January 22, 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 EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000266/2007007(DRS); 05000301/2007007(DRS)

Dear Mr. McCarthy:

On December 14, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Point Beach Nuclear Plant, Units 1 and 2. The enclosed report documents the results of the inspection, which were discussed with Mr. J. McCarthy, and others of your staff at the completion of the inspection on December 14, 2007.

The inspectors examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Based on the results of the inspection, one NRC identified finding of very low safety significance was identified, which involved a violation of NRC requirements. However, because this violation was of very low safety significance and because it was entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation (NCV) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of an NCV, 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 Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: Inspection Report 05000266/2007007(DRS); 05000301/2007007(DRS)
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
State Liaison Officer

component of NRC's Agencywide Documents Access and Management 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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State Liaison Officer

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DATE	1/22/08		1/22/08					

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Inspection Report to Mr. J. McCarthy from Mr. D. E. Hills dated January 22, 2008

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000266/2007007(DRS); 05000301/2007007(DRS)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-266; 50-301
License Nos:	DPR-24; DPR-27
Report No:	05000266/2007007(DRS); 05000301/2007007(DRS)
Licensee:	FPL Energy Point Beach, LLC
Facility:	Point Beach Nuclear Plant, Units 1 and 2
Location:	Two Rivers, WI 54241
Dates:	November 26 – December 14, 2007
Inspectors:	Z. Falevits, Senior Reactor Inspector S. Sheldon, Senior Reactor Inspector J. Bozga, Reactor Inspector
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000266/2007007(DRS); 05000301/2007007(DRS); 11/26/2007 through 12/14/2007; Point Beach Nuclear Plant, Units 1 and 2; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59) and Permanent Plant Modifications.

The inspection covered a two week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three regional based engineering inspectors. One Green Non-Cited Violation (NCV) was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red), using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP)." Findings for which the SDP did not apply, may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3; dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance involving a calculation that designed the Unit 1 Steam Generator Blowdown (SGBD) Heat Exchanger (HX) Platform to withstand a load from a postulated pipe whip of the condensate return line resulting from a High-Energy Line Break (HELB). The load from a postulated pipe whip applied to the platform was evaluated in calculation PBNP-994-10-S01, "SGBD HX Platform Mod. For Addition of Pipe Rupture Restraint for Condensate Return Line" which was approved on April 28, 2007. As a result of this calculation, the design function of the Unit 1 SGBD HX Platform was revised to hold and maintain the steam generator blowdown heat exchangers and condensate return line in position and assure that the platform did not fall onto the safety related Refueling Water Storage Tank (RWST) during a safe shutdown earthquake and a HELB simultaneously. Specifically, the licensee failed to correctly use the original design anchor bolt safety factor in the supporting calculation. This issue was entered into the licensee's corrective action program as condition report CAP 1118144.

The issue was more than minor because the calculation error would be expected to necessitate extensive calculation rework and possibly a modification in order to demonstrate that the platform meets design acceptance limits commensurate with those applied to original design. The finding screened as having very low safety significance (Green) because the inspectors answered "yes" to question 1 under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, the platform remained "operable but degraded". The cause of the finding was related to the cross-cutting element in Human Performance, Work Practices because the licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported (item H.4(c) of IMC 0305). The licensee had failed to correctly use the original design anchor bolt safety factor in all three revisions of the design basis calculation. (Section 1R02.1.b.1)

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

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02, TI 2515/166)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From November 26, 2007 through December 14, 2007, the inspectors reviewed six safety evaluations (SEs) performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 12 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors is included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations, and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

b. Findings

b.1 Incorrect Factor of Safety Specified in Design Evaluation of Unit 1 SGBD HX Platform

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, the inspectors identified that the design bases calculation for the Unit 1 SGBD HX Platform failed to use the correct anchor bolt safety factor.

Description: The design function of the Unit 1 SGBD HX Platform from initial construction to 2006 was to hold and maintain the steam generator blowdown heat exchangers in position and assure that the platform did not fall onto the safety related Refueling Water Storage Tank (RWST) during a safe shutdown earthquake. The platform was a braced steel frame with the columns at each corner anchored to the floor at their base. The design function of the RWST was to provide borated water to the safety injection pumps, the residual heat exchangers pumps and the containment spray pumps during a loss-of-coolant accident or a steam line break accident. The RWST was safety related and designed as Seismic Class I. Screening 2007-0066 "Installation of

Pipe Whip Restraint per Mod EC 10533", applied to the "at risk" installation and the controls for acceptance and use of a whip restraint for the Unit 1 condensate return line piping. The whip restraint involved installation of steel beams on the existing SGBD HX platform. The whip restraint steel beams were supported from this platform and were attached to the north flange face of two columns. The platform was designed to withstand a load from a postulated pipe whip of the condensate return line resulting from a High-Energy Line Break (HELB). The load from a postulated pipe whip that the whip restraint adds to the platform was evaluated in calculation PBNP-994-10-S01, "SGBD HX Platform Mod. For Addition of Pipe Rupture Restraint for Condensate Return Line" which was approved on April 28, 2007. As a result of this calculation, the design function of the Unit 1 SGBD HX Platform was revised to hold and maintain the steam generator blowdown heat exchangers and condensate return line in position and assure that the platform did not fall onto the safety related Refueling Water Storage Tank (RWST) during a safe shutdown earthquake and a HELB simultaneously. The inspectors identified a non-conservative technical error in calculation PBNP-994-10-S01. The inspectors identified that the calculation evaluated acceptability of the anchor bolts based on a factor of safety of 2 under the faulted load condition. The acceptance criteria for anchor bolt design established in the design basis calculation was in accordance with Specification 10447-P500(Q) "Technical Specification for Inspection and Testing Concrete Expansion Bolts for Seismic Category 1 Pipe Supports for the Point Beach Generating Station." The requirement in the specification was that shell or wedge type anchor bolts had a factor of safety greater than or equal to four based on the ultimate bolt capacity. The licensee determined that the calculation PBNP-994-10-S01 specified the operability limit for the anchor bolt allowable instead of the design limit. The operability limit only required that shell or wedge type anchor bolts had a safety factor equal to two based on the ultimate bolt capacity. The use of the operability limit did not meet design requirements.

The inspectors also identified non-typical computer modeling used in the calculation and the as built design of the welded connection for attachment of the pipe whip restraint steel to the platform steel columns. Specifically, the inspectors identified that the whip restraint base plates were welded to the platform columns using a single line weld along one vertical edge of the baseplate. This condition was not reflected in the evaluation which models these attachments as simple pin connections without any eccentricity. The line of action of the loading was offset or eccentric from the center of gravity of the single line fillet weld connection. When the applied forces do not pass through the center of gravity of a weld configuration, a moment is created. The moment was equal to the force times the eccentric length. The computer model of this pin connection did not consider these induced moments. The offsets or eccentricities also produce potential additional torsional stresses on the existing column which were not considered in the calculation. The calculation therefore did not account for the effects of the connection eccentricity on the welds, the whip restraint steel and the platform column steel. In addition, the calculation did not provide the basis for treating welded connection as pin connection.

The licensee agreed that the connection was not typical of code design. In addition, the design basis calculation did not consider the effects of base plate flexibility and prying action in the calculation of anchor bolt loads. The baseplate was analyzed as a rigid plate but this assumption was never validated in the analysis. As a result, the inspectors could not be sure the modeling was adequate without the licensee performing confirmatory calculations that addressed these concerns. The licensee planned to revise this calculation to address the anchor bolt factor of safety error and as part of that effort to re-evaluate with respect to these other calculational concerns to verify whether the current design is adequate. This issue was entered in the licensee's corrective action program as CAP 1118200.

Analysis: The inspectors determined that the failure to use correct anchor bolt safety factor in the design was a performance deficiency and a finding. The mitigating systems cornerstone was affected as the failure of the Unit 1 SGBD HX Platform could result in the failure of the RWST to fulfill its design function. No other cornerstones were determined to be degraded as a result of this issue. The finding was determined to be greater than minor because the calculation error would be expected to necessitate extensive calculational rework and possibly a modification to ensure that the platform met design acceptance limits.

The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "yes" to question 1 under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. The inspectors agreed with the licensee's position that the platform was "operable but degraded." Therefore, the inspectors concluded that the finding did not represent an actual loss of safety function, and the issue screened out as having a very low safety significance or Green.

The cause of the finding is related to the cross-cutting element in Human Performance, Work Practices because the licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported (item H.4(c) of IMC 0305). Specifically, the licensee had failed to correctly use the original design anchor bolt safety factor in all three revisions of the design basis calculation. Each calculation revision was performed by a different contractor and failed to identify this non-conservative technical error.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, the performance of design reviews was inadequate, in that design calculation PBNP-994-10-S01, "SGBD HX Platform Mod. For Addition of Pipe Rupture Restraint for Condensate Return Line" did not demonstrate the Unit 1 SGBD HX Platform will maintain structural integrity and assure that the platform will not fall onto the safety related Refueling Water Storage Tank (RWST) during a safe shutdown earthquake and a HELB simultaneously. Because the violation was of very low safety significance and the licensee entered the violation into their corrective action system as condition report CAP 1118144, this violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000266/2007007-01(DRS); NCV 05000331/2007007-01(DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From November 26, 2007 through December 14, 2007, the inspectors reviewed six permanent plant modifications (six samples) that had been installed in the plant during the last two years. The modifications were chosen based upon risk significance, safety significance, and complexity. As per inspection procedure 71111.17B, one modification was chosen that affected the barrier integrity cornerstone. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements, and the licensing bases, and to confirm that the changes did not adversely affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Other Activities

.1 (Open) Temporary Instruction (TI) 2515/166, "Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02)"

Generic Letter (GL) 2004-02 requested that licensees to perform a mechanistic evaluation of the potential for the adverse effects of post-accident debris blockage and operation with debris-laden fluids to impede or prevent the recirculation functions of the Emergency Core Cooling (ECCS) systems and Containment Spray (CSS) systems following all postulated accidents for which the recirculation of these systems is required. The generic letter also requested that addressees implement any needed plant modifications. The purpose of the TI 2515/166 is to verify the licensee has implemented plant modifications and procedure changes committed to by the licensee in their GL 2004-02 responses. As discussed in Section 1R02 and 1R17 of this report, the inspectors reviewed the licensee's modifications and 10 CFR 50.59 safety evaluations.

Related documents were reviewed in IR 2006013 and IR 2007003. The inspectors did not identify any violations with NRC regulations and as required by the TI, addressed the questions below:

a. Did the licensee implement the plant modifications and procedure changes committed to in their GL 2004-02 responses?

The inspectors reviewed the licensee's responses and identified the following commitments in Nuclear Management Company, LLC (NMC) letter L-HU-05-004 dated March 7, 2005:

> NMC will perform latent debris sampling at Point Beach Nuclear Plant (PBNP), Unit 1, during the Fall 2005 refueling outage and at Unit 2 during the Spring 2005 refueling outage.

The licensee completed these commitments in October, and April 2005 respectively.

The inspectors reviewed the licensee's responses and identified the following commitments in NRC 2005-0109 dated September 1, 2005:

1. *NMC will evaluate and modify as appropriate the PBNP Unit 1 and Unit 2 Emergency Core Cooling (ECCS) systems to support long-term decay heat removal and resolve the issues identified in GL 2004-02 by December 31, 2007.*

   The inspectors determined that the licensee completed the sump modifications for Unit 1 and Unit 2 in April, 2007 and November, 2006 respectively. The licensee has requested to extend the final submittal date to June 30, 2008 in FPL Energy letter NRC 2007-0085.

2. *NMC will update the PBNP licensing basis to reflect the results of the analyses and modifications performed to demonstrate compliance with the regulatory requirements of GL 2004-02. This update will be performed in accordance with 10 CFR 50.71.*

   The inspectors verified that the June 2007 Updated Final Safety Analysis Report (UFSAR) has been updated to reflect the new sump strainer design and containment coating program requirements.

3. *NMC will establish administrative controls at PBNP to have proposed insulation changes inside containment reviewed and approved by engineering. This will ensure insulation upgrades, repairs, and replacements do not result in an unanalyzed debris mix or quantity. These controls will be established prior to the beginning of the spring 2007 (Unit 1) refueling outage.*

   The inspectors reviewed procedure NP 7.2.28, "Containment Debris Control Program" Revision 2 and verified that it implements this commitment.

4. *NMC will provide a separate submittal to update the responses to requests (d)(i) through (d)(iii), and (d)(v) through (d)(vii) within 60 days of acceptance of the final screen design. Acceptance of the final screen designs by the PBNP Plant Oversight Review Committee (PORC) is scheduled for February 15, 2006 (Unit 2) and June 22, 2006 (Unit 1).*

   The licensee updated the commitment date for this item to Dec 31, 2007 in NMC letters NRC 2006-0077 and NRC 2006-0092. This item had not been completed by the end of the inspection.

b. Has the licensee updated its licensing bases to reflect the corrective actions taken in response to GL 2004-02?

   The inspectors verified that the June 2007 Updated Final Safety Analysis Report (UFSAR) has been updated to reflect the new sump strainer design and containment coating program requirements. The licensee stated that any further changes associated with the final submittal will be included in the June 2008 UFSAR update.

c. If the licensee or plant has obtained an extension past the completion date of this TI, document what actions have been completed, what actions are outstanding, and close the TI for the plant that has the extension.

   The licensee has requested to extend the final submittal date to June 30, 2008 in FPL Energy letter NRC 2007-0085, dated November 16, 2007. This extension was to allow for developing, performing and documenting additional chemical effect testing.

   This TI is open pending further NRC review.

4OA2 <u>Identification and Resolution of Problems</u>

.1 <u>Routine Review of Condition Reports</u>

a. <u>Inspection Scope</u>

   From November 26 through December 14, 2007, the inspectors reviewed six Corrective Action Process (CAP) documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions (CA) related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. <u>Findings</u>

   No findings of significance were identified.

4OA6 <u>Meetings</u>

.1 Exit Meeting

The inspectors presented the inspection results to Mr. J. McCarthy and others of the licensee's staff, on December 14, 2007. Licensee personnel acknowledged the inspection results presented. Licensee personnel were asked to identify any documents, materials, or information provided during the inspection that were considered proprietary.

No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
B. Scherwinski, Regulatory Affairs
F. Flentje, Licensing Supervisor
K. Locke, Regulatory Assurance
L. Peterson, Design Engineer Manager
P. Wild, Design Engineering Projects Supervisor
B. Woyak, Design Engineering Supervisor
R. Chapman, Design Engineering

Nuclear Regulatory Commission
R. Ruiz, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000266/2007007-01 05000331/2007007-01	NCV	Incorrect Factor of Safety Specified in Design Evaluation of Unit 1 SGBD HX Platform

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list did not imply that 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 in this list did not imply NRC acceptance of the document, unless specifically stated in the inspection report.

IR02 Evaluation of Changes, Tests, or Experiments 71111.02

10 CFR 50.59 Evaluations

Evaluation 2005-002; MR 03-047 - Replacement of Unit 1 Reactor Vessel Closure Head; dated June 28, 2005

Evaluation 2006-007; 480V Electrical Load Conservation; dated May 30, 2006

Evaluation 2005-004; Clarification of FSAR 14.2.5 Rupture of a Steam Pipe; Containment Response Calculation; dated February 7, 2006

Evaluation 2007-001; Reduced Zero Power Main Feedwater Temperature; dated March 9, 2007

Evaluation 2007-003; Defeating Pzr Low Level Letdown and Heater Cutoff to Replace Bistables Online; dated July 5, 2007

Evaluation 2006-011; Revision to the FSAR to Reflect the New SI Pump Motor Acceleration Times and to Reference Westinghouse Letter WEP-05-029; dated January 18, 2007

10 CFR 50.59 Screenings

Screening No. 2007-0066; “Installation of Pipe Whip Restraint per Mod 10533”; dated April 20, 2007

Screening No. 2007-0017; “480V Switchgear Cable Protection”; dated March 26, 2007

Screening No. 2007-0010; “480V Switchgear Cable Protection”; dated March 26, 2007

Screening No. 2006-0248; “Evaluation of Polar Crane Bracket Compression Anchorage; dated December 11, 2006

Screening No. 2007-0016; “Evaluation of Scaffold in place over 90 days in Unit 1 Façade Near RWST (1T-013)”; dated January 27, 2007

Screening No. 2006-0191; “Use of Replacement Biach Stud Tensioners”; dated October 16, 2006

Screening 2007-0037; Changing Maximum Allowable Feedwater Flow for P-38A or P-38B, Motor Driven Auxiliary Pump; dated February 23, 2007

Screening 2006-0015; Change UV Testing Acceptance Criteria in 1/2RMP 9075-1 and 2; dated January 25, 2006

Screening 2005-0287-01; Revision to the FSAR to Reflect the New SI Pump Motor Acceleration Times and to Reference Westinghouse Letter WEP-05-209; dated January 9, 2007

Screening 2006-0088; 13.8 kV and 4.16 kV Protection and Coordination; dated August 7, 2006

Screening 2006-0086; Performance of Calculation 2005-0008 "Minimum Voltage Requirements for Safety Related Motor Control Center (MCC) Control Circuits"; dated June 17, 2006

Screening 2006-0087; 480V Switchgear Coordination and Protection; dated October 18, 2006

IR17 Permanent Plant Modifications 71111.17B

Modifications

EC 1608; Removal of 2SI-839A, B, C and D Test Line Isolation Valve; Revision 1

EC 7794; Revise Calculation PBNP-IC-34 Refueling Water Storage Tank Level Scaling Calc. (CRR); Revision 0

EC 8759; MCC 2B32 Feeder Cable Replacement; Revision 1

EC 9239; Unit 2 RCP 2P-2B Motor Rail and Sled System; Revision 0

EC 9240; RCP Transfer Bridge Lateral Seismic Evaluation; Revision 0

EC 1603; Install New ECCS Sump (Sump B) Screen – Unit 2; Revision 0, 1 and 2

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

CAP 01117062; SLP/RMP Inconsistencies; dated November 28, 2007

CAP 01117126; AFW Flow Indicators; dated November 28, 2007

CAP 01117163; Scaffold Stabilization Criteria; dated November 29, 2007

CAP 01117170; U2 RCP "B" Motor Sled Rubber Pads; dated November 28, 2007

CAP 01117637; Errors in calculations PCI-5344-S02 and PCI-5344-S03; dated December 10, 2007

CAP 01118002; Errors in calculation PCI-5344-S01; dated December 11, 2007

CAP 01118074; Low Design Margin in Calc. For MCC 2B32-1 Conduit Support; dated December 12, 2007

CAP 01118105; ACE 1044692 actions didn't identify potential 50.59 enhancement; dated December 13, 2007

CAP 01118144; Errors in structural calculation for U1 SGBD HX platform; dated December 13, 2007

CAP 01118148; Rigging Evaluation documentation; dated December 13, 2007

CAP 01118185; Evaluate Load Handling Procedure for Load Restrictions; dated December 14, 2007

CAP 01118194; Recommended Improvement to DG-M10; dated December 13, 2007

CAP 01118200; Support Model in Calculation PBNP-994-10-S01 in Question; dated December 14, 2007

CAP 01118202; Low Design Margin for Plant Component During 50.59/Mod Inspection; dated December 14, 2007

CAP 01118259; PBNP Ground Detection Threshold; dated December 18, 2007

<u>Corrective Action Program Documents Reviewed During the Inspection</u>

CAP 01045377; SCR 2005-0287 has an Incorrect Conclusion; dated August 19, 2006

CAP 01109352; EC Conclusions for Load Conservation were not Conservative; dated August 29, 2007

CAP 01022586; CST Level Indication Calibration Revision; dated October 13, 2006

CAP 01044692; CDBI-Inadequate 50.59 Process applied to TS Bases Change; dated August 15, 2006

CAP 01073222; Inadequate 50.59 Support for Calc. 2005-004; dated January 22, 2007

CAP 01092684; U1 and U2 Polar Crane Trolley Hard Stops; dated May 15, 2007

CAP 01103190; Scaffolding Standing over 90 Days; dated July 24, 2007

CAP 01089506; Mod EC 1569 did not Review Seismic Qualification of New Component; dated April 25, 2007

Calculations

Calc. PCI-5344-S02; Evaluation of Sump Cover and Piping for the Containment Sump Strainers; Revision 3

Calc. S-11165-075-001; Evaluation of Polar Crane Bracket Compression Anchorage; Revision 1

Calc 2006-0039; 480 V Switchgear Cable Protection; Revision 0

Calc PBNP-994-10-S01; SGBD HX Platform Mod. For Addition of Pipe Rupture Restraint for Condensate Return Line; Revision 1

Calc PBNP-994-01-S01; RCP Motor Cubicle Transfer System Bridge Seismic and Stress Analysis; Revision 0

Calc PBNP-994-01-S02; Lateral Seismic Evaluation of RCP Transfer Bridge Structure; Revision 0

Calc. 6704.001-C-067; Unique Conduit Supports; Minor Revision 2B and 2C

Drawings

Drawing SK-994-10-01; SGBD HX Condensate Return Rupture Restraint R-453; Revision 3

Procedures

NP 8.4.8; Requirements for Scaffold Near Safety Related Equipment; Revision 12

Miscellaneous Documents

EC 9229; Evaluation the Structural Adequacy of the Floor for Temporary Storage of the RCP Motor Transfer Bridge; Revision 0

EC 10489; Design Beam Clamp for Temporary Lead Shielding; Revision 0

EC 9363; Evaluate Rigging Beam for Removing Purge Valves VPSE-3213 and 3245; Revision 0

Specification 10447-P500(Q); Technical Specification for Inspection and Testing Concrete Expansion Bolts for Seismic Category 1 Pipe Supports for the Point Beach Generating Station; Revision 2

DG-M09; Design Requirements for Piping Stress Analysis; Revision 2

DG-M10; Pipe Support Guidelines; Revision 2

DG-E02; Point Beach Nuclear Plant Design and Installation Guidelines Seismic Conduit Support Manual; Revision 4

DG-C01; Guidelines for Design, Qualification and Installation of Concrete Expansion Anchors at Point Beach Nuclear Plant; Revision 0

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
CA	Corrective Action
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CSS	Containment Spray System
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
GL	Generic Letter
HELB	High Energy Line Break
HX	Heat Exchanger
IMC	Inspection Manual Chapter
IR	Inspection Report
NCV	Non Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PORC	Plant Oversight Review Committee
RWST	Refueling Water Storage Tank
SE	Safety Evaluation (50.59)
SGBD	Steam Generator Blowdown
TI	Temporary Instruction
UFSAR	Updated Final Safety Analysis Report