather. Additionally, when the freezing issues were again identified on January 18, 2008, the prompt corrective actions did not correct the condition as evidenced by the discovery of the relief stack drain being frozen again on January 20. The failure to implement prompt corrective actions for these conditions adverse to quality is a violation of 10 CFR 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 as CAP 01123137, it is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000266/2008002-01; 05000301/2008002-01).

As stated previously, the licensee took corrective actions to correct the condition upon discovery, and at the end of the inspection period, was performing an apparent cause evaluation.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

Partial system walkdowns of the following systems constituted five inspection procedure samples:

- Emergency diesel generator (EDG) G-03 when EDG G-04 was out-of-service;
- Unit 2 component cooling water (CCW) train 'B' when CCW train 'A' was out-of-service;
- Electrical busses 1A-01, 1A-02, 1A-03, and 1A-04 cross-tie lineups to the Unit 2 busses 2A-03 and 2A-04 during Unit 1 forced outage;
- Transformers 1X-01 and 1X-02 electrical backfeed lineups during Unit 1 forced outage; and
- Alignment of normal Unit 1 electrical busses 1A-01, 1A-02, 1A-03, and 1A-04 configurations at the end of the Unit 1 forced outage.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

.1 Routine Resident Inspector Tours

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 the following fire protection attributes during walkdowns: fire hoses and extinguishers were in the designated locations and available for immediate use; unobstructed fire detectors and sprinklers; transient material loading within the analyzed limits; and fire doors, dampers, and penetration seals in satisfactory condition. The inspectors also determined whether minor issues identified during the inspection were entered into the licensee's corrective action program.

The walkdown of the following selected fire zones constituted six inspection procedure samples:

- Fire Zone 583/Fire Area A01-E: Unit 2 turbine building general area, 8-foot elevation;
- Fire Zone 308/Fire Area A24: G-01 EDG room;
- Fire Zone 309/Fire Area A28: G-02 EDG room;
- Fire Zone 321/Fire Area A54: swing station battery D-305 room;
- Fire Zone 552/Fire Area A38: service water (SW) pump room; and
- Fire Zone 553/Fire Area A38: circulating water pump room.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

In March 2008 the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance;
- Crew's clarity and formality of communications;
- Ability to take timely actions in the conservative direction;
- Prioritization, interpretation, and verification of annunciator alarms;
- Correct use and implementation of abnormal and emergency procedures;
- Control board manipulations;
- Oversight and direction from supervisors; and
- Ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

This inspection constitutes one quarterly licensed operator requalification program sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors performed maintenance effectiveness reviews of the CCW systems for Units 1 and 2. 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, observation of repetitive maintenance activities in the field, 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 the systems reviewed, the inspectors reviewed significant WOs and corrective action program documents to determine whether failures were appropriately identified, classified, and corrected, and if unavailable time was correctly calculated.

The reviews of maintenance effectiveness for the following components and systems constituted one inspection procedure sample:

- Unit 1 and Unit 2 CCW systems.

b. Findings

No findings of significance were identified.

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

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

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

The reviews of maintenance risk assessment and emergent work evaluation constituted five inspection procedure samples:

- Emergent electrical bus lineups during the week of January 14, 2008;
- Planned and emergent maintenance during the week of January 21;
- Planned and emergent maintenance during the week of January 28;
- Planned and emergent maintenance during the week of February 3; and
- Planned and emergent maintenance during the week of March 24.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

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

The reviews of the following operability evaluations constituted four samples:

- CAP 01118180, Pump House Ventilation Failure During Loss of Offsite Power;
- CAP 01121068, Service Water Pump P-032E High Vibrations;
- CAP 00896611, Trip/Emergency Safeguard Feature Actuations System Setpoint Allowable Values; and
- CAP 01124170, Unit 2 Transformer 2X-04 Low Voltage Station Auxiliary Transformer Cable Tower Installation.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Routine Resident Review

a. Inspection Scope

The following engineering design package was reviewed in detail and selected aspects were verified in the field for engineering change (EC) 11877, "1X-04 temporary power cable installation."

This document and related documentation were reviewed to assess adequacy of the associated 10 CFR 50.59 safety evaluation screening; consideration of design parameters; implementation of the modification; post-modification testing, and proper updating of procedures, design, and licensing documents. The inspectors observed in-progress and completed work activities to verify that installation was consistent with the design control documents. The modification was installed to re-establish offsite power to the 4160-Volt busses 1A-03 and 1A-04 through the 1X-04 transformer, following a cable fault on the associated underground cables.

This inspection constituted one temporary modification sample.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Primary Auxiliary Building (PAB) Crane

a. Inspection Scope

The inspectors reviewed the post-maintenance testing activities and procedures associated with the return to service of the PAB crane to verify they were adequate to ensure system functional capability. The inspectors evaluated this activity for the following (as applicable): testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; tests were performed in accordance with properly reviewed and approved procedures; and test documentation was properly evaluated. The inspectors evaluated the activities against the Technical Requirements Manual, the FSAR, and licensee procedures to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements.

This inspection constituted one sample.

b. Findings

Failure to Take Adequate Corrective Actions to Address a Degraded Condition with the Single Failure Proof Crane

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the license's failure to implement prompt corrective actions for the degraded conditions initially identified with the single failure proof PAB crane by maintenance personnel on January 17, 2008. As a result, on March 4, 2008, while lowering a new fuel storage canister in a laydown area after traversing the width of the spent fuel pool, the crane failed to the safe position with the load suspended approximately one foot off the floor. In a review of WO and corrective action history, the inspectors determined that all of the degraded conditions were not corrected during the maintenance activities conducted on February 21, 2008.

Description: On March 4, 2008, operators were performing new fuel receipt inspections for the upcoming Unit 2 refueling outage. Using the single failure proof PAB crane, operators had lifted a container with new fuel assemblies weighing several thousand pounds from the PAB truck bay, traversed over the spent fuel pool, and began to lower it in the new fuel receipt laydown area. While the operators were lowering the canister, the crane stopped working and failed in a safe position with the canister approximately one foot above the floor. The operators appropriately informed site management and an Emergency Response Team was formed. Site personnel placed cribbing under the canister and the crane was detached from the load to allow for troubleshooting.

On March 7, the licensee continued to troubleshoot and develop repair plans for the PAB crane. The inspectors reviewed condition reports and WOs to ascertain the material condition of the crane prior to this event. The inspectors determined that on January 17, maintenance personnel initiated CAP 01120105 following an inspection of the PAB crane and identification of issues associated with the auxiliary hoist comparator. Specifically, set screws had loosened on the coupling of the comparator, the shaft key had worked out of the keyway and had contacted and deformed the comparator bearing, and the keyway of the flexible cable coupling was deformed. The comparator was an integral feature that qualified the crane as single failure proof and was an augmented quality component. The comparator compared the motor speed of the hoist to the actual speed of the drum. When a large enough difference was detected between those two speeds, the PAB crane brake was actuated, causing the crane to lock in place, i.e., fail in the safe position. Pictures attached to the condition report documented the damage to the comparator.

The inspectors determined that condition report CAP 01120105 was closed to the creation of a Danger Tag and Work Request. On January 25, 2008, condition report CAP 01120610 was created by maintenance personnel to document that the PAB crane comparator cable was worn. The condition report also documented that the flex shaft internal to the assembly was excessively worn in the bearing area of the clutch/brake end of the PAB crane. Finally, the condition report documented that this could not be permanently repaired and required new parts.

The inspectors noted that WO 350916 was written to correct the conditions identified in condition reports CAP 01120105 and CAP 01120610. The work scope and purpose stated that the comparator couplings were damaged and could not be repaired due to excessive wear; however, the WO was later modified to include repair vice replacement. In addition, the inspectors noted that the WO did not repair the bearing associated with the comparator which was shown as being affected in condition report CAP 01120105. The inspectors determined that the failure to adequately correct the conditions originally identified in January 2008 resulted in the failure of the crane on March 4, 2008. Licensee personnel subsequently initiated a review of WOs and condition report history to determine if additional repairs were warranted.

The licensee completed repairs to the PAB crane identified by troubleshooting and performed the annual PAB crane inspection prior to returning the crane to service. In addition, the licensee developed and implemented additional post-maintenance testing to ensure functionality of the PAB crane.

Analysis: The inspectors determined that the licensee's failure to implement prompt corrective actions to address the degraded PAB crane, a condition adverse to quality originally identified in January 2008, was a performance deficiency requiring a significance evaluation. The inspectors concluded that the finding is more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 20, 2007, in that the finding could reasonably be viewed as a precursor to a significant event. Specifically, the failure to promptly correct the cause of the PAB crane degradation in a timely manner resulted in the crane being used to transport loads over the spent fuel pool in a degraded condition. This resulted in the crane failing to the safe position, with a load suspended over the new fuel cask laydown area, shortly after traversing the spent fuel pool.

The significance of this finding was evaluated using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," dated January 10, 2008, for the Barrier Integrity Cornerstone and spent fuel pool issues. The inspectors determined that the finding is of very low safety significance (Green), because it did not result in the loss of spent fuel pool cooling, did not result in damage to fuel clad integrity in the spent fuel pool and did not result in a loss of spent fuel pool inventory. The inspectors also determined that a primary cause of this finding is related to the cross-cutting area of problem identification and resolution. Specifically, the licensee failed to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity (P.1(d)).

Enforcement: Title 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality, such as malfunctions, deficiencies, deviations, defective equipment and non-conformances are promptly identified and corrected. Contrary to this, a condition adverse to quality, associated with the single failure proof, augmented quality PAB crane, was not promptly corrected following identification in January 2008. Specifically, upon identification of the degraded PAB crane comparator, a condition adverse to quality, the licensee did not take prompt actions to correct all the degraded conditions identified. As a result of the failure to take prompt corrective actions, the PAB crane failed, in the safe position, with a load suspended in the new fuel laydown area, shortly after traversing the spent fuel pool with a container of new fuel. Because of the very low safety significance of this finding and because it was entered into the licensee's corrective action program as CAP 01122858, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2008002-02; 05000301/2008002-02).

The licensee took immediate corrective actions to address the issue, which included repair of the PAB crane and a review of all outstanding condition reports and work orders associated with the crane. At the end of the inspection period the licensee continued to evaluate the causes associated with this finding.

.2 Additional Post-Maintenance Testing Activities

a. Inspection Scope

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

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

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

- Unit 2 P-11B CCW pump;
- P-32E SW pump; and
- Unit 1 P-29 turbine-driven auxiliary feedwater (TDAFW) pump valve 1MS-2082.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 January 2008 Unit 1 Forced Outage

a. Inspection Scope

The inspectors observed activities during the Unit 1 forced outage that occurred January 16 through February 5, 2008. This inspection consisted of an in-office review of the licensee's outage schedule, safe shutdown plan, and administrative procedures governing the outage; and periodic observations of equipment alignment and plant and control room outage activities. Specifically, the inspectors assessed the licensee's ability to effectively manage elements of shutdown risk pertaining to reactivity control, decay heat removal, inventory control, electrical power control, and containment integrity.

The inspectors conducted the following inspection activities:

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

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

- Verified that the flow paths, configurations, and alternative means for inventory addition were consistent with the outage risk plan;
- Observed operators align the residual heat removal system for shutdown cooling and verified the system was functioning properly to remove decay heat;
- Reviewed mode-change checklists to verify that selected requirements were met while transitioning from the refueling mode to full power operation; and
- Observed portions of the plant ascension to full power operations.

These inspection activities constituted one forced outage inspection sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

- Preconditioning occurred;
- Effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- Acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- Plant equipment calibration was correct, accurate, and properly documented; as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the FSAR, procedures, and applicable commitments;
- Measuring and test equipment calibration was current;
- Test equipment was used within the required range and accuracy;
- Applicable prerequisites described in the test procedures were satisfied;
- Test frequencies met TS requirements to demonstrate operability and reliability;
- Tests were performed in accordance with the test procedures and other applicable procedures;
- Jumpers and lifted leads were controlled and restored where used;
- Test data and results were accurate, complete, within limits, and valid;
- Test equipment was removed after testing;
- Where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers (ASME) 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 included one routine surveillance and three inservice tests, for a total of four quarterly inspection procedure samples:

- Reactor protection and safeguards logic testing - yellow channel;
- IT-07B P32-B SW pump quarterly;
- IT-08A Unit 1 quarterly TDAFW cold start and valve test on March 22 and 23, 2008; and
- IT-09A Unit 2 quarterly TDAFW cold start and valve test on March 23.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

.1 Data Submission Validation

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the fourth quarter 2007 PIs for any obvious inconsistencies prior to its public release in accordance with IMC 0608, “Performance Indicator Program.”

This review was performed as part of the inspectors’ normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours PI for Units 1 and 2 for the first quarter 2007 through the fourth quarter 2007. To determine the accuracy of the PI data reported during those periods, the inspectors used PI definitions and guidance contained in Revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment Performance Indicator Guideline.” The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports, and NRC integrated inspection reports for this period to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Attachment to this report.

This inspection constitutes two samples of the Unplanned Scrams per 7000 Critical Hours PI.

b. Findings

No findings of significance were identified.

.3 Unplanned Power Changes per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Power Changes per 7000 Critical Hours PI for Units 1 and 2 for first quarter 2007 through the fourth quarter 2007. To determine the accuracy of the PI data reported during those periods, the inspectors used PI definitions and guidance contained in Revision 5 of NEI 99-02. The inspectors reviewed the licensee's operator narrative logs, issue reports, maintenance rule records, event reports, and NRC integrated inspection reports for this period to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Attachment to this report.

This inspection constitutes two samples of the Unplanned Power Changes per 7000 Critical Hours PI.

b. Findings

No findings of significance were identified.

.4 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications PI for Units 1 and 2 for the first quarter 2007 through the fourth quarter 2007. To determine the accuracy of the PI data reported during those periods, the inspectors used PI definitions and guidance contained in Revision 5 of NEI 99-02. The inspectors reviewed the licensee's operator narrative logs, issue reports, maintenance rule records, event reports, and NRC integrated inspection reports for this period to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Attachment to this report.

This inspection constitutes two samples of the Unplanned Scrams with Complications.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Resident Inspector Review

a. Inspection Scope

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

c. Findings

No findings of significance were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 Inadvertent Draining of a Unit 1 Safety Injection (SI) Accumulator Resulting in TS Inoperability

a. Inspection Scope

Through record reviews and discussion with plant staff, the inspectors assessed the circumstances of an unplanned TS entry caused by the inadvertent excessive draining of an SI accumulator on January 4, 2008. The licensee took immediate corrective actions to restore the SI accumulator to operable status and evaluate the causes of the event. The inspection scope included a review of the control room logs and events leading to the inadvertent inoperability of the SI accumulator.

This inspection constitutes one event Follow-up sample.

b. Findings

Introduction: A self-revealed finding of very low safety significance (Green) and an associated NCV of TS 5.4.1, "Procedures," was identified for the failure to implement procedures associated with the conduct of operations for operation of plant systems. Specifically, on January 4, 2008, control room operators responded to a Unit 1 'A' SI Accumulator Level High Alarm and initiated actions to drain the accumulator in response to the alarm, without utilizing the redundant or backup indication for the draining evolution. This resulted in the inadvertent draining and inoperability of the 'A' SI accumulator with respect to the minimum TS required accumulator pressure.

Description: On January 4, 2008, control operators received a Unit 1 'A' SI Accumulator Level High Alarm. The operators noted that accumulator level was near the high level alarm setpoint, and based on previous experience with the alarm actuating early, the decision was made to drain the accumulator to clear the alarm. The operators referenced the alarm response book and initiated a work request. The operators entered the alarm into the control room log and documented the following: "Level determined to be high. Level indication on 1LI-938 was 38.5%, 1LI-939A was 37.5% and 1LI-939 (CO1-Rear) was 38.5%. Alarm setpoint from the setpoint document is 40%." There are three indications of accumulator level in the control room: two independent channels on the front panel (1LI938 and 1LI-939A) and a redundant indication of the 1LI-939 channel behind the control panel used for accumulator draining activities.

During the performance of Operating Instruction OI-100, "Adjustment of SI Accumulator Level and Pressure," to drain water from the Unit 1 'A' SI accumulator, the control operators received an unexpected Accumulator Low Pressure Alarm. The operators performing the drain activity behind the control panels immediately secured the drain down; however, accumulator pressure had dropped to 710 pounds per square inch gauge (psig). The TS parametric value for minimum accumulator pressure was 720 psig. Approximately 45 minutes later, operators restored the 'A' SI accumulator pressure to greater than 720 psig. Maintenance personnel subsequently determined that the 1LT-939 level transmitter had failed "as-is."

The licensee's subsequent investigation revealed that during the draining evolution, the control operator stationed at the front panel with the two independent level indicators was not monitoring those indications. Because of this, the accumulator was inadvertently drained to the point that the minimum required pressure was not maintained. In addition, the licensee determined that the crew did have an auxiliary operator monitoring reactor coolant drain tank level, which would be an indication of excessive draining of the accumulator. However, the auxiliary operator was not given any specific instruction by control room operators with respect to an expected level change for the draining of the accumulator. The licensee concluded that both the control operator not monitoring the indications during the draining evolution and the auxiliary operator not being properly informed to be effective in the field, illustrated a loss of control for the evolution by the senior reactor operator. In addition, the licensee concluded that the operators involved did not utilize all the available human performance tools, which also would have precluded this event.

Analysis: The inspectors determined that the failure to implement operating procedures, which resulted in the inadvertent inoperability of the Unit 1 'A' SI accumulator, while draining, was a performance deficiency requiring a significance evaluation. The finding is more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 20, 2007, because it is associated with the human performance attributes of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The significance of this finding was evaluated using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," dated January 10, 2008, for the Mitigating Systems Cornerstone. The inspectors determined that the finding is of very low safety significance (Green) because the finding did not involve a design or qualification deficiency, there was no actual loss of safety function, no single train loss of safety function for greater than the TS allowed outage time, and no risk due to external events. The inspectors also determined that the finding had a cross-cutting aspect in the area of human performance. Specifically, human error prevention techniques were not utilized following the receipt of the accumulator level alarm and during the drain down evolution (H.4(a)).

Enforcement: Technical Specification 5.4.1, "Procedures," required, in part, that written procedures be implemented for normal sequences of operation of components, systems and the overall plant. Nuclear Procedure, NP 2.1.1, "Conduct of Operations," Section 4.1.3, stated, in part, that when unusual or unexpected indications occur, operators will check any redundant or backup indications available to validate instrument response. Contrary to this, on January 4, 2008, control room operators responded to a Unit 1 'A' SI Accumulator Level High Alarm and initiated actions to drain the accumulator in response to the alarm, without utilizing the redundant or backup indication of accumulator level for the draining evolution. This resulted in the inadvertent excessive draining and inoperability of the 'A' SI accumulator with respect to the minimum TS required accumulator pressure. The only level accumulator channel the operators used to drain the accumulator had actually failed to the "as-is" position, causing the initial alarm. Because of the very low safety significance of this finding and because the finding was entered into the licensee's corrective action program as CAP 01119180, the violation is being treated as an NCV, consistent with Section VI.A.1 of NRC Enforcement Policy (NCV 05000266/2008002-03).

The licensee took immediate corrective actions which included restoration of the Unit 1 SI accumulator to an operable status, repair of the level indicator, and establishment of a new conduct of operations procedure. In addition, the licensee completed an apparent cause evaluation and developed additional corrective actions to correct this performance deficiency.

.2 Unit 1 X-04 Transformer Lock-out and Concurrent Loss of Offsite Power to Bus 1B-04

a. Inspection Scope

The inspectors evaluated the licensee's response to a Notification of Unusual Event which was declared on January 15, 2008. That event led to a Unit 1 forced outage that began on January 16, and continued through February 5. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing troubleshooting and recovery activities.

The outage was required by TS when the Unit 1 X-04 transformer could not be returned to service within the 24-hour TS allowed outage time. The inspectors observed or reviewed the reactor shutdown and cool-down, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, and startup and heatup activities. The NRC also conducted a special inspection at Point Beach that began on January 18, to monitor and assess the licensee's response to the 1X-04 transformer cable failures and the concurrent loss of safety bus 1B-04, and its identification and resolution of problems associated with the cause of the forced outage.

This inspection constitutes one event Follow-up sample.

b. Findings

The findings associated with this event were documented in NRC Special Inspection Report 05000266/2008007; 05000301/2008007.

.3 Unit 2 Heater Drain Tank Pump ‘B’ Trip Rapid Downpower

a. Inspection Scope

The inspectors evaluated the licensee’s response to the tripping of the ‘B’ Unit 2 heater drain tank pump on March 18, 2008, which caused a feedwater perturbation and resulted in the performance of a rapid downpower of Unit 2. The inspectors evaluated the operators’ use of appropriate annunciator response book instructions and abnormal operating procedures. The licensee’s troubleshooting efforts were also observed from the control room, as well as in the field. Troubleshooting identified the cause of the pump trip as the result of momentary inadvertent contact made to the heater drain tank sensing column by a scaffold section that was being built at the time. Once the area around the heater drain tank was secured, and the operators stabilized the unit, the licensee returned to full power shortly thereafter.

This inspection constitutes one event Follow-up sample.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000266/2004004-06; 05000301/2004004-06: Additional Information Needed to Determine Adequacy of Piping Anchor Design for Service Water

a. Inspection Scope

In following up on URI 05000266/2004004-06; 05000301/2004004-06, the inspectors reviewed the licensee’s (currently FPL Energy Point Beach, LLC) evaluation and corrective actions associated with the adequacy of design analyses for SW system pipe anchors. These design analyses had been completed by the former licensees (Wisconsin Electric Power Company and Nuclear Management Company, LLC) as part of their resolution to NRC Generic Letter (GL) 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions.” The inspection consisted of review of documentation related to the licensee’s corrective actions, correspondence related to GL 96-06, design and licensing basis requirements for SW piping and pipe supports, and SW system piping and pipe support design calculations. The inspectors reviewed the documentation to ensure that the SW system piping and pipe anchor designs were consistent with the design and licensing bases specified for GL 96-06 resolution.

b. Findings

Piping Anchor Design Not in Conformance with Design Basis Code Requirements

Introduction: A finding of very low safety significance (Green) was identified by the inspectors for the licensee's failure to evaluate the SW piping to pipe anchor integral welded attachments (IWAs) in conformance with the design requirements of the design basis ASME Boiler and Pressure Vessel Code. The licensee entered this issue into its corrective action program as CAP 01124284.

Description: During the 2004 Safety System Design and Performance Capability Inspection (Inspection Report 05000266/2004004(DRS); 05000301/2004004(DRS) dated September 7, 2004), the inspectors identified a URI concerning pipe anchors that were not evaluated in detail to demonstrate compliance with the design basis code associated with SW supply and return subsystems for the primary containment fan coolers (CFCs). Specifically, the design calculations for Unit 2 SW pipe anchor HB-19-A2 and SW pipe anchor HB-19-A4 qualified the pipe anchor IWA to SW piping using engineering judgment; the design calculations considered the structural capacity of the pipe anchor's 14-inch pipe cap component to be equal to or greater than that for the 8-inch SW piping. The design calculations indicated a full penetration weld attached the SW piping to the pipe cap anchor component. Since the piping stress was determined to meet design code acceptance criteria, the design calculations concluded the IWA, using a full penetration weld to attach the SW piping to the pipe cap anchor component, was qualified by comparison. The inspectors further reviewed the ASME Code to determine code jurisdictional boundaries between piping and pipe supports, design requirements, and acceptance criteria related to IWA pipe supports. By reviewing the load path of the piping reactions through the pipe cap anchor component, the inspectors determined that the engineering judgment used in the design calculations to qualify the pipe cap component and IWA to the SW piping was not appropriate. Specifically, the resultant piping stress at the IWA needed to be determined using all piping reaction forces and bending moments, not just the piping reaction moments used to calculate piping stress in accordance with the ASME Code for piping not at an IWA location. Also, the inspectors determined that piping reaction forces would cause bending stress in the pipe cap anchor components. Therefore, the inspectors determined that the pipe cap anchor components may not have greater structural capacity than the piping. Based on the magnitude of the piping reaction forces and moments specified for Unit 2 SW anchors HB-19-A1, HB-19-A2, HB-19-A3, and HB-19-A4, the inspectors could not verify design code compliance without a detailed evaluation of the anchor's IWA and pipe cap structural component. The licensee entered this issue into its corrective action program as condition report CAP 057947. The licensee identified that this concern was also applicable to similar Unit 1 SW pipe anchor designs.

As a result of its evaluation of this issue, the licensee performed additional analysis, calculation M-11165-035-SW.1, "Evaluation of Cap Anchors for SW Containment Fan Cooler Piping," in March 2005. This calculation evaluated the piping stresses, including the effect of the IWA of the SW piping to the pipe cap anchor component, in accordance with Section III, Division 1, Subsection NC of the ASME Code, 1977 Edition up to and including 1978 Winter Addenda (reconciled to the original design basis code of record, United States of America Standard B31.1-1967, "Power Piping"). Use of this ASME Code was consistent with previous licensee SW subsystem piping design calculations performed as part of GL 96-06 resolution. (Enclosure 1 to licensee submittal dated

February 27, 2004, “Response to Request for Additional Information Regarding Generic Letter 96-06,” referenced specific design analyses, including calculation Accession No. WE-20093, Revision 01, performed to evaluate associated pipe support modifications that have been installed as a result of GL 96-06. Calculation WE-20093, Revision 01, qualified the associated piping subsystem to the above ASME Code Edition and Addenda and evaluated the design of the IWA at SW anchor HB-19-A2 using the inappropriate engineering judgment discussed above.)

The effect of the IWAs on the piping stress was evaluated in calculation M-11165-035-SW.1 utilizing finite element stress analysis of the existing SW pipe anchor designs in conjunction with the acceptance criteria specified in ASME Code Case N-318-5, “Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping.” However, calculation M-11165-035-SW.1 applied a factor of 2/3 to piping reaction force and moment load combinations evaluated in the associated finite element analyses that determined the effect of the IWAs. Calculation M-11165-035-SW.1 justification for utilizing this reduction in the applied piping reaction forces and moments was, “the local membrane plus bending stress in the shell from the ANSYS results is the C index stress, which can be converted to the Code B index stress for use in the Code Equation 9 stresses as specified in the Code Case N-318-5.”

The inspectors determined that the licensee had misapplied the code case. Code Case N-318-5 was categorized as acceptable to the NRC for application in the design and construction of components and their supports for water-cooled nuclear power plants in Regulatory Guide 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.” However, the inspectors determined that the existing SW anchor IWA designs were outside of ASME Code Case N-318-5 Section 1.0, “Limitations to Applicability.” Specifically, this code case was designed to evaluate a relatively small rectangular cross section welded to the piping, not a pipe cap welded along the entire piping circumference which violated the attachment dimension limitations specified in Code Case N-318-5 Paragraph 1.3. Therefore, this code case method of analysis to determine the effect of an IWA, including related piping stress indices, was not applicable to the associated SW piping to pipe anchor IWAs and did not justify the application of a factor of 2/3 to piping reaction force and moment load combinations.

The inspectors determined the licensee had not correctly conformed to the requirements of ASME Code Subarticle NC-3600, “Piping Design,” the prescribed design basis for this calculation. Paragraph NC-3611.3 (2004 Edition), “Alternative Analysis Methods,” allowed for a more rigorous analysis as described in NB-3200, “Design by Analysis,” than the simplified engineering approach used to calculate stresses required to satisfy NC-3600. Within the Design Rules of NB-3200, the inspectors did not identify justification for the application of the factor of 2/3 to piping reaction force and moment load combinations when more rigorous finite element analysis is utilized to calculate stress. Therefore, the inspectors determined that the licensee’s design methodology, as utilized to evaluate the effect of the associated IWAs in calculation M-11165-035-SW.1, was not in conformance with the design requirements of the design basis ASME Code.

The inspectors determined that if this factor of 2/3 had not been applied in calculation M-11165-035-SW.1 finite element analysis models, the resulting maximum pipe/cap stress calculated for finite element analysis Model A would have exceeded the allowable Service Level C design stress limits for piping components specified in NC-3654.2(a) of the ASME Code (2001 Edition through and including 2003 Addenda). Furthermore, the

Service Level C design stress limits of NC-3654.2(a) were identical to the IWA Service Level C design stress limits specified in ASME Code Case N-318-5 and calculation M-11165-035-SW.1. The inspectors evaluated this issue using guidance from NRC Inspection Manual, Part 9900: Technical Guidance, “Operability Determinations and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety,” which is included as an Attachment to NRC Regulatory Issue Summary 2005-20: Revision to Guidance Formerly Contained in Generic Letter 91-18, “Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability.” Section C.10, “Piping and Pipe Support Requirements,” of Part 9900 Technical Guidance specified criteria in Section III, Appendix F, “Rules for Evaluation of Service Loadings with Service Level D Service Limits,” of the ASME Code for operability determinations. After increasing the maximum stress determined in calculation M-11165-035-SW.1 for finite element analysis Model A by a conservative factor of 3/2, the inspectors determined for the SW pipe anchors associated with Model A that the maximum calculated pipe/cap stress was less than Service Level D stress limits of NC-3655(a)(2), as allowed by Appendix F Paragraph F-1430. Therefore, the inspectors determined that the SW piping and pipe anchors associated with Model A were operable but nonconforming to the ASME Code requirements, and that the SW system was functionally capable of supplying cooling water to the CFCs during accident conditions.

The inspectors identified additional technical concerns related to design calculations M-11165-035-SW.1 and S-11165-035-SW.1, “Structural Evaluation of Pipe Anchor for CFCs 2-HX15 (‘A’, ‘B’, ‘C’ and ‘D’) and 1-HX15 (‘A’, ‘B’ and ‘D’) Piping,” that required further clarification and were potentially non-conservative. These additional concerns were not resolved by the licensee at the conclusion of the inspection. However, these concerns were entered into the licensee’s corrective action program, Activity Ratio (AR) 01124284, for review and final disposition.

Analysis: The inspectors determined that the licensee’s original engineering judgment used to demonstrate SW anchor design compliance with the design basis ASME Code was inappropriate and a performance deficiency requiring a significance evaluation. Furthermore, the application of a factor of 2/3 not specified in the ASME Code to piping reaction force and moment load combinations evaluated in the associated finite element analyses that calculated piping stresses was contrary to the design requirements of the design basis ASME Code and a performance deficiency.

The finding was determined to be more than minor because the finding was similar to IMC 0612, Appendix E, Example 3a because a revision to the design calculation with potential modification to the SW pipe anchors was necessary to demonstrate SW piping and pipe anchor compliance with the design basis ASME Code. Therefore, these performance deficiencies also impacted the Barrier Integrity Cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, this non-compliance with the ASME Code affected the Barrier Integrity Cornerstone design control attribute to maintain the structural integrity of the SW system, structures, and components and the operational capability of the CFCs.

The inspectors determined the finding, conservatively considering the effects of the outstanding technical concerns to result in a loss of CFC heat removal capability, could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings." In Table 2, under the Barriers Cornerstone column, the Containment Barrier Degraded box was checked based on Item 7 in Table 3b to enter the Containment Barrier Column in Table 4a. Question 3 in this column was answered "Yes" (postulated a loss of CFC heat removal) to enter Appendix H of IMC 0309, "Containment Integrity Significance Determination Process." Table 4.1 of Appendix H indicates that pressurized water reactor CFCs impact late containment failure and source terms, but not large early release frequency (LERF). Using Figure 4.1 of Appendix H for Type B LERF, the finding screened as Green.

The inspectors did not identify a cross-cutting aspect associated with this finding.

Enforcement: Title 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis, as defined in Section 50.2, are correctly translated into specifications, drawings, procedures, and instructions. Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.

Contrary to the above, in March 2005, the licensee failed to evaluate SW piping and pipe anchor designs in accordance with design requirements of the design basis ASME Code. Specifically, licensee calculations for Unit 2 SW pipe anchor HB-19-A2 and SW pipe anchor HB-19-A4 and other similar SW pipe anchors qualified the pipe anchor IWA to SW piping using inappropriate engineering judgment, and calculation M-11165-035-SW.1 applied a factor of 2/3 not specified in the ASME Code to piping reaction force and moment load combinations evaluated in the associated finite element analyses that calculated piping stress. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CAP 01124284, this violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2008002-04; 05000301/2008002-04).

Based on the above discussion, URI 5000266/2004004-06; 05000301/2004004-06 is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 9, 2008, the inspectors presented the inspection results to Mr. James McCarthy and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meeting

An interim exit meeting was conducted for:

- The closure of Safety System Design Performance Capability Inspection URI 5000266/2004004-06; 05000301/2004004-06 as an NCV with Regulatory Affairs Supervisor, Ms. F. Flentje , and other licensee staff via telephone on March 27, 2008. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Amundson, General Supervisor, Operations Supervisor
J. Bjorseth, Plant Manager
F. Flentje, Regulatory Affairs Supervisor
T. Gemskie, Emergency Preparedness Manager
R. Harrsch, Operations Manager
C. Jilek, Site Maintenance Rule Coordinator
D. Lowens, Nuclear Oversight Manager
J. McCarthy, Site Vice-President
S. Pfaff, Performance Assessment Supervisor
M. Ray, Regulatory Affairs Manager
C. Sizemore, Training Manager
T. Staskal, Regulatory Affairs Specialist
D. Tomaszewski, Site Engineering Director
B. Vandervelde, Maintenance Manager
P. Wild, Design Engineering
B. Woyak, Design Engineering

Nuclear Regulatory Commission

J. Cushing, Point Beach Project Manager, Office of Nuclear Reactor Regulations
D. Hills, Chief, Reactor Safety, Engineering Branch 1
M. Kunowski, Chief, Reactor Projects, Branch 5

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000266/2008002-01; 05000301/2008002-01	NCV	Failure to Take Prompt Corrective Actions for Recurring Cold Weather Issues (Section 1R01.1)
05000266/2008002-02; 05000301/2008002-02	NCV	Failure to Take Prompt Corrective Actions for Conditions Adverse to Quality Associated with the PAB Crane (Section 1R19.1)
05000266/2008002-03	NCV	Failure to Follow Procedures Resulted in Inadvertent Draining of Unit 1 SI Accumulator (Section 4OA3.1)
05000266/2008002-04 05000301/2008002-04	NCV	Piping Anchor Design Not in Conformance with Design Basis Code Requirements (Section 4OA5.1)

Closed

05000266/2004004-06 05000301/2004004-06	URI	Additional Information Needed to Determine Adequacy of Piping Anchor Design for Service Water (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

- CAP 01120522; Radwaste Vent Stack Icing Issues
- CAP 01120019; RS-SA-4 Radwaste Steam Relief Valve Lifted

1R04 Equipment Alignment

- 1-SOP-19KV-001; 1X-01 Main Power Transformer Backfeed Operation; Revisions 1 and 2
- 1-SOP-4KV-A01; Unit 1 Non Vital Train A 4160V Bus; Revision 3
- 1-SOP-4KV-A02; Unit 1 Non Vital Train B 4160V Bus; Revision 3
- 1-SOP-4KV-A05; Unit 1 Vital Train A 4160V Bus; Revision 2
- 1-SOP-4KV-A06; Unit 1 Vital Train B 4160V Bus; Revision 3
- 0-SOP-13.8KV-H01; H-01,13.8KV Bus; Revision 1
- 1-SOP-480-001; 480V System Normal Operations; Revision 11
- CL 11A G-03; G-03 Diesel Generator Checklist; Revision 6
- 2-CL-CC-001; Component Cooling Unit 2; Revision 11

1R05 Fire Protection

- Fire Hazards Analysis Report; January 2007 Revision
- NP 1.9.9; Transient Combustible Control

1R11 Licensed Operator Requalification Program

- PB-LOR-08A-001E; Point Beach Licensed Operator Requalification Simulator Guide for Cycle 08A

1R12 Maintenance Rule Implementation

- PBNP Maintenance Rule Unavailability Data Sheets for CCW from January 1, 2006 to January 1, 2008
- Performance Criteria assessments for CCW from January 1, 2006 to January 1, 2008
- Documentation of Maintenance Rule Performance Criteria for CCW, dated January 23, 2006
- Function List for CCW, dated February 26, 2008
- AR Search for CCW from January 1, 2006 to January 1, 2008
- WO Package 00341465; 2P-11B Grease Coupling, dated January 30, 2008
- WO Package 00341464; 2P-11B Change Oil, Flush Bearings and Clean Air Intake Grills, dated January 30, 2008

1R13 Maintenance Risk Assessments and Emergent Work Control

- NP 10.3.6; Shutdown Safety Review and Safety Assessment; Revision 19
- Safety Monitor Calculation Reports for Units 1 and 2 for Applicable Work Weeks

- Work Week Execution Schedules for the Applicable Work Weeks
- Operator Logs for the Applicable Work Weeks

1R15 Operability Evaluations

- Generic Letter 79-36, "Adequacy of Station Electric Distribution System Voltages," and associated licensee responses
- NRC Safety Evaluation of the Preferred Power Systems Conformance to General Design Criteria 17; February 1983
- OPR 01118180; PBNP Calculation 2005-0056 indicates that ambient temperatures in the SW Pump area could exceed 104 degrees following a Loss of Offsite Power (LOOP) without operator intervention if the outside air temperature were >95 degrees Fahrenheit; Revision 2
- PBNP Design Basis Document, Module A, Revision 5
- CAP 01121900; NRC Concerns with Additional Info for CAP 01118180
- CAP 01118180; Circulating Water Pump House Temperatures Issues
- Fire Area Analysis Summary Report for Fire Area A38: Circulating Water Service Water Pump House; dated August 8, 2005
- Circulating Water Service Water Pump House Cold Weather Calculation and Evaluation Associated With CAP 01118180; dated December 12, 2007
- CAP 00896611; Non-Conservative Protection System Setpoint Allowable Values
- OPR 00896611; Non-Conservative Protection System Setpoint Allowable Values
- CAP 00896436; Protection System Setpoint Methodology Changes
- CAP 01121068; Service Water Pump P-032E Higher Than Normal Vibration
- CAP 01124170; At Risk 50.59 Screening Inadequate for Engineering Change EC11938

1R18 Plant Modifications

- EC11877; Install Temporary Power Cables from 1X-04 to 1A-03 and 1A-04

1R19 Post-Maintenance Testing

- CAP 01120610; Z-15 Crane Flex (X-SAM) Cable Worn
- CAP 01120105; Z-15 Crane Auxiliary Hoist Speed Comparator
- WO 350916; Primary Auxiliary Building Crane; February 21, 2008
- CAP 01123605; PAB Crane Event Response Team Documentation and Critique
- CAP 01122858; PAB Crane has Stopped Working with a New Fuel Canister Suspended
- WO 348627; PAB Crane Repairs
- WO 352097; PAB Crane Repairs
- WO Package 00341465; 2P-11B Grease Coupling; dated January 30, 2008
- WO Package 00341464; 2P-11B Change Oil, Flush Bearings and Clean Air Intake Grills; dated January 30, 2008
- NP 10.2.7; Post-Maintenance/Return to Service Testing; Revision 7
- CCW Vendor Manual: Ingersoll-Rand, Installation-Operation and Maintenance of "S" Line General Service Pumps; Revision 32
- CAP 01122383; 2P-11B Coupling Cover Shows Wear Marks from Grid
- CAP 01122372; 2P-11B Has No Oil Level Sightglass for I.B. Bearing
- IT-08A; Cold Start of Turbine Driven Auxiliary Feed Pump and Valve Test; Revision 49; completed March 27, 2008

1R20 Forced Outage

- Boric Acid Leakage and Corrosion Monitoring Program; Revision 4
- NP 7.4.14; Boric Acid Leakage and Corrosion Monitoring
- Licensee Response to Generic Letter 88-05; dated May 24, 1988
- CL-4D; Outage Valve Inspection Unit 1
- OP 3A; Power Operation to hot Standby Unit 1; Revision 0
- OP 3B; Reactor Shutdown; Revision 39
- OP 1B; Reactor Startup; Revision 56
- OP 1C; Startup to Power Operation Unit 1
- OP 2A; Normal Power Operation; Revision 61

1R22 Surveillance Testing

- IT-08A; Cold Start of Turbine Driven Auxiliary Feed Pump and Valve Test; Revision 49; completed March 16, 2008
- NP 7.4.4; ASME OM Code Pump and Valve Inservice Testing
- CAP 01099536; Preconditioning of MS-2082 TDAFW Trip and Throttle Valve
- IT-9A; Cold Start of Turbine Driven Auxiliary Feed Pump and Valve Test Unit 2; Revision 49; completed March 16, 2008
- IT-07B; P-32B Service Water Pump (Quarterly); Revision 20; completed February 27, 2008
- 2ICP 02.001YL; Reactor Protection and Engineered safety Features Yellow Channel Analog 92 Day Surveillance Test; Revision 12; completed February 27, 2008

4OA3 Follow-up of Events and Notices of Enforcement Discretion

- AOP 2B; Feedwater System Malfunction; Revision 15

4OA5 Other Activities

- NP 2.1.1; Conduct of Operations; Revision 3
- OI-100; Adjusting SI Accumulator Level and Pressure
- CAP 01119180; Inadvertent Draining of SI Accumulator and associated Apparent Cause Evaluation
- CAP 011191177; Received C01B Accumulator Level High or Low
- CAP 01119592; OI 100
- CAP 01119997; Unable to Open 1CV-371A Letdown Containment Isolation Valve

Closure of URI 05000266/2004004-06; 05000301/2004004-06

Calculations

- M-11165-035-SW.1; Evaluation of Cap Anchors for SW Containment Fan Cooler Piping; Revision 0
- M-11165-035-SW.2; SW Cap Anchor Load Evaluation; Revision 0
- S-11165-035-SW.1; Structural Evaluation of Pipe Anchor for Containment Fan Coolers 2-HX15 (‘A’, ‘B’, ‘C’ & ‘D’) and 1-HX15 (‘A’, ‘B’ & ‘D’) Piping; Revision 2

Corrective Action Documents Reviewed

- CAP 057947; Questionable Qualification of Service Water Anchor HB-19-A2; dated July 15, 2004
- CA058528; Verify and/or Revised Affected Calculations - Service Water Anchor HB-19-A2; dated July 19, 2004
- CA061992; Follow on Actions to CA058528; dated March 11, 2005

GL 96-06 Related Correspondence

- WEPCO Letter to NRC; Subject: Summary of Actions to Ensure Continued Operability of the Service Water System; dated July 23, 1996
- Licensee Event Report No. 96-005-00; Potential Service Water Flashing in Containment Fan Coolers, Point Beach Nuclear Plant, Units 1 and 2; report date August 30, 1996
- WEPCO Letter to NRC; Subject: Detailed Operability Evaluation of the Service Water System with Respect to Post-Accident Boiling in Containment Fan Coolers; dated September 9, 1996
- WEPCO Letter to NRC; Subject: Evaluation of Steady State Service Water System Hydraulic Characteristics during a Design Basis Accident; dated September 30, 1996
- WEPCO Letter to NRC; Subject: Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions; dated October 30, 1996
- WEPCO Letter to NRC; Subject: GL 96-06, 120-Day Response, Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions; dated January 28, 1997
- WEPCO Letter to NRC; Subject: Revision to GL 96-06, 120-Day Response and Supplement to Technical Specifications Change Request 192; dated June 25, 1997
- WEPCO Letter to NRC; Subject: Information Pertaining to Implementation of Modifications Associated with Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions”; dated December 18, 1997
- WEPCO Letter to NRC; Subject: Reply to Request for Information to Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions”; dated September 4, 1998
- NMC Letter to NRC; Subject: Reply to Request for Information to Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions”; dated October 12, 2000
- NMC Letter to NRC; Subject: Electric Power Research Institute Report TR-113594, “Resolution of Generic Letter 96-06 Waterhammer Issues,” Volumes 1 and 2; dated July 30, 2002
- NMC Letter to NRC; Subject: Supplement to Generic Letter 96-06 Resolution; dated March 27, 2003
- NMC Letter to NRC; Subject: Response to Request for Additional Information Regarding Generic Letter 96-06, PBNP Units 1 and 2; dated November 3, 2003
- NMC Letter to NRC; Subject: Response to Request for Additional Information Regarding Generic Letter 96-06, PBNP Units 1 and 2; dated February 27, 2004
- NRC Letter to EPRI Waterhammer Project Utility Advisory Group; Subject: NRC Acceptance of EPRI Report TR-113594, “Resolution of Generic Letter 96-06 Waterhammer Issues,” Volumes 1 and 2; dated April 3, 2002
- NRC Letter to NMC; Subject: Point Beach Nuclear Plant, Units 1 and 2 - Completion of Licensing Action for Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions”; dated October 5, 2004

Miscellaneous Documents

- Drawing BECH M-400; Sheet 9; Service Water Pipe Cap Anchors; Revision 0
- Drawing P-315, Sheet 23; Pipe Hanger/Support Detail, HB-19-A2A and HB-19-A4A, Unit 1; Revision 0
- Drawing P-315, Sheet 24; Pipe Hanger/Support Detail, HB-19-A1A and HB-19-A3A, Unit 1; Revision 1
- Drawing P-316; Sheet 6; Pipe Hanger/Support Detail, HB-19-A1 and HB-19-A3, Unit 1; Revision
- Drawing P-415, Sheet 9; Pipe Hanger/Support Detail, HB-19-A1 and HB-19-A2, Unit 2; Revision
- Drawing P-416, Sheet 6; Pipe Hanger/Support Detail, HB-19-A3, Unit 2; Revision 0
- Drawing P-438, Sheet 2; Pipe Hanger/Support Detail, HB-19-A2, Unit 2; Revision 0
- Drawing P-438, Sheet 12; Pipe Hanger/Support Detail, HB-19-A1 and HB-19-A4, Unit 2; Revision 0

Reference Documents

- NRC Generic Letter 96-06; Assurance of Equipment Operability and Containment integrity during Design-Basis Accident Conditions; dated September 30, 1996
- NRC Generic Letter 96-06, Supplement 1; Assurance of Equipment Operability and Containment integrity during Design-Basis Accident Conditions; dated November 13, 1997
- NRC Regulatory Guide 1.84; Design, Fabrication, and materials Code Case Acceptability, ASME Section III; Revision 34
- NRC Regulatory Issue Summary 2005-20; Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, “Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions on Operability”; dated September 26, 2006

NRC-Identified Condition Reports

- AR 01120530; Heater in U-2 Façade Tent not Performing
- AR 01120605; Procedure Step Missed during Training
- AR 01120318; Delay in Recognizing Effect on Operation
- AR 01120794; Operator Aids
- AR 01120196; Log Entries for Iodine Sampling
- AR 01121438; U1R30 Steam Generator Inspection Report
- AR 01122642; References in EPIP 1.2.1
- AR 01121012; Underground Electrical Cable Management
- AR 01123137; RW Relief Stack Drain Heat Lamp Disabled
- AR 01123881; Pre-Outage Work Schedule Grading
- AR 01119241; Concerns of PBNP’s Use of IST Trend Data
- AR 01124232; MRE for AR 01119997-02 not Identified
- AR 01121685; Potential Vulnerability of X04 Transformer
- AR 01121900; NRC Concerns with Additional Information
- AR 01124161; Unit 2 4th Quarter 2007 NRC Data Submitted
- AR 01122866; TRM 2.2 Discrepancy – EFPY Basis
- AR 01120637; AQ not Fully Defined
- AR 01119822; Tornado/High Wind AR’s
- AR 01123627; ACE 01121068 Rejected
- AR 01124051; Possible Incorrect PI Entry

- AR 01120870; Lack of Challenging Information
- AR 01120811; GL 88-05, Reactor Coolant System Leakage Commitments
- AR 01122951; Inadequate Assessment/Close of AR
- AR 01120237; Concern Regarding 50G Relay Operability
- AR 01121756; NRC Submittal Contained Interoffice E-mail
- AR 01121535; LAR 260 Request for Additional Information
- AR 01121346; System Mission Time Clarification
- AR 01124284; URI 2004-004-06
- AR 01121825; Input to OPR 01118180-02 not Supported
- AR 01119935; NRC GL 2008-01

LIST OF ACRONYMS USED

AR	Activity Ratio
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program Document (Condition Report)
CCW	Component Cooling Water
CFC	Containment Fan Cooler
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EC	Engineering Change
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
GL	Generic Letter
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
IST	Inservice Test
IWA	Integral Welded Attachment
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOOP	Loss of Off-site Power
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NMC	Nuclear Management Corporation
NRC	U.S. Nuclear Regulatory Commission
OPR	Operability Evaluation
PAB	Primary Auxiliary Building
PI	Performance Indicator
psig	Pounds Per Square Inch Gauge
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
SI	Safety Injection
SW	Service Water
TDAFW	Turbine-Driven Auxiliary Feedwater
TS	Technical Specification
URI	Unresolved Item
WO	Work Order