February 10, 2006

CAL 3-04-001

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

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 NRC INTEGRATED INSPECTION REPORT 05000266/2005013; 05000301/2005013

Dear Mr. Koehl:

On December 31, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Point Beach Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on January 19, 2006, with you and members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed your personnel.

Based on the results of this inspection, six findings of very low safety significance and a Severity Level IV violation were identified. These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these eight findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector at the Point Beach Nuclear Plant.

In addition to the routine NRC inspection and assessment activities, Point Beach performance is being evaluated quarterly as described in the Mid-Cycle Performance Review Letter - Point Beach Nuclear Plant, dated August 30, 2005. Consistent with Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," plants in the multiple/repetitive degraded cornerstone column of the Action Matrix are given consideration at each quarterly performance assessment review for (1) declaring plant performance to be unacceptable in accordance with the guidance in IMC 0305; (2) transferring to the IMC 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems," process; and (3) taking additional regulatory actions, as appropriate. During this inspection period, the NRC reviewed Point Beach operational performance, inspection findings, and performance indicators. Based on this review, we concluded that Point Beach is operating safely. We determined that no additional regulatory actions, beyond the already increased inspection activities and management oversight, are currently warranted.

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

Sincerely,

/RA/

Mark A. Satorius, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000266/2005013; 05000301/2005013
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degraded cornerstone column of the Action Matrix are given consideration at each quarterly performance assessment review for (1) declaring plant performance to be unacceptable in accordance with the guidance in IMC 0305; (2) transferring to the IMC 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems," process; and (3) taking additional regulatory actions, as appropriate. During this inspection period, the NRC reviewed Point Beach operational performance, inspection findings, and performance indicators. Based on this review, we concluded that Point Beach is operating safely. We determined that no additional regulatory actions, beyond the already increased inspection activities and management oversight, are currently warranted.

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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:	05000266/2005013; 05000301/2005013
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Plant, Units 1 and 2
Location:	Two Rivers, Wisconsin
Dates:	October 1, 2005, through December 31, 2005
Inspectors:	R. Krsek, Senior Resident Inspector G. Gibbs, Resident Inspector L. Haeg, Reactor Engineer M. Holmberg, Senior Reactor Inspector J. Jacobson, Senior Reactor Inspector J. Neurauter, Reactor Inspector T. Bilik, Reactor Inspector S. Orth, Health Physics Program Manager W. Slawinski, Senior Radiation Specialist M. Phalen, Radiation Specialist T. Ploski, Senior Emergency Preparedness Analyst M. Kunowski, Project Engineer
Approved by:	P. Louden, Chief Projects Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000266/2005013, 05000301/2005013; 10/01/2005 - 12/31/2005; Point Beach Nuclear Plant, Units 1 and 2; Post-Maintenance Testing, Refueling and Outage Activities, Identification and Resolution of Problems, Event Followup, and Other Activities.

This report covers a 3-month period of inspection by resident inspectors and announced inspection by regional specialists of inservice inspection, reactor vessel head replacement, operator licensing requalification examination, and radiation protection. A Severity Level IV non-cited violation (Green) and six Green findings with associated non-cited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process (SDP)." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance associated with the replacement of the 1P-10A residual heat removal pump (RHR) motor. This finding was also a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the failure to perform an equivalency evaluation for exceptions taken to motor specifications in the refurbishment of safety-related equipment. Specifically, the licensee failed to perform a technical evaluation for exceptions taken by the vendor to the licensee's motor specification for the 1P-10A RHR pump motor. Once identified, the licensee initiated a corrective action program document to perform an engineering evaluation before placing 1P-10A in service. The licensee also initiated an extent-of-condition review to ensure that other equipment was not subject to the same issue.

  This finding is greater than minor because it is associated with the design control 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. This finding was determined to be of very low safety significance because the 'A' RHR loop was not in operation and the 'B' train RHR loop was operable and in operation with support systems available, and because the finding did not: increase the likelihood of a loss of reactor coolant system (RCS) inventory, including a loss of RCS level instrumentation; degrade the licensee's ability to terminate a leak path or add RCS inventory when needed; or degrade the licensee's ability to recover decay heat removal once it was lost. Also, the finding is of very low safety significance because no event occurred that could be characterized as a loss of control of RCS temperature or inventory control. (Section 1R19)

- Green. The inspectors identified a finding associated with a non-cited violation of Technical Specification 5.4.1, “Procedures,” when the licensee failed, on two different occasions during the refueling outage, to perform adequate containment walkdowns to verify that no debris was present in the vicinity of the emergency core cooling system containment sump which could potentially impact operability.

  This finding is greater than minor because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely impacted the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Additionally, if left uncorrected, the finding would become a more significant safety concern. Specifically, debris left in the vicinity of the sump screen could partially impede flow to the residual heat removal (RHR) pumps, or result in head loss across a blocked sump screen affecting the net positive suction head available to the pumps, during the recirculation phase and long-term cooling following a loss-of-coolant accident or following a reactor vessel head drop event.

  This finding is of very low safety significance as the finding did not increase the likelihood that a loss of RHR reactor coolant system (RCS) inventory, RCS level control, or power would occur. The finding did not degrade the licensee’s ability to terminate a leak path, add RCS inventory, recover RHR once lost, establish an alternate core cooling path if RHR could not be re-established, or degrade the ability of containment to remain intact following a severe accident. The primary cause of this finding is related to the cross-cutting area of problem identification and resolution. The licensee failed to perform a causal analysis or extent-of-condition review, for the first instance of an inadequate sump debris inspection identified by the inspectors on October 4, 2005. (Section 1R20.1)

- Green. The inspectors identified a finding involving a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” having very low safety significance for the licensee’s failure to complete testing, to demonstrate that the containment sump isolation valves (SI-850s) would remain open during post-loss-of-coolant accident containment recirculation. This finding was entered into the licensee’s corrective action program. Part of the licensee’s corrective actions for this issue involved a visual examination of the valve actuator limit switches to determine the accuracy of valve status lights in the main control room.

  This finding is greater than minor because it is associated with the design control and equipment performance attributes of the Mitigating Systems Cornerstone and affected the equipment reliability objective for this cornerstone. Equipment reliability was affected because, as these valves begin to drift shut, the post-loss-of-coolant accident recirculation flow would be affected and require operator actions to compensate for valve drift to ensure adequate long-term core cooling. This finding is determined to be of very low safety significance because it was a design or qualification deficiency that did not result in loss of function per NRC Generic Letter 91-18. (Section 4OA2.1b.(1))

- Green. The inspectors identified a finding involving a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," having very low safety significance for the licensee's failure to implement prompt corrective actions and inspect carbon steel hydraulic operating cylinder components on the 1(2)SI-850(A)(B) valve actuators after the licensee became aware of the nonconforming and potentially degraded conditions involving boric acid deposits and associated corrosion. The licensee implemented actions to clean up the deposits and entered this finding into the corrective action program.

  This finding is greater than minor because, absent NRC intervention, this issue would have become a more significant safety concern if left uncorrected. Specifically, the licensee would have allowed an acidic environment (boric acid deposits) or aqueous environment (submerged fasteners) for these carbon steel components to continue for an indefinite period of time which could have resulted in corrosion-induced failures of the SI-850 valve actuators. In addition, the finding is associated with the equipment reliability attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of equipment reliability. This finding is determined to be of very low safety significance because it was a design or qualification deficiency that did not result in loss of function per NRC Generic Letter 91-18. The cause of the finding was related to the cross-cutting area of problem identification and resolution.
  (Section 4OA2.1b.(2))

- Green. The inspectors identified a finding involving a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety significance for the licensee's failure to correctly perform a static lift test of the 2SI-850B valve. This test was designed to record the hydraulic actuator pressure necessary to overcome valve dead weight and packing friction. This finding was entered into the licensee's corrective action program.

  This finding is greater than minor significance because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the equipment reliability objective for this cornerstone. Equipment reliability was affected because the incorrectly performed as-found static lift test of 2SI-850B did not provide the information needed to demonstrate the functional capability of this degraded valve. Although no definitive test data existed, the licensee staff believed that this degraded valve would have been functional with an identified oil leak (400 milliliters lost per closing stroke) because it stroked only 0.5 seconds slow for its open acceptance time during the quarterly stroke test and enough oil existed in the hydraulic reservoir to allow at least 10 open/close cycles. The licensee concluded that the valve was functional for past periods of operation with this hydraulic leak. This finding is determined to be of very low safety significance because it was a design or qualification deficiency that did not result in loss of function per NRC Generic Letter 91-18.
  (Section 4OA2.1b.(3))

- Green. The inspectors identified a finding of very low safety significance (with three examples) for the licensee's failure to notify the NRC within 8 hours in accordance with 10 CFR 50.72(b)(3)(ii)(B), following the identification that the nuclear power plant was in an unanalyzed condition that significantly degraded plant safety. Each occurrence was reported by the licensee following repeated questioning by the inspectors that occurred in April, September, and November 2005. Following the November occurrence, the inspectors reviewed the licensee's previous causal evaluations and corrective actions. The inspectors noted that while the licensee had appropriately evaluated and initiated corrective actions for the technical issues in April and September 2005, the licensee had not appropriately evaluated or developed any corrective actions to address the failure to adequately report these issues to the NRC in a timely manner.

  Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated using the traditional enforcement process. The finding has been reviewed by NRC management and is determined to be a Green finding of very low safety significance. Because the licensee entered the issue into their corrective action program (CAP068938), this violation is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The licensee has taken actions to perform a causal evaluation and address the knowledge and procedural aspects of this finding. The cause of the finding is related to the cross-cutting area of problem identification and resolution because the licensee failed to appropriately evaluate and take adequate corrective actions for the reportability aspect of these issues. (Section 4OA3.1)

- Green. A non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," having very low safety significance was identified by the inspector. Specifically, the licensee failed to promptly correct a condition adverse to quality, the potential for the auxiliary feedwater (AFW) recirculation line to crimp during a design basis earthquake (DBE) or design basis tornado (DBT) event. The licensee missed prior opportunities to correct the adverse condition: 1) as a result of the two Red findings related to the AFW system, the licensee reviewed the AFW system for the effects of high energy line break, DBE, and DBT events and identified crimping of the non-safety related portion of the common AFW recirculation line as a potential common mode failure; and 2) an external self-assessment in mid-2003 also concluded that crimping of the AFW recirculation line was credible and a potential common mode failure.

  The licensee corrected this adverse condition by: 1) installing a pretested replacement for AFW pump recirculation line relief valve AF-4035 that was manufactured to meet ASME Code Section VIII requirements; and 2) having commitments to periodically replace AFW recirculation line relief valve AF-4035 with a pretested valve. These actions provided reasonable assurance that AF-4035 would provide the required flowpath to protect the AFW pumps if the AFW recirculation line crimped during a DBE or DBT event. The licensee planned to supplement CAP066199 to address the inadequate corrective actions.

The finding is more than minor because it is associated with the design control and equipment performance attributes of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that mitigate transients and the reactor accidents. In addition, if left uncorrected, the finding would become a more significant safety concern. Specifically, if left uncorrected the AFW recirculation line relief valve could have deteriorated over time, failed to open as designed, and not provided the required recirculation line flowpath to protect the AFW pumps if the recirculation line crimped during a DBE or DBT event. The finding is of very low safety significance because testing of the original AFW recirculation line relief valve demonstrated that the relief valve would have opened as designed and would have provided the required AFW recirculation flowpath if the AFW recirculation line crimped during a DBE or DBT event. The cause of the finding is related to the cross-cutting area of problem identification and resolution, because the licensee failed to take adequate corrective actions.
(Section 4OA5.1)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period shutdown in the Cycle 29 Refueling Outage (U1R29) and remained there until December 6, 2005, when Unit 1 was returned to full power. Power remained at 100 percent until December 13, 2005, when the reactor was manually tripped by reactor operators in response to a loss of condenser vacuum, which was caused by the failure of the single running circulating water pump. Unit 1 returned to full power on December 17 and remained there, with the exception of routine downpowers for surveillances and testing, for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent power and remained there until November 2, 2005, when a shutdown was commenced to address an emergency core cooling system (ECCS) containment sump concern. The shutdown was suspended when the degraded coatings issue was resolved and the unit returned to full power the same day. Unit 2 remained at power until December 11, when power was reduced to approximately 65 percent to perform routine surveillances and testing. Unit 2 was returned to full power the same day and remained there, with the exception of routine downpowers for surveillances and testing, for the remainder of the inspection period.

4. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors walked down accessible portions of risk-significant equipment and systems susceptible to cold weather freezing. The inspectors also reviewed the licensee's preparation of the facade structures (around the reactor containments) and buildings inside the protected area. The inspectors reviewed the corrective actions and work orders (WOs) written to correct identified problems and assessed whether completion dates would ensure that corrective maintenance was completed prior to the onset of cold weather. The inspectors also walked down areas which had a history of freeze problems. This observation constituted one inspection procedure sample.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments (71111.02)

Reactor Vessel Closure Head (RVCH) Replacement (71007)

a. Inspection Scope

From September 12 through October 14, 2005, the inspectors reviewed the licensee's evaluations of applicability determinations and screening questions for the design changes associated with the Unit 1 RVCH replacement to determine, for each change, whether the requirements of 10 CFR 50.59 had been appropriately applied. Specifically, the inspectors reviewed Modification No. 03047, which included a review of the function of each changed component, the change description and scope, the 10 CFR 50.59/72.48 screening, and 10 CFR 50.59 evaluation for the:

- Replacement head
- Penetrations
- Lifting lugs
- Shroud support ring
- Thermal sleeves/guide funnels
- Control rod drive mechanism (CRDM) changes
- Removal of unused CRDM penetrations
- Core exit thermocouple penetration changes
- Reactor coolant gas ventilation system (RCGVS) changes
- Reactor vessel level indication system (RVLIS) changes

The inspectors also reviewed the 10 CFR 50.59/72.48 pre-screenings and screenings associated with Modification Nos. 03-046, 03-048, 03-049, 03-050, 03-051, 03-052, and 03-053 for:

- Install jib crane in containment
- Replace reactor vessel head insulation
- Head assembly upgrade package
- Unit 1 containment equipment hatch shield wall modification
- Analog rod position indicator system cable and connector modification
- CRDM cable and connector modification
- Replace core exit thermocouple cable and connectors

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," to determine acceptability of the completed pre-screenings and screening. 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." The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

The above constituted one completed sample for the inspection procedure.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of risk-significant systems to determine the operability of the systems. The inspectors utilized system valve lineup and electrical breaker checklists, tank level books, plant drawings, and selected operating procedures to determine if the systems were correctly aligned to perform the intended design functions. The inspectors also examined the material condition of the components and observed operating equipment parameters to determine if there were no obvious deficiencies. 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 two inspection procedure samples:

- Unit 1 residual heat removal (RHR) system alignment for refueling outage decay heat removal
- Unit 1 and 2 RHR system alignment for the ECCS recirculation phase

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Walkdown of Selected Fire Zones

a. Inspection Scope

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

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

- Fire Zone FZ-505; Unit 1 Containment, 8-foot elevation
- Fire Zone FZ-511; Unit 1 Containment, 21-foot elevation
- Fire Zone FZ-516; Unit 1 Containment, 46-foot elevation
- Fire Zone FZ-520; Unit 1 Containment, 66-foot elevation
- Fire Zone FZ-273; Unit 1 Heating, Ventilation and Air Conditioning Fan Room
- Fire Zone FZ-524; Unit 1 Containment Facade
- Fire Zone FZ-596; Unit 2 Containment Facade
- Fire Zone FZ-304N; Auxiliary Feedwater Pump Room North Section

b. Findings

No findings of significance were identified.

.2 Fire Protection Annual Fire Drill Observation (71111.05A)

a. Inspection Scope

The inspectors observed and evaluated the effectiveness of the fire brigade response to a simulated fire in the plant. The inspectors verified that protective clothing was available and properly donned by participants. The inspectors also verified that self-contained breathing apparatus equipment was properly utilized. In addition, the inspectors verified that all fire fighting equipment was in good condition and properly utilized. Finally, the inspectors verified that radio communications were effective between all participants involved in the drill.

The inspectors observed the actions of the fire brigade leader and the manner in which the fire strategy was implemented to extinguish the simulated fire. The fire drill plan contained evaluation criteria and was followed appropriately by fire drill coordinators. This inspection constituted one annual inspection procedure sample

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures - Internal Floods (71111.06)

a. Inspection Scope

The inspectors completed a walkdown of the AFW pump rooms to assess the overall readiness of flood protection equipment and barriers for protection against external flood sources. The inspectors evaluated flood protection features, such as flood doors, door gaps, and subsoil drains, to determine if the components were in satisfactory physical condition, unobstructed, and capable of providing an adequate flood barrier. The inspectors also reviewed design basis documents and risk analyses to evaluate the affects of the nonseismically-qualified condensate storage tank rupture on the AFW pump room components. This walkdown of the flood protection measures constituted one inspection procedure sample.

b. <u>Findings</u>

No findings of significance were identified.

1R08 <u>Inservice Inspection (ISI) Activities</u> (IP 71111.08)

.1 <u>Piping Systems ISI</u>

a. <u>Inspection Scope</u>

The inspectors conducted a review of the implementation of the licensee's Unit 1 ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and risk-significant piping system boundaries.

The inspectors reviewed the following nondestructive examination activities to evaluate compliance with the American Society of Mechanical Engineers (ASME) Code, Section XI and Section V, requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements. Specifically, the inspectors observed the following examinations:

- Ultrasonic examination (UT) of the U1C/A Loop welds RC-03-BP-1001-06 (pipe-to-elbow), RC-03-BP-1001-07 (elbow-to-pipe), RC-03-BP-1001-08 (pipe-to-elbow), RC-03-BP-1001-09 (elbow-to-pipe), and RC-03-BP-1001-11 (pipe-to-pipe)
- Dye penetrant examination (PT) of the number RC-03-PSF-1002-03 and 04, pressurizer power-operated relief valve piping welds and RC-04-PR-1001-04 and 05, pressurizer safety valve line welds

The inspectors reviewed the following examination with a recordable indication that was accepted by the licensee for continued service to verify that the licensee's acceptance was in accordance with the ASME Code or an NRC-approved alternative:

> The PT of welded Attachment AC-601R-2-H2-IWA, Weld 4 identified a 5/32-inch long linear indication. This indication was found to be properly evaluated in accordance with ASME 1998 Edition, 2000 Addenda, Section XI, IWB 3514.3, and was acceptable to leave in service.

The inspectors reviewed documents for the following completed pressure boundary welds for ASME Class 1 or 2 systems to verify that the welding process and welding examinations were performed in accordance with ASME Code requirements or an NRC-approved alternative:

- Repair ISI Class 2 safety injection (SI) accumulator T-34A nozzle indications
- Repair ISI Class 1 reactor vessel CRDM nozzle.

The above activities constituted one completed sample for the inspection procedure.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head Penetration Inspection

a. Inspection Scope

As part of the head replacement inspection activities discussed in Section 4OA5.1 of this report, the inspectors reviewed a sample of the weld records and nondestructive examination/preservice examination records for the replacement head. As this was conducted as part of Inspection Procedure IP 71007, this activity will not constitute a sample for IP 71111.08.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

Following shutdown, the inspectors reviewed a sample of boric acid corrosion control walkdown visual examination activities through direct observation. This walkdown was completed with Unit 1 in Mode 3 and included containment. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety significant components. The inspectors also reviewed the documentation of the VT-2 visual examination of the reactor pressure vessel bottom head penetrations.

The inspectors performed a review of ISI and boric acid related issues that were identified by the licensee and entered into the corrective action program. The review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspector performed these reviews to ensure compliance with the ASME Code and 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspector are listed in the Attachment to this report.

The above activities constituted one completed sample for the inspection procedure.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

From October 4 through October 7, 2005, the inspectors performed an on-site review of Unit 1 SG tube examination activities conducted pursuant to Technical Specifications (TS) and ASME Code Section XI requirements.

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

- The in-situ SG tube pressure testing screening criteria and the methodologies used to derive these criteria were consistent with the Electric Power Research Institute (EPRI) TR-107620, "Steam Generator In Situ Pressure Test Guidelines".
- The in-situ SG tube pressure testing screening criteria were properly applied in terms of SG tube selection based upon evaluation of the list of tubes with measured/sized flaws (none of the degraded SG tubes met the screening requirements for pressure testing).
- The numbers and sizes of SG tube flaws/degradations identified were bound by the licensee's previous outage operational assessment predictions.
- The SG tube ET examination scope and expansion criteria were sufficient to identify tube degradation based on-site and industry operating experience by confirming that the ET scope completed was consistent with the licensee's procedures, plant TS requirements and EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6".
- The SG tube ET examination scope included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions, and support plates.
- The licensee-identified new tube degradation mechanisms.
- The licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements.
- The licensee primary-to-secondary leakage (e.g., SG tube leakage) was below the detection threshold during the previous operating cycle.
- The ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, "Performance Demonstration for Eddy Current Examination," of EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6.
- The licensee-identified deviations from ET data acquisition or analysis procedures.

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

- The licensee had described the scope of the SG-related problems.
- The licensee had established an appropriate threshold for identifying issues.

- The licensee had evaluated industry generic issues related to SG tube integrity.
- The licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

The reviews discussed above did not count as a completed inspection sample as described in Section 71111.08-5 of the inspection procedure, but the sample was completed to the extent possible. Specifically, several portions of the procedure were not available for review because the licensee was not required to perform those activities during the outage.

For example, in-situ pressure testing and tube performance criteria were not available for review because none of the degraded SG tubes met the screening criteria for pressure testing.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Resident Inspector Quarterly Observation of Licensed Operator Training

a. Inspection Scope

On October 15, 2005, inspectors observed simulator training for filling and venting the reactor coolant system and the initial reactor coolant pump operation during solid plant operations. The inspectors verified that the training focused on high-risk operator actions, operator actions associated with normal operating procedures, operator actions associated with abnormal and shutdown emergency procedures required as contingencies, and recent operating experience along with previous lessons learned regarding these evolutions. Observation of the training evolutions constituted one inspection procedure sample.

b. Findings

No findings of significance were identified.

.2 Resident Inspector Quarterly Observation of Licensed Operator Requalification

a. Inspection Scope

On December 2, 2005, the inspectors observed the operating crew performance during a simulator requalification timed scenario. Observation of the requalification quarterly evaluation constituted one inspection procedure sample.

The inspectors assessed crew performance in the areas of:

- Clarity and formality of communications
- Understanding of the interactions and function of the operating crew during an emergency
- Prioritization, interpretation, and verification of actions required for emergency procedure use and interpretation
- Recognition of conditions requiring development of protective actions and preparation and communication of protective action recommendations
- Oversight and direction from supervisors
- Group dynamics

Crew performance in these areas was also compared to licensee management expectations and guidelines, as presented in Nuclear Procedure (NP) NP-2.1.1, "Conduct of Operations."

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors performed maintenance effectiveness reviews of the systems listed below. The inspectors reviewed repetitive maintenance activities to assess maintenance effectiveness, including maintenance rule activities, work practices, and common cause issues. Inspection activities included, but were not limited to, the licensee's categorization of specific issues, including evaluation of performance criteria, appropriate work practices, identification of common cause errors, extent of condition, and trending of key parameters. Additionally, the inspectors reviewed implementation of the Maintenance Rule (10 CFR 50.65) requirements, including a review of scoping, goal setting, performance monitoring, short-term and long-term corrective actions, functional failure determinations, and current equipment performance status.

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

- Unit 1 and Unit 2 Containment Sump Recirculation Valves (SI-850)

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

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

- Planned and emergent maintenance during the week of October 24, 2005
- Planned and emergent maintenance during the week of October 31, 2005
- Planned and emergent maintenance during the week of November 15, 2005
- Planned and emergent maintenance during the week of December 15, 2005

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Nonroutine Evolutions and Events (71111.14)

.1 Unit 2 TS 3.0.3 Entry Due to Containment Coating Inventories

a. Inspection Scope

On November 1, 2005, while addressing issues related to degraded coatings in the Unit 1 containment, licensee personnel discovered a potential issue associated with the degraded coating inventory in the zone of transport for the Unit 2 containment. Upon confirmation by personnel of the degraded coatings in Unit 2, the Shift Manager declared both trains of the ECCS inoperable for containment sump recirculation and entered TS 3.0.3. The licensee initiated actions to remediate the coatings and the plant operators initiated a downpower of Unit 2. The downpower was stopped at 97 percent when the degraded coatings in Unit 2 were remediated. The inspectors observed control room and in-plant activities during this nonrountine evolution.

The inspectors evaluated the licensee's operational decision making involved with this nonroutine evolution. In addition, the inspectors evaluated the operator's communications during the evolution, and the operator's application and adherence to the operating procedures and TS. This inspection constituted one annual inspection sample.

b. Findings

No findings of significance were identified.

.2 Retrieval of Unit 1 Control Rod F-8 Stuck Out in the Reactor Core

a. Inspection Scope

On November 10, 2005, the inspectors observed control room operators perform a final rod position indication functional test while Unit 1 was in Mode 3. Control rod F-8 would not go into the core past a position of 210 steps. Following some additional troubleshooting, the licensee returned Unit 1 to Mode 6 to perform an inspection of control rod F-8. On November 15, the licensee discovered and retrieved a piece of debris that was lodged in a guide card in the control rod guide tube. The inspectors observed and reviewed licensed operator performance in the control room and on the refueling bridge during this nonroutine evolution.

The inspectors also evaluated the licensee's operational decision making involved with this nonroutine evolution. In addition, the inspectors evaluated the operators' communications during the evolution, and the operators' application and adherence to the operating procedures. This inspection constituted one annual inspection sample.

b. Findings

No findings of significance were identified.

.3 Failure of the Operating Circulating Water Pump on Unit 1

a. Inspection Scope

The inspectors responded to a manual reactor trip that occurred on December 13, 2005, due to a loss of condenser vacuum caused by a mechanical failure of the running circulating water pump 1P-30B. The inspectors discussed the events associated with the reactor trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess followup actions. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify that actions and system responses were as expected. The inspectors evaluated portions of the licensee's Incident Investigation and Post-Trip Review Team meetings, discussed the scram with team members, and assessed the team's actions to gather, review, and assess information leading up to and following the scram. The inspectors later reviewed the initial investigation report to assess the detail of review and adequacy of the proposed corrective actions prior to unit restart.

At the end of the inspection period, the licensee was performing a root cause evaluation for the mechanical failure of circulating water pump 1P-30B. The inspectors will complete the event review when the licensee root cause is completed. This inspection constituted one annual inspection sample.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations Reviewed

a. Inspection Scope

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

- OPR 153; Calculated Short Circuit Currents Exceed Equipment Ratings (CAP067167)
- OPR 154; Overload Concerns of Safety Related Equipment (CAP067181)
- OPR 155; Transformers 1X-13 and 2X-14 Potentially Overloaded During LOCA (CAP067192)
- OPR156; Non-Conservative TS and Degraded Voltage Time Delay Setting (CAP067183)
- OPR 161; Containment Coatings Not Maintained Within Analyzed Limits (CAP068373)
- OPR 162; Questions with the Ability of ECCS Sump Screens to Pass Required Flow (CAP068447)
- OPR 164; Wax Found on Two Elevations on Floors in Unit 2 Containment (CAP068534)
- OPR 165; Tendon Gallery Inspection Results (CAP068629)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Review of Operator Workarounds

a. Inspection Scope

The inspectors reviewed operator workarounds with particular focus on the method by which instructions and contingency actions were communicated to and reviewed with on-shift licensed operators. The inspectors reviewed selected operator workarounds to determine if the functional capability of systems or human reliability in responding to an initiating event was affected. This review constituted two inspection samples.

The inspectors completed the sample by reviewing:

- Operator Workaround for Procedure OP-1C, "Startup to Power Operation"
- Operator Workaround for the Unit 2 Charging Pump 2-P2C

b. Findings

No findings of significance were identified.

.2 Cumulative Effect of Operator Workarounds

a. Inspection Scope

The inspectors reviewed selected operator workarounds to determine if the functional capability of systems or human reliability in responding to an initiating event was affected and assessed the cumulative effect of operator workarounds on plant operations. The inspectors reviewed outstanding operator workarounds to determine the overall complexity and aggregate effects on operator performance. The inspectors also reviewed selected control room WO deficiency tags and operator workaround meeting minutes, to determine if the licensee conducted periodic reviews and considered the total impact of outstanding WOs on risk and plant operations. Equipment out-of-service lists were reviewed to determine if there were operator workarounds that had not been identified as such. The review of selected operator workarounds and the cumulative effect of operator workarounds constituted one inspection procedure sample.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- Testing activities satisfied the test procedure acceptance criteria
- Effects of the testing were adequately addressed prior to the commencement of the testing
- Measuring and test equipment calibration was current
- Test equipment was within the required range and accuracy
- Applicable prerequisites described in the test procedures were satisfied
- Affected systems or components were removed from service in accordance with approved procedures
- Testing activities were performed in accordance with the test procedures and other applicable procedures
- 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
- All problems identified during the testing were appropriately entered into the corrective action program

During this inspection period, the inspectors completed the following inspection, which constituted one quarterly inspection procedure sample:

- Review of 1P-10A RHR pump motor replacement documentation and PMT procedure IT-03F, "1P-10A Low Head Safety Injection Pump Profile Test Mode 6 High Cavity Water Level Unit 1," performed October 20, 2005

b. Findings

Introduction: The inspectors identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for the failure to perform an equivalency evaluation for exceptions taken to motor specifications in the refurbishment of safety-related equipment, specifically the 1P-10A RHR pump motor. This finding was identified by the inspectors during the review of procurement documentation associated with the post-maintenance testing for 1P-10A following the motor replacement.

Description: On July 5, 2005, purchase order P304927, Revision 3, was initiated with the licensee's vendor to rewind, refurbish, and test the spare 1P-10 RHR pump motor in accordance with procedure PB 638, "Motor Repair Specification," Revision 1. The 1P-10A RHR pump motor was to be replaced during the Unit 1 refueling and reactor vessel head replacement in the fall of 2005. On July 22, 2005, the vendor sent a letter to the licensee acknowledging receipt of purchase order P304927 and included a completion schedule, quality assurance provisions, comments/exceptions to the purchase order, and equipment qualifications. The comments and exceptions section contained several exceptions to Point Beach's PB 638 safety-related motor specification, for example:

- Section 10.1.1 of PB 638 stated, in part, that resistance balance among the motor windings shall be within +/-0.25 percent of the average winding resistance. The vendor stated in the July 2005 letter that this criteria might not be met and that their standard was +/-1.5 percent of the average.
- Section 10.1.5 of PB 638 stated, in part, that core hot spot thermography testing personnel shall be qualified to ASNT SNT-TC-IA Level II. The vendor stated in the July 2005 letter that none of their personnel were qualified to this standard and that thermography would be performed by individuals qualified in accordance with the vendor's instructions.
- Section 10.2.4 of PB 638 stated, in part, that the maximum differential air gap shall be 5 percent. The vendor stated in the July 2005 letter that they considered a maximum air gap differential of 10 percent to be acceptable.

During the week of October 10, 2005, the inspectors reviewed procurement documents associated with the 1P-10 motor to determine the scope of the refurbishment activities and documents related to the upcoming post-maintenance test and return to service of the RHR 'A' train. During this time, Unit 1 was in Mode 6 and RHR was in the decay heat removal mode. The inspectors questioned whether the licensee adequately evaluated the exceptions taken by vendor to the licensee's safety-related motor specification. Further inspection revealed that there was no engineering or equivalency evaluation performed for the exceptions, and that the licensee's spare parts equivalency evaluation determination (SPEED) process was typically used to document equivalency issues.

The licensee agreed with the inspectors that an engineering evaluation was required to evaluate the exceptions taken to the specification prior to returning the 1P-10A RHR pump to service. The licensee also initiated a CAP to determine the cause and extent-of-condition of this issue.

Analysis: The inspectors determined that the failure to perform an equivalency evaluation for exceptions taken to specifications for the refurbishment of the safety-related 1P-10A RHR pump motor was a performance deficiency. The inspectors determined that the finding is greater than minor because it was associated with the design control 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. The inspectors evaluated the finding using Inspection Manual Chapter (IMC) 0609, Appendix G, Phase 1 Screening, and determined that Checklist 4, "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer," applied, specifically Section I.C, "Core Heat Removal Guidelines - Equipment." However, because the 'A' RHR loop was not in operation and the 'B' train RHR loop was operable and in operation with support systems available, the inspectors determined that Section I.C was not affected. In addition, the finding did not meet the Checklist 4 criteria for Phase 2 or Phase 3 quantitative analysis because the finding did not: increase the likelihood of a loss of RCS inventory, including a loss of

RCS level instrumentation; degrade the licensee's ability to terminate a leak path or add RCS inventory when needed; or degrade the licensee's ability to recover decay heat removal once it was lost. Consequently, the finding was determined to be of very low safety significance (Green). The inspectors also determined that the finding is of very low safety significance because no event occurred that could be characterized as a loss of control of RCS temperature or inventory, as listed in Table 1 of IMC 0609, Appendix G.

Enforcement: 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Contrary to this, the licensee failed to perform an equivalency evaluation for exceptions taken to motor specifications in the refurbishment of safety-related equipment, specifically the 1P-10A RHR pump motor. Because this violation is of very low safety significance and it was entered into the licensee's corrective action program (as CAP030273), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000266/2005013-01; 05000301/2005013-01).

The licensee took several corrective actions, including completion of a 10 CFR 50.59 screening, an engineering evaluation, and a SPEED before returning the 1P-10A RHR pump to service. The licensee determined from these analyses that the 1P-10A motor was acceptable. The licensee also initiated an apparent cause evaluation including an extent-of-condition review to ensure that other equivalent plant replacements that were affected are addressed.

1R20 Refueling and Outage Activities (71111.20)

.1 Failure to Adequately Implement Procedures Affecting U1 Containment Walkdowns

a. Inspection Scope

During the Unit 1 Cycle 29 Refueling Outage (U1R29), on October 5 and November 4, 2005, the inspectors conducted walkdowns of the Unit 1 Containment to verify the ECCS sumps were free of debris for the reactor vessel (RV) head lift and the transition from Mode 5 to Mode 4. The inspectors reviewed the licensee's compliance with Technical Requirements Manual (TRM) 3.9.4, "Administrative Controls during Reactor Vessel Head Lift" and with TS Surveillance Requirement (SR) 3.5.2.6, verification of ECCS sump suction inlet free of restriction.

b. Findings

Introduction: A Green finding associated with a non-cited violation (NCV) of TS 5.4.1, Procedures, was identified by the inspectors when the licensee failed on two different occasions to perform adequate containment walkdowns to verify that: (1) no foreign material was stored within 20 feet of the ECCS sump screen and all tape was removed from the 8-foot and 10-foot elevation platforms for the RV head lift; and (2) the sump suction inlet was not restricted by debris and the suction inlet debris screens showed no evidence of structural distress or abnormal corrosion prior to entry into Mode 4.

Description: On October 5, 2005, after the licensee completed a walkdown to verify the containment sump flow path inspection required by procedure RP-1A, Attachment H, to satisfy TRM requirements for Unit 1 reactor vessel head lift, the inspectors identified tape used to post radiation boundaries for high contamination areas, duct tape on other miscellaneous objects, and other debris on the 8-foot and 10-foot levels of containment in the vicinity of the ECCS sump. Procedure RP-1A, 'Preparation for Refueling' required, in part, that all floatable items shall be secured and no foreign material be stored within 20 feet of the sump screen. The licensee subsequently removed these items to address the deficiencies prior to the RV head lift and initiated CAP067629. The licensee assigned a significance level 'C' to CAP067629. The licensee's corrective action Procedure, NP 5.3.1, "Action Request Process," defined a level 'C' action request as a condition adverse to quality, or a detrimental condition which typically resulted in minor impact to the plant and/or organization. Level C CAPs did not have either a root cause or apparent cause, or extent-of-condition evaluation performed. The licensee prepared a procedure change request to revise Attachment H of RP-1A to further clarify the requirements for cleanliness in the vicinity of the ECCS sump.

On November 4, 2005, the inspectors performed a walkdown of the Unit 1 containment to verify general containment cleanliness, and specifically to verify that each ECCS train containment sump suction inlet was not restricted by debris and that the suction inlet debris screens showed no evidence of structural distress or abnormal corrosion as required by the TS SR 3.5.2.6, prior to entry into Mode 4. The licensee had completed the subject inspection on November 2. On November 4, the inspectors identified several issues associated with the Unit 1 containment in the vicinity of ECCS sump 'B', including loose paint chips on equipment next to the sump, floor epoxy lifting in the vicinity of the ECCS sump screen, loose paint chips and encrusted boric acid between the sump coarse outer screen and fine mesh inner screen, debris on top of the sump screen, apparent loose paint on rubber expansion joints, degraded coatings in the vicinity of the ECCS sump, and discoloration that appeared to be acrylic polymer in the vicinity of the sump screen.

The licensee confirmed the resident's observations, initiated CAP068637, and performed additional inspections and cleanup. The licensee identified additional debris and remediated the issues, in addition to removing degraded coatings that were in the general area of the ECCS sump. Additionally, the licensee performed an apparent cause evaluation for the subject CAP. The evaluation identified the apparent cause was a lack of clear understanding of management's expectations for containment closure and current industry standards, compounded by inadequate guidance within the checklist to conduct an effective containment closure inspection.

In reviewing these two failures to perform adequate containment inspections for ECCS sump debris walkdowns to ensure operability, the inspectors identified that no causal analysis or extent-of-condition review was performed for the first instance identified by the inspectors on October 4, 2005, for the RV head lift walkdown deficiencies (CAP067629). The inspectors concluded, upon reviewing the licensee's main corrective action program procedure that CAP067629 was not adequately scoped; therefore, an appropriate causal/extent-of-condition analysis was not performed.

Analysis: The inspectors determined that the licensee's failure to adequately perform containment walkdowns to ensure ECCS sump operability is a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding is greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on September 30, 2005, in that, the finding was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely impacted the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In addition, if left uncorrected, the finding would become a more significant safety concern. Specifically, debris left in the vicinity of the ECCS sump screen could partially impede flow to the RHR pumps or result in head loss across a blocked sump screen affecting the net positive suction head available to the RHR pumps during the recirculation phase and long-term cooling following a loss-of-coolant accident (LOCA).

The inspectors evaluated the finding using IMC 0609, Appendix G, Phase 1 Screening, CL 3, "PWR Cold Shutdown and Refueling Operation, RCS Open and Refueling Cavity Level < 23', Time to Boiling < 2 hours," which was applicable to this finding. The finding affected the ECCS sump which was required to be operable in Mode 4 and the RV head lift; however, the finding did not meet the requirement for a Phase 2 or Phase 3 analysis per Appendix G, because the finding did not increase the likelihood that a loss of: RHR, RCS inventory, RCS level control, or power would occur. The finding also did not degrade the licensee's ability to terminate a leak path, add RCS inventory, recover RHR once lost, establish an alternate core cooling path if RHR could not be re-established, or degrade the ability of containment to remain intact following a severe accident. Therefore, the finding was considered to be of very low significance (Green).

The inspectors also determined that a primary cause of this finding was related to the cross-cutting area of problem identification and resolution. The licensee failed to perform a causal analysis or extent of condition review, for the first instance of an inadequate ECCS sump debris inspection identified by the inspectors on October 4, 2005. The inspectors concluded, upon reviewing the licensee's CAP procedures that CAP067629 was not adequately scoped; therefore, an appropriate causal/extent-of-condition analysis was not performed.

Enforcement: Technical Specifications 5.4.1 requires, in part, that written procedures be established, implemented, and maintained in part covering surveillance of safety-related equipment. Contrary to this, the licensee failed to adequately implement procedure RP-1A and checklist (CL)-20 for verification that the area of the containment ECCS sump was free of foreign material and debris. The procedure requirements were signed off with debris still existing in the vicinity of the ECCS sump. Because this violation was of very low safety significance and has been entered into the licensee's corrective action program (as CAP067629 and CAP068637), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 0500266/2005013-02).

The licensee took immediate interim corrective actions which included removal of the identified debris from containment and coaching responsible personnel on expectations for following the requirements of RP-1A and CL-20. Subsequently, the licensee completed apparent cause evaluation ACE001970 which developed additional corrective actions, including the operations manager discussing this incident with all senior reactor operators (SROs) during training cycle operating experience sessions to ensure management expectations are clearly disseminated, updating the checklist to include what to look for during closure activity, final closeout inspection conducted/approved by plant manager or designee, and inclusion of an SRO sign-off for general containment cleanliness.

.2 Routine Refueling Outage Inspection Activities

a. Inspection Scope

The inspectors observed activities during the Unit 1 refueling outage (U1R29) conducted from September 24 to December 6, 2005. These inspection activities constituted one refueling outage inspection sample.

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

The inspectors conducted the following inspection activities:

- Attended outage management turnover meetings to determine if the current shutdown risk status was accurate, well understood, and adequately communicated
- Performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk
- Observed the operability of RCS instrumentation and compared channels and trains against one another
- Performed in-plant walkdowns to observe ongoing work activities
- 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:

- Performed Mode 3 walkdowns at the start and end of the refueling outage to check for active boric acid leak indications
- Observed the control room staff perform the Unit 1 shutdown and initial cooldown
- Verified that RCS cooldown rates were within TS limits
- Observed control room staff operations during reduced inventory conditions
- Observed core unloading activities in the containment, spent fuel pool, and control room
- Observed core reload from containment

- Observed preparations for and portions of stuck rod F-8 recovery and debris inspection
- Observed inspection results from recovery of foreign material from camera inspections of rod location H-3
- Observed operators align the RHR system for shutdown cooling
- Observed placement of the over-pressure protection system into operation
- Monitored a pre-job briefing for fuel handling evolutions
- Observed lifting and transport of the RVCH in preparation for core offload
- Observed the RVCH head replacement
- Performed a closeout inspection of the Unit 1 containment, including a review of the results of the emergency core cooling sump inspection that had been performed earlier by the licensee. As part of this inspection, the inspectors also assessed whether all discrepancies noted during the walkdown were recorded and corrected.
- Reviewed shutdown margin calculations
- Reviewed spent fuel pool cooling and service water pump configurations during partial core offload
- Observed operation of the fuel handling bridges in containment and the spent fuel pool
- Reviewed Mode - change checklists (CLs) to verify that selected requirements were met while transitioning from the refueling Mode to full power operations
- Observed portions of low power physics testing and approach to criticality
- Observed portions of the plant ascension to full power operations

b. Findings

No findings of significance were identified other than the finding discussed in Section R20.1.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- Preconditioning occurred
- Effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing
- Acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis
- Plant equipment calibration was correct, accurate, properly documented, as-left setpoints were within required ranges, and the calibration frequency was in accordance with TSs, the FSAR, procedures, and applicable commitments
- Measuring and test equipment calibration was current
- Test equipment was used within the required range and accuracy
- Applicable prerequisites described in the test procedures were satisfied
- Test frequencies met TS requirements to demonstrate operability and reliability

- Tests were performed in accordance with the test procedures and other applicable procedures
- 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, ASME Code, and reference values were consistent with the system design basis
- Equipment was returned to a position or status required to support the performance of its safety functions
- All problems identified during the testing were appropriately documented and dispositioned in the corrective action program

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

- Unit 1 ECCS SI valves SI-850A and SI-850B
- Unit 1 and 2 unidentified leakrate calculations

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

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

- 2004-001, Temporary Replacement of Unit 1 Purge Supply/Return Valves
- 2005-016, Unit 1 Reactor Cavity Portable Sump Pump Installation
- 2003-014 and 2005-005, Sump Pumps for Manholes 1 and 2

The review included associated work orders, temporary modification instructions/procedures, and 10 CFR 50.59 screenings and evaluations. The review of the temporary modifications constituted three inspection procedure samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspectors performed a screening review of the following revisions of portions of the Point Beach Nuclear Plant Emergency Plan to determine whether the changes made in these revisions decreased the effectiveness of the licensee's emergency planning: Appendix C, Revision 14; Appendix J, Revision 13; and Appendix L, Revision 2. This screening review did not constitute an approval of the changes and, as such, the changes are subject to future NRC inspection to ensure that the emergency plan continues to meet NRC regulations.

These activities completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors selected emergency preparedness exercises that the licensee had scheduled. The inspection activities included, but were not limited to, the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critiques. Observations were compared with the licensee's observations and corrective action program entries. The inspectors verified that there were no discrepancies between observed performance and performance indicator (PI) reported statistics.

The inspectors selected the following emergency preparedness activities for review for a total of three samples:

- An emergency preparedness practice drill conducted with the Technical Support Center and Emergency Operations Facility
- Operating crew performance during a simulator training exercise addressing the classification of emergency events and the development and notification of protective action recommendations
- Operating crew performance during a training exercise addressing the classification of emergency events during a simulated fire scenario

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns/Boundary Verifications and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors identified recently completed and ongoing work performed within high and locked high radiation areas of the plant and other potentially exposure significant work activities and selectively reviewed radiation work permit (RWP) packages and radiation surveys for these areas. The inspectors evaluated the radiological controls to determine if these controls, including postings and access control barriers, were adequate.

The inspectors walked down radiologically significant area boundaries and other radiological areas in the auxiliary building and the Unit 1 containment building to determine if the prescribed radiological access controls were in place, that licensee postings were complete and accurate, and that physical barricades/barriers were adequate. During the walkdowns, the inspectors physically challenged locked gates/doors to verify that high radiation area and locked high radiation area (LHRA) access was controlled in compliance with the licensee's TS and the requirements of 10 CFR 20.1601, and were consistent with Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

The inspectors reviewed RWP work packages for selected radiologically significant activities that were recently completed or were ongoing during the Unit 1 refueling outage to determine if barrier integrity and engineering controls performance (for example, filtered ventilation system operation) were adequate and to determine if there was a potential for individual worker internal exposures of greater than 50 millirem committed effective dose equivalent.

The inspectors reviewed the licensee's physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) that could be stored within the spent fuel pool. Specifically, applicable radiation protection (RP) procedures were reviewed, RP staff were interviewed, and a walkdown of the refuel floor was conducted. Although highly activated/contaminated materials were not stored in the spent fuel pool in a manner that readily allowed their inadvertent movement, the radiological controls for the storage of such materials was discussed with RP staff to ensure adequate barriers would be established should the licensee change its practices.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

.2 High Risk Significant, LHRA, and Very High Radiation Area (VHRA) Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's generic practices for the control of access to radiologically significant areas (high, locked high, and very high radiation areas). The inspectors assessed compliance with the licensee's TS and the requirements of 10 CFR Part 20, along with the guidance contained in Regulatory Guide 8.38. The inspectors also reviewed RP procedures and practices for obtaining RP management approval for access into high dose rate LHRAs or VHRAs and its practices for the use of flashing lights in lieu of locking areas to verify compliance with TS and with 10 CFR 20.1601 and 10 CFR 20.1602.

The inspectors discussed with RP staff the controls that were in place for areas that had the potential to become high or locked high radiation areas during certain plant operations to determine if these plant operations required communication before hand with the RP group, so as to allow corresponding timely actions to properly post and control the radiation hazards. In particular, reactor operations procedures and RP procedures/job files developed to identify vulnerable areas subject to changing radiological conditions were reviewed and their implementation was discussed with RP supervisory staff.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant outage exposure history, current Unit 1 outage exposure trends, and ongoing outage activities in order to assess performance and exposure challenges. This included determining the plant's current 3-year rolling average for collective exposure in order to provide a perspective of significance for any resulting inspection finding assessment.

The inspectors reviewed the Unit 1 refueling outage work and the associated work activity exposures and time/labor estimates for the following outage work activities which were likely to result in the highest personnel collective exposures or were otherwise radiologically significant activities:

- Reactor Vessel Head Replacement Activities
- Steam Generator Maintenance Activities
- Containment Scaffolding Installation/Removal
- Containment Mechanical Valve Maintenance Activities

- Containment Insulation Installation/Removal
- Under-Vessel Boric Acid Cleanup

The inspectors determined site specific trends in collective exposures based on plant historical data. The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and assessed those processes used for the Unit 1 outage to project dose and track work activity exposures.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors obtained the licensee's list of Unit 1 outage work activities ranked by estimated exposure and reviewed the following radiologically significant work activities:

- Steam Generator Nozzle Dam Installation/Removal (RWPs 05-133 and 05-164)
- Reactor Upper Cavity Decontamination (RWP 05-140)
- Containment Mechanical Valve Maintenance Activities (RWP 05-123)
- Steam Generator Eddy Current Testing (RWP 05-134)
- Under-Vessel Boric Acid Cleanup (RWP 05-167)
- Reactor Vessel Head Replacement Activities (RWP 05-157)

For each of the activities listed above, the inspectors reviewed the ALARA plan and associated total effective dose equivalent (TEDE) ALARA evaluation, as applicable, time/labor estimates and dose projections, and exposure mitigation criteria to determine if the licensee had established radiological engineering controls that were based on sound radiation protection principles and to achieve occupational exposures that were ALARA. This also involved determining if the licensee had reasonably grouped the radiological work into activities that were based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the exposure results achieved through approximately 70 percent of the scheduled 5-week outage including the person-rem expended with the doses projected through the licensee's ALARA planning for the above listed work and for several other selected outage activities. Reasons for inconsistencies between intended (projected) and actual work activity doses were evaluated to determine if the activities were planned reasonably well and to ensure the licensee identified any work planning deficiencies. The integration of ALARA requirements into work packages and RWP documents was reviewed to verify that the licensee's radiological job planning was adequately captured so as to reduce dose.

The licensee's ALARA in progress reports were reviewed by the inspectors for those outage jobs that approached their respective dose estimates or ALARA procedure specified exposure thresholds. The reports were reviewed to verify that the licensee could identify problems and address them as work progressed. Dose significant outage jobs that exceeded 125 percent of their respective dose projections or were anticipated to exceed dose projections were also reviewed to ensure that work was suspended, if warranted, and that identified problems were entered into the licensee's corrective action program.

These reviews represented five inspection samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

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

The inspectors reviewed the licensee's process for adjusting outage exposure estimates when unexpected changes in scope, emergent work or other unanticipated problems were encountered which could significantly impact worker exposures. This included determining that adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to effectively plan or control the work.

The licensee's exposure tracking methods were reviewed to determine whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of outage work exposures. RWPs were reviewed to determine if they covered an excessive number of work activities to allow specific exposure trends to be detected and controlled.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.4 Radiation Worker Performance

a. Inspection Scope

Radiation worker and radiation protection technician performance was observed during work activities being performed in the Unit 1 containment. The inspectors determined whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope, by utilizing low dose waiting areas, and by adhering to the ALARA requirements for the work activity.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems Associated with the Radiological Access and ALARA Programs

a. Inspection Scope

The inspectors reviewed the results of a recent RP department self-assessment related to the radiological access/ALARA program, reports of Nuclear Oversight Department (quality assurance) field observations of various radiological activities, and the corrective action program database along with individual CAPs related to the radiological access and exposure control programs to determine if identified problems were entered into the corrective action program for resolution. The inspectors focused on radiological problems which occurred over the approximate 6-month period that preceded the inspection (since the previous NRC review of this area), including the review of any high radiation area (HRA) radiological incidents (non-PI occurrences identified by the licensee in high and locked high radiation areas) to determine if follow-up activities were conducted in an effective and timely manner commensurate with their importance to safety and risk, based on the following:

- Initial problem identification, characterization, and tracking
- Disposition of operability/reportability issues
- Evaluation of safety significance/risk and priority for resolution
- Identification of repetitive problems
- Identification of contributing causes
- Identification and implementation of corrective actions
- Implementation of risk significant operational experience feedback

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

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the FSAR for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit program with regard to radioactive material processing and transportation programs to verify that it met the requirements of 10 CFR 20.1101(c).

This represents one sample.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors performed walkdowns of the solid radwaste processing systems to verify that the systems agreed with the descriptions in the FSAR and the Process Control Program and to assess the material condition and operability of the systems. The inspectors reviewed the status of radioactive waste process equipment that was not operational and/or was abandoned in place. The inspectors reviewed the licensee's administrative and physical controls to ensure that the equipment would not contribute to an unmonitored release path or be a source of unnecessary personnel exposure.

The inspectors reviewed changes to the waste processing system to verify that the changes were reviewed and documented in accordance with 10 CFR 50.59 and to assess the impact of the changes on radiation dose to members of the public, as applicable. The inspectors reviewed the current processes for transferring waste resin, and evaporator bottoms into shipping containers to determine if appropriate waste stream mixing and sampling methods were utilized. The inspector also reviewed the methodologies for waste concentration averaging to determine if representative samples of the waste product were provided for the purposes of waste classification in 10 CFR 61.55.

This represents one sample.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee’s radiochemical sample analysis results for each of the licensee’s waste streams, including dry active waste (DAW), spent primary resins, blowdown evaporator bottoms, and process stream filters. The inspectors also reviewed the licensee’s use of scaling factors to quantify difficult-to-measure radionuclides, for example, pure alpha or beta emitting radionuclides. The reviews were conducted to verify that the licensee’s program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee’s waste characterization and classification program to ensure that the waste stream composition data accounted for changing operational parameters and thus remained valid between the annual sample analysis updates.

This represents one sample.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation

a. Inspection Scope

The inspectors reviewed shipment surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness for the following:

- Shipment 2004-021, Temporary Demin Equipment
- Shipment 2004-055, Blowdown Evaporator Bottoms
- Shipment 2005-011, Resin for Processing
- Shipment 2005-028, RT-11 Source
- Shipment 2005-032, Contaminated Split Pin Equipment for Return
- Shipment 2005-033, Contaminated Split Pin Equipment for Return
- Shipment 2005-036, Reactor Head Replacement Equipment
- Shipment 2005-045, Unit-2 Reactor Vessel Head
- Shipment 2005-068, Contaminated Laundry for Processing
- Shipment 2005-070, Unit-1 Reactor Vessel Head

The inspectors verified that the requirements of applicable transport cask Certificate of Compliance were met and verified that the receiving licensee was authorized to receive the packages. The inspectors verified that the licensee’s procedures for cask loading and closure were consistent with the vendor’s approved procedures. The inspectors verified that the workers had adequate skills to accomplish each task and determined that the shippers were knowledgeable of the shipping regulations and that shipping personnel demonstrated adequate skills to accomplish the package preparation requirements for public transport with respect to NRC Bulletin 79-19 and 49 CFR Part 172 Subpart H. The inspectors reviewed the training records provided to personnel responsible for the conduct of radioactive waste processing and radioactive shipment preparation activities. The review was conducted to verify that the licensee's training program provided training consistent with NRC and Department of Transportation (DOT) requirements.

This represents one sample.

b. Findings

No findings of significance were identified.

.5 Shipping Records

a. Inspection Scope

The inspectors reviewed 10 non-excepted package shipment manifests/documents completed in 2004-2005 to verify compliance with the NRC and DOT requirements of 10 CFR Parts 20 and 71 and 49 CFR Parts 172 and 173.

This represents one sample.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed CAPs, audits, and self-assessments that addressed radioactive waste and radioactive materials shipping program deficiencies since the last inspection to verify that the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized, and corrected. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also reviewed corrective action reports from the radioactive material and shipping programs since the previous inspection, interviewed staff, and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- Initial problem identification, characterization, and tracking
- Disposition of operability/reportability issues
- Evaluation of safety significance/risk and priority for resolution
- Identification of repetitive problems
- Identification of contributing causes
- Identification and implementation of effective corrective actions

- Resolution of NCVs tracked in corrective action system(s)
- Implementation/consideration of risk significant operational experience feedback

This represents one sample.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs (71122.03)

.1 Inspection Planning and Reviews of Radiological Environmental Monitoring Reports and Data

b. Inspection Scope

The inspectors reviewed the 2003 and 2004 Annual Monitoring Reports and licensee assessment results to verify that the Radiological Environmental Monitoring Program (REMP) was implemented as required by the licensee's TS, the Offsite Dose Calculation Manual (ODCM), and the licensee's Environmental Manual (EM). The inspectors reviewed the reports for changes to the ODCM and EM with respect to environmental monitoring: commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, interlaboratory comparison program, and analysis of data. The inspectors reviewed the ODCM and EM to identify environmental monitoring stations and reviewed licensee self-assessments, audits, licensee event reports (LERs), and interlaboratory comparison program results. The inspectors reviewed the FSAR for information regarding the environmental monitoring program and meteorological monitoring instrumentation.

These reviews represented one sample.

b. Findings

No findings of significance were identified.

.2 On-site Inspection

a. Inspection Scope

The inspectors walked down four on-site and one offsite environmental air sample monitoring stations and examined each station's location as described in the EM. The inspectors observed equipment material condition and operability and verified proper monitoring station orientation, equipment configuration, and vegetation growth control to assess if each station allowed for the collection of representative samples. The inspectors walked down the locations of 13 thermoluminescent dosimeters (TLDs), which measured radiation levels directly, to verify they were positioned as described in the EM. The inspectors accompanied a radiation protection technician and observed sample collection and handling associated with the changing-out of air particulate filters and charcoal cartridges. This review covered all five of the licensee's environmental air sampling indicator stations. The purpose of the accompaniment was to evaluate whether samples were collected in accordance with the applicable sampling procedure and whether appropriate practices were used to ensure sample integrity and chain-of-custody. The inspectors also observed the performance of air sampling device leak checks to verify that they were accomplished consistent with the EM and were adequate to ensure no in-leakage paths existed which could impact sample representativeness.

The inspectors also walked down equipment located at the primary and backup meteorological towers to assess whether the towers were sited adequately, the instrumentation was installed consistent with NRC Safety Guide 23, "Onsite Meteorological Programs," and the instrumentation was operable, calibrated, and maintained in accordance with guidance contained in the FSAR, NRC Safety Guide 23, and licensee procedures. The inspectors verified that the meteorological data readout and recording instruments in the control room and at the tower were operable. In addition, the inspector observed a calibration test of the primary meteorological tower and discussed data recording capabilities with the licensee's staff to verify that meteorological data were sampled and compiled consistent with the NRC Safety Guide.

The inspectors reviewed each event documented in the Annual Monitoring Reports which involved a missed sample, inoperable sampler, lost TLD, or anomalous measurement for the cause and corrective actions and conducted a review of the licensee's assessment of any positive sample results, that is, licensed radioactive material detected above the lower limits of detection (LLDs). The inspectors reviewed the data for the associated radioactive effluent release that was the likely source of the positive result.

The inspectors reviewed significant changes made by the licensee to the ODCM and EM as the result of changes to the land census or sampler station modifications since the last inspection. There were no significant changes made during the period reviewed. The inspectors reviewed technical justifications for changed sampling locations. The inspectors verified that the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

The inspectors reviewed calibration and maintenance records for 2004 and 2005 which documented work on environmental air sampling pumps and meteorological tower equipment. This review encompassed calibration records for associated measurement and test equipment, such as the rotameters used for air sampling pump calibration, to verify that the testing and maintenance programs for this equipment were implemented consistent with procedural requirements and industry standards, including traceability to the National Institute of Standards and Technology. The inspectors discussed equipment maintenance practices with the licensee's environmental staff.

The inspectors reviewed the results of the REMP sample vendor's quality control program including the inter-laboratory comparison program to assess the adequacy of the vendor's program and the corrective actions for any identified deficiencies. The inspectors reviewed audits and technical evaluations the licensee performed on the vendor's program. The inspectors reviewed Nuclear Oversight Department audit results of the program to determine whether the licensee met the ODCM requirements.

These reviews represented six samples.

b. Findings

No findings of significance were identified.

.3 Unrestricted Release of Material From Radiologically Controlled Areas

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiologically controlled area and inspected the methods used for control, survey, and release from these areas. The inspectors observed RP personnel surveying and releasing material for unrestricted use to verify that the work was performed in accordance with plant procedures.

The inspectors verified that the radiation monitoring instrumentation was appropriate for the radiation types present and was calibrated with appropriate radiation sources. The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material and verified that there was guidance on how to respond to an alarm which indicates the presence of licensed radioactive material. The inspectors reviewed the licensee's equipment to ensure the radiation detection sensitivities were consistent with the NRC guidance contained in IE Circular 81-07 and IE Information Notice 85-92 for surface contamination and Health Physics Position (HPPOS-221) for volumetrically contaminated material. The inspectors verified that the licensee performed radiation surveys to detect or otherwise evaluated the impact of radionuclides that decay via electron capture. The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters, that is, counting times and background radiation levels. The inspectors verified that the licensee had not established a "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high radiation background area.

These reviews represented two samples.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed licensee corrective action documents originated during 2003 through October 2005 that related to the REMP or to radioactive material control issues.

The results of a Nuclear Oversight Department audit and a REMP self-assessment completed during the same time were also reviewed, as were the results of a joint nuclear utility audit of the vendor laboratory. These reviews were conducted to determine if the licensee adequately assessed the effectiveness of these programs and whether the licensee, through its corrective action program, identified individual problems and trends, evaluated contributing causes and extent of condition, and developed corrective actions to achieve lasting results. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also reviewed corrective action reports from the radioactive environmental monitoring program and unconditional release program since the previous inspection, interviewed staff, and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- Initial problem identification, characterization, and tracking
- Disposition of operability/reportability issues
- Evaluation of safety significance/risk and priority for resolution
- Identification of repetitive problems
- Identification of contributing causes
- Identification and implementation of effective corrective actions
- Implementation/consideration of risk significant operational experience feedback

These reviews represented one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Public Radiation Safety

a. Inspection Scope

The inspectors sampled the licensee’s submittals for the performance indicator (PI) and periods listed below. The inspectors used PI definitions and guidance contained in Revision 3 of NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” to verify the accuracy of the PI data. The following PI was reviewed:

- Radiological Effluent TS/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrence

The inspectors reviewed the licensee’s CAP database and selected CAPs generated from the 3rd quarter 2004 through 3rd quarter 2005, to identify any potential occurrences, such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous and liquid effluent monthly summary data and the results of selected offsite dose calculations through 3rd quarter 2005 to determine if indicator results were accurately reported.

This represents one sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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

a. Inspection Scope

The inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to determine if issues were entered into the licensee's corrective action system at an appropriate threshold and that adequate attention was given to timely corrective actions. The inspectors also reviewed CAPs written by licensee personnel during the inspection period. A more in-depth review was conducted on conditions that potentially affected long-term emergency core cooling and offsite leakage performance. This review constituted two inspection samples:

For the first sample, the inspectors reviewed CAPs, OPRs, and supporting documents related to coating degradation and debris accumulation in the Unit 1 and Unit 2 sumps, and performed walkdowns of the Unit 1 containment including the sump 'B' (the post-accident recirculation sump) area.

For the second sample, inspectors performed an in-depth review of the four sump isolation valves 1(2)SI-850(A)(B) and leakage testing for SI system piping outside containment. This review included CAPs, drawings, completed maintenance, and/or testing documented in WOs, the SI system Design Basis Document (DBD) Volume 11, Sections 5 and 6 of the FSAR, vendor manuals, and a walkdown of the containment sump and tendon gallery areas.

The inspectors conducted these reviews to determine if the sump and SI-850 valve design basis had been maintained and that the maintenance and test program was sufficient to ensure adequate sump and SI-850 valve function to support long term emergency core cooling.

b. Findings

(1) Inadequate Design Verification Testing of SI 850 Valves

Introduction: The inspectors identified a finding involving an NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety significance (Green) for failure to complete testing (pre-installation, preoperation, or operational) to demonstrate that the containment sump isolation valves (SI-850s) would remain open during post-LOCA containment recirculation.

Description: The SI-850 containment sump isolation valves are required to open and to stay open for long-term sump recirculation and containment heat removal following a LOCA. The license identified that these valves have a mission time to remain open for approximately 1 year in support of long-term emergency core cooling. The licensee performed quarterly valve testing to demonstrate that these valves would open and close as designed. However, this testing did not measure or determine the capability of these valves to remain open under the influence of valve shaft dead weight and flow induced drag force on the valve seat which would tend to close the valve. Leakage of hydraulic fluid past the actuator piston seals, or leakage by the seats of the hydraulic bypass line isolation valves, or at numerous hydraulic line compression/threaded fittings would allow these valves to move toward the close position. The rate of valve closure was unknown, but the licensee assumed that it would be slow if leakage from the above components was minimal, or if packing friction was sufficient to offset the closing force. The inspectors identified a number of factors which would shorten the time that these valves would stay open, including the hydraulic actuator piston seals had been in-service for more than 8 years without periodic inspections, the hydraulic line bypass valves had not been seat leak tested, and the licensee did not routinely look for hydraulic actuator fitting leaks. Additionally, the licensee could not identify any vendor information or calculations that demonstrated that the SI 850 valves would not drift closed during their mission time.

The licensee staff believed that there would be ample time for an operator to recognize when an SI-850 valve drifted close from the full open position and take actions to reopen the valve before it adversely affected system flow. However, the licensee could not provide quantitative data to demonstrate the effects of a partial valve closure on sump recirculation flowrate because the rate of valve closure and the amount of closure needed to lose the open indication on the control room panel was not known. The licensee documented this issue in CAP068923. Part of the licensee’s corrective actions for this issue involved a visual examination of the valve actuator limit switches to determine the accuracy of valve status lights in the main control room.

The inspectors noted that the licensee had prior opportunities to identify this design verification testing error. For example, the licensee did not identify the need for these valves to remain open in the update to their DBD for the SI system issued on January 12, 2005, nor in review and revision to the SI system flow model for the SI-850 valves which occurred on November 3, 2005, in support of OPR 162.

Analysis: The inspectors determined that the licensee’s failure to perform testing to demonstrate that the SI-850 valves would remain open for their required mission time following a LOCA was a performance deficiency and a finding. The inspectors concluded that this finding is greater than minor because it affected the design control and the equipment performance attributes of the Mitigating Systems Cornerstone and affected the equipment reliability objective for this cornerstone. Equipment reliability was affected because, as these valves begin to drift shut, the post-LOCA recirculation flow would be affected and require operator actions to compensate for valve drift to ensure adequate long term core cooling. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors answered “yes” to the question in the Mitigating Systems Cornerstone worksheet which asked if the finding was a design or qualification deficiency confirmed to not result in loss of function per NRC Generic Letter 91-18. Therefore, the inspectors determined that this finding was a licensee performance deficiency of very low risk significance (Green).

Enforcement: 10 CFR 50, Appendix B, Criterion XI, “Test Control,” states, in part, that “A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits... The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests...”

Contrary to this, as of November 29, 2005, the licensee had not completed tests (pre-installation, pre-operation nor operational) to demonstrate that the four 1(2)SI-850(A)(B) valves would perform satisfactorily in service, for example, remain open during their design 1 year mission time. Failure to perform adequate testing for these valves is a violation of 10 CFR 50, Appendix B, Criterion XI. Because of the very low safety significance of this finding and because the issue was entered into the licensee’s corrective action program (CAP068923), it is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000266/2005013-03; NCV 05000301/2005013-03).

(2) Inadequate Corrective Actions for Potential Boric Acid Corrosion of SI-850 Valves

Introduction: The inspectors identified a finding involving a non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” having very low safety significance (Green) for the licensee’s failure to implement prompt corrective actions and inspect carbon steel hydraulic operating cylinder components on the 1(2)SI-850(A)(B) valve actuators after becoming aware of the nonconforming and potentially degraded conditions involving boric acid deposits and associated corrosion.

Description: On November 4, 2005, the inspectors completed a walkdown in the Unit 1 and Unit 2 tendon galleries and noted boric acid deposits and evidence of corrosion at the packing area and on the hydraulic actuators for the SI-850 valves (containment sump isolation valves). These NRC observations prompted the licensee to perform an inspection of these valves and document the condition of these valves in CAP068629 and the basis for operability in OPR 165, dated November 6, 2005. The licensee concluded these valves were operable, in part, due to the stainless steel materials used in the packing gland area that were not subject to boric acid corrosion. However, the licensee failed to implement corrective actions to inspect other carbon steel portions of the valve actuator, including the hydraulic actuator connecting rod or closure nuts that were exposed to boric acid deposits or submerged in water and which were not examined during the licensee’s original inspection.

On November 29, 2005, the inspectors’ questions prompted licensee staff to implement another visual examination supplemented with mirrors and a boroscope focused on the hydraulic actuator connecting rod and closure nuts. This inspection identified boric acid deposits at the hydraulic operator connecting rod to valve shaft connection and the licensee performed cleaning to remove these deposits. The licensee recorded this condition in CAP069078. Additionally, the licensee had to remove corrosion and deposits from the hydraulic operator closure nuts to confirm that significant corrosion had not yet occurred. Although the licensee did not identify significant corrosion of these components, the areas inspected were not subject to routine inspection and cleaning. Therefore, the boric acid deposits left on the hydraulic actuator connecting rod may have resulted in corrosion-induced failure of the SI-850 valve actuator. The licensee also entered this issue into the corrective action program, as CAP069111.

<u>Analysis</u>: The inspectors determined that the licensee's failure to promptly identify the boric acid and corrosion of carbon steel portions of the hydraulic actuator as a condition adverse to quality and implement prompt corrective actions to inspect and evaluate these components was a performance deficiency and a finding. The inspectors concluded that this finding is greater than minor because absent NRC intervention, this issue would have become a more significant safety concern. Specifically, the licensee would have allowed an acidic environment (boric acid deposits) or aqueous environment (submerged fasteners) for these carbon steel components to continue for an indefinite period of time which could have resulted in corrosion induced failures of the SI-850 valve actuators . Additionally, the finding is associated with the equipment 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. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors answered "yes" to the question in the Mitigating Systems Cornerstone worksheet which asked if the finding was a design or qualification deficiency confirmed to not result in loss of function per NRC Generic Letter 91-18. Therefore, the inspectors determined that this finding was a licensee performance deficiency of very low risk significance (Green). The cause of this finding was related to the cross-cutting area of problem identification and resolution.

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material, and equipment and nonconformances are promptly identified and corrected." Contrary to this, the licensee was made aware of a potential condition adverse to quality associated with the potential corrosion of the SI-850 valve actuators on November 4, 2005, and did not take prompt corrective actions to identify nonconforming material (for example, boric acid deposits on carbon steel hydraulic actuator connecting rod or corrosion of hydraulic cylinder closure nuts) until prompted by the NRC on November 29. Failure to implement prompt corrective actions for these conditions adverse to quality is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (CAP069111), it is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000266/2005013-04; NCV 05000301/2005013-04).

(3) Incorrect Performance of As-Found Static Lift Test of Valve 2SI-850B

Introduction: The inspectors identified a finding involving an NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” having very low safety significance (Green) for failure to correctly perform a static lift test of the 2SI-850B valve. This test was designed to record the hydraulic actuator pressure necessary to overcome valve dead weight and packing friction.

Description: On November 30, 2005, the inspectors’ questions regarding the results of an as-found static lift test performed for valve 2SI-850B on November 27, 2002, prompted the licensee to recognize that this test had been done incorrectly.

The licensee periodically performed a static lift test of the SI-850 valves in accordance with procedure RMP 9314, “1(2)SI-850A/B Maintenance, Static Test and Adjustment,” to measure the minimum hydraulic force required to overcome valve dead weight (for example, internal shaft and disc weight) and static valve packing friction. The results of the as-found static lift test were used in conjunction with the normal hydraulic operating pressures recorded during the quarterly stroke time test to confirm that adequate thrust margins exist for the valve to open under accident conditions.

On April 11, 2002, (reference CAP004283), the licensee needed to add hydraulic oil to the reservoir supplying the 2SI-850B valve operator. The licensee staff stated that they had suspected an oil leak somewhere on the hydraulic actuator or supply lines but could not find the leak at that time. On November 27, 2002, the licensee identified that valve 2SI-850B was stroking too slow in the open direction to meet the quarterly inservice test acceptance criteria. During subsequent investigation, the licensee identified an oil leak from the hydraulic rod follower packing on the valve actuator. To assess the impact of the as-found condition, the licensee performed a static lift test in accordance with procedure RMP 9314. During this test the licensee applied 1500 pounds per square inch gage (psig) to the hydraulic valve actuator of valve 2SI-850B and the valve failed to open. The licensee also recorded excessive hydraulic fluid leakage during this test (leak was from the closing side of hydraulic operator) which indicated to the inspectors that the static lift test was done improperly (e.g., valve was being driven in shut direction, vice the open direction). Because the licensee had subsequently disassembled, refurbished, and placed the rebuilt hydraulic operator back into service, the capability of the valve to function in the as-found condition could no longer be determined. The inspectors’ questions and conclusions concerning the test data recorded in WO 0216212 prompted the licensee staff to recognize that the as-found static lift test had been performed improperly. The licensee recorded this improperly performed test in (CAP069087).

Analysis: The inspectors determined that the licensee’s failure to correctly perform the as-found static lift test on 2SI-850B on November 27, 2002, was a performance deficiency and a finding. The inspectors concluded that this finding is greater than minor because it was associated with the equipment 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 consequence. Equipment reliability was affected because the incorrectly performed as-found static lift test of 2SI-850B did not provide the information needed to demonstrate the functional capability of this degraded valve. Although no definitive test data existed, the licensee staff believed that this degraded valve would have been functional with the oil leak (400 milliliters lost per closing stroke) because it stroked only 0.5 seconds slow for its open acceptance time during the quarterly stroke test and enough oil existed in the hydraulic reservoir to allow at least 10 open/close cycles. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because the licensee did not consider the valve nonfunctional for past periods of operation with this hydraulic leak, the inspectors answered “yes” to the question in the Mitigating Systems Cornerstone worksheet which asked if the finding was a design or qualification deficiency confirmed to not result in loss of function per NRC Generic Letter 91-18. Therefore, the inspectors determined that this finding was a licensee performance deficiency of very low risk significance (Green).

Enforcement: 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” states, in part, that “A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits...” Contrary to this, on November 27, 2002, for 2SI-850B the licensee failed to complete testing in accordance with procedure steps 5.2 and 5.3 of RMP 9314, “1(2)SI-850A/B Maintenance, Static Test and Adjustment.” Specifically, step 5.2, which provided instructions on where and how to connect a manual hydraulic pump to the hydraulic valve actuator, and step 5.3, to open the valve, were not correctly followed as evidenced by the 1500 psig high pressures achieved (well above the pressures normally required to open the valve) and leakage from the closing side of the hydraulic valve operating cylinder. Consequently, the valve did not open and erroneous test data were recorded. Failure to perform the as-found static lift test for this valve in accordance with RMP 9314 is a violation of 10 CFR 50, Appendix B, Criterion XI. Because of the very low safety significance of this finding and because the issue was entered into the licensee’s corrective action program (CAP069087), it is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000301/2005013-05).

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

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

.3 Resident Inspector Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of licensee trending activities to determine if emerging adverse trends might indicate the existence of a more significant safety issue not previously identified. The inspectors also determined whether the trends were entered into the licensee's corrective action system at an appropriate threshold, and timely corrective actions were planned or implemented by the licensee. The effectiveness of licensee trending activities was assessed by comparing trends identified by the licensee with those trends identified by the NRC during the daily reviews of CAPs, as discussed in Section 4OA2.2 of this report.

The inspector's review considered the 6-month period of July 2005 through December 2005, although some examples extended beyond those dates when the scope of the trend warranted. The inspectors also reviewed the Department Roll-Up Meeting Reports and Quarterly Department Roll-Up Meeting Summary from January 2005 through December 2005. Finally, the inspectors reviewed the third and fourth quarter 2005 human performance trend reports. The inspector's review was focused on licensee human performance errors, but also considered the results of daily inspector corrective action program item screening, licensee trending efforts, and licensee human performance results. This inspection effort constituted one semi-annual trending inspection procedure sample.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

.1 Follow-Up on Corrective Actions Taken for Failure to Report Issues in a Timely Manner

a. Inspection Scope

The inspectors reviewed the licensee's corrective actions following the inspectors identification of the licensee's repetitive failure to report non-emergency events within 8 hours, in accordance with 10 CFR 50.72(b)(3)(ii)(B). Included in this review was an evaluation of the licensee's initial notification, and the licensee's corrective action documents associated with the first two inspector identified occurrences in April and September 2005.

b. Findings

Introduction: The inspectors identified an NCV, with three examples, for the failure to notify the NRC within 8 hours in accordance with 10 CFR 50.72(b)(3)(ii)(B), following the identification that the nuclear power plant was in an unanalyzed condition that significantly degraded plant safety. Each occurrence was reported by the licensee following repeated questioning by the inspectors which occurred in April, September, and November 2005. Following the November occurrence, the inspectors reviewed the licensee's previous causal evaluations and corrective actions. The inspectors noted that while the licensee had appropriately evaluated and initiated corrective actions for the technical issues in April and September 2005, the licensee had not appropriately evaluated or developed any corrective actions to address the failure to adequately report these issues to the NRC in a timely manner.

Description: On April 8, 2005, CAP063467 was written when the licensee identified an unanalyzed condition associated with the licensee's current Appendix R Safe Shutdown Analysis. The issue dealt with control and power cables for the three Unit 1 charging pumps routed through a fire area and the discovery that one of the charging pumps was credited as being used for shutting down the reactor in this fire area. Following the daily review of CAPs, the inspectors questioned the licensee as to whether this event was reportable in 8 hours, in accordance with 10 CFR 50.72(b)(3)(ii)(B). On April 28, CAP063467 was written for a similar issue on Unit 2, and the residents continued to question the licensee as to whether the events were reportable. The 8-hour report was made by the licensee on June 8 in Event Notification Number 41754.

On September 22, 2005, CAP067167 was written when the licensee identified a calculation from the bolted-fault calculation project that showed equipment would experience short circuit fault currents greater than equipment ratings. These electrical issues directly affected the licensee's Appendix R analyses, which relied on breaker coordination and fault current interruption to prevent the loss of safe shutdown equipment due to common enclosure/power supply associated circuit concerns. Following the daily review of CAPs, the inspectors questioned the licensee as to whether this event was reportable in 8 hours, in accordance with 10 CFR 50.72(b)(3)(ii)(B). The 8-hour report was made by the licensee on September 27 in Event Notification Number 42020.

On October 27, 2005, CAP068373 was written by the licensee to document that the inspectors' identification that containment coatings in both units were not maintained with the licensee's current analysis and that the licensee's calculations utilized to derive head loss across the containment sump screens contained errors. The inspectors questioned the licensee as to whether this event was reportable in 8 hours, in accordance with 10 CFR 50.72(b)(3)(ii)(B). The 8-hour report was made by the licensee on November 8 in Event Notification Number 41754.

Following this third instance, the inspectors reviewed CAP067656, "Timeliness in Determining Regulatory Reportability," written by the licensee in response to the first two untimeliness issues. The inspectors also reviewed the previous condition reports written for the technical issues. Upon reviewing Condition Evaluation CE016321, the inspectors determined that the evaluation incorrectly concluded that the requirements of

10 CFR 50.72 were met for the first two instances. As a result, the licensee took no further corrective actions. Following discussions with the inspectors, the licensee agreed that the previous Condition Evaluation was inadequate and initiated CAP068398.

Analysis: The inspectors determined that this issue constituted a performance deficiency because the licensee initially failed to report these unanalyzed conditions which significantly degraded plant safety in a timely manner, and did not take corrective actions following the first two instances identified by the NRC. This issue is greater than minor because the failure to report unanalyzed conditions in a timely affected the NRC's ability to perform a regulatory function. Because violations of 10 CFR 50.72 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process. The finding has been reviewed by NRC management and is determined to be a Green finding of very low safety significance.

The inspectors noted that while the licensee had appropriately evaluated and initiated corrective actions for the technical issues in April and September 2005, the licensee had not appropriately evaluated or developed any corrective actions to address the failure to adequately report these issues in a timely manner to the NRC. Therefore, the inspectors also determined that a primary cause of this finding was related to the cross-cutting area of problem identification and resolution.

Enforcement: 10 CFR Part 50.72(b)(3)(B)(ii), "Non-Emergency Events - 8 Hour Reports," requires, in part, that upon the occurrence of the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety, the licensee shall notify the NRC as soon as practical and in all cases within 8 hours. Contrary to this, the licensee failed to report within 8 hours the following occurrences where the licensee discovered the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety: (1) On April 8, 2005, the licensee identified an unanalyzed condition associated with the licensee's current Appendix R Safe Shutdown Analysis; however, this was not reported to the NRC until June 7,2005; (2) On September 22, 2005, the licensee identified a separate unanalyzed condition associated with the licensee's current Appendix R Safe Shutdown Analysis; however, this was not reported to the NRC until September 27, 2005; and (3) On October 27, 2005, the licensee identified an unanalyzed condition in response to inspector questions concerning licensee calculations regarding the emergency core cooling system and long-term reactor core cooling; however, this was not reported to the NRC until November 8, 2005.

The finding is not suitable for SDP evaluation, but has been reviewed by NRC management and determined to be a finding of very low safety significance (Green). Because this violation is of very low safety significance and because the licensee entered the issue into its corrective action program (as CAP068938), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2005013-06; NCV 05000301/2005013-06).

The licensee performed a causal evaluation and determined that a lack of understanding with respect to operability and reportability, a lack of understanding of the "time of discovery" as defined in NRC NUREG-1022 ("Event Reporting Guidelines

10 CFR 50.72 and 50.73," Revision 2) and a lack of understanding of reportability requirements for issues not covered by TS as the causes of this finding. The licensee developed approximately 14 corrective actions which included lessons learned, training of licensee staff on an initial and continuing cycle with respect to reportability and also procedure revisions to correct this violation.

.2 <u>(Closed) LER 05000266/2005003-00</u>, Potentially Inoperable Safety Injection Pumps

On July 15, 2005, the licensee identified a potential condition that during a design basis accident with degraded safeguards bus voltage, the Unit 1 "A" SI pump motor could trip and lockout on overcurrent prior to the associated safeguards bus stripping on undervoltage. The SI pump lockout could then prevent an automatic start of the SI pump during emergency diesel generator load sequencing. The Unit 1 "A" SI pump was declared out-of-service. The time overcurrent setpoint was subsequently adjusted to correct this condition for the Unit 1 "A" SI pump motor. The Unit 2 Train "A" SI pump motor time overcurrent setpoint was also reset. Further investigation by the licensee revealed a similar condition for the Unit 1 and Unit 2 "B" SI pump motors which were declared out-of-service and time overcurrent setpoints were also adjusted.

Based on the combined effect of all SI pump motor time overcurrent setpoints, a condition existed which could have impacted the design function of Unit 1 or Unit 2 SI pumps during a design basis accident with degraded safeguards bus voltage. Both units' SI pumps had the potential to not automatically start on the emergency diesel generator loading sequence. The enforcement aspects of this licensee identified violation are discussed in Section 4OA7 of this inspection report. This LER is closed.

.3 <u>(Closed) LER 05000266/2005004-00</u>, Auxiliary Feedwater Recirculation Capability in Local Operating Mode

On July 19, 2005, with both units at normal full power operation, a test start of the motor-driven auxiliary feedwater (AFW) pumps, P-38A and P-38B, was being performed from the local control station. During this activity it was discovered that valve AF-4007, the recirculation valve for P-38A, would not automatically open in the local mode of operation. The test was suspended and the pump was declared out-of-service. The licensee's investigation determined that taking the pumps to local control at cabinets -01 and -02 disabled the expected opening of the recirculation valve when the pump was started. The cause was a deficient procedure caused by inadequate review of abnormal operating procedure AOP-10, "Control Room Inaccessibility," Revision 3, during development. The licensee took immediate action to revise AOP-10. This event was previously evaluated in NRC Special Inspection Report 05000266/2005011; 05000301/2005011. This LER is closed.

.4 <u>(Closed) LER 05000301/2005001-00</u>, Main Steam Safety Valve 2MS-02008 Lift Set Point Exceeds Acceptance Criteria

On April 20, 2005, Point Beach Nuclear Plant, Unit 2 was in a routine refueling shutdown. Three main steam safety valves were removed for inservice testing and replaced with similar valves of identical set pressure and capacity. During testing of 2MS-02008, the valve exceeded the lift setpoint acceptance criterion. The cause for the failure was indeterminate. Documentation received with the vendor's Certificate of Conformance, dated September 22, 2005, provided no indications as to why the valve failed. All parts were reported to be in good condition and the seating areas were clean. The failure of 2MS-02008 is a violation of TS SR 3.7.1.1, setpoint lift pressure criteria. The licensee concluded that the safety significance of operating with the setpoint above the limit was minimal. The inspectors reviewed and verified the analysis and identified no concerns. While this is a violation of NRC requirements, no licensee performance deficiency, as defined in NRC Inspection Manual Chapter 0612, dated September 30, 2005, was identified and the issue is considered minor. This LER is closed.

.5 (Closed) Unresolved Item (URI) 05000266/2005004-06; 05000301/2005004-06; Unanalyzed Condition Due to Appendix R Safe Shutdown Strategy Deficiency

On June 7, 2005, the licensee reported a condition under Title 10 CFR Part 50.72(b)(3)(ii)(B) as a result of an ongoing evaluation of a previously identified deficiency with the Appendix R Safe Shutdown Strategy with respect to the use of charging pumps for a fire in Fire Area A06, Bus 1B-32 480-Volt Motor Control Center Area. The licensee took immediate corrective actions to mitigate the consequences of this issue.

The inspectors are closing this Unresolved Item to the pending NRC review of the licensee's evaluations associated with the 10 CFR Part 50.73 report documented in LER 05000266/2005-002-00; 05000301/2005-002-00 for this same issue.

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000266/2005011-01; 05000301/2005011-01, Potential Vulnerability of Auxiliary Feedwater (AFW) Recirculation Line

During an inspection from July 25 through August 24, 2005, the inspectors reviewed the licensee's corrective actions to prevent recurrence of several potential AFW system common mode failures. Among the potential common mode failures was crimping of the common AFW recirculation line from where it exited the seismic AFW pump room and traversed a portion of the nonseismic turbine building to the nonseismic condensate storage tanks. A subsequent detailed design review by a licensee's contractor team in mid-2003 also indicated that crimping of the AFW recirculation line was a potential common mode failure. However, the licensee eventually concluded that crimping of this line was not credible.

The inspectors determined that, since the licensee was crediting the non-safety related AFW recirculation with performing a safety function (to pass required AFW pump recirculation flow) in a non-safety related area, crimping of the recirculation line should be evaluated as a failure mode unless it could be shown that the line was not susceptible to crimping. The licensee documented the inspectors' concern in CAP066199 and subsequently formally evaluated the issue, Engineering Evaluation 2005-0012, "Auxiliary Feedwater Recirculation Line Crimping Evaluation," dated August 17, 2005, and OPR000148, dated August 19, 2005.

From their initial review of these documents, the inspectors identified several errors and several questions, which were communicated to the licensee. Because the licensee was still addressing the errors and questions by the end of the inspection, the concern of crimping of the AFW recirculation line was tracked as a URI.

a. Inspection Scope

During the current inspection, from December 12 through 16, 2005, the inspector reviewed the licensee's corrective actions for the crimping concern. Specifically, the inspector reviewed the licensee's corrective action documents, a design calculation, an engineering evaluation, the component maintenance program administrative document for relief/safety valves, relief valve AF-4035 procurement documents, and relief valve AF-4035 test reports. The inspector also verified that the seismic reinforcement for the Work Control Center (Operations Building) south wall was installed as analyzed in the design calculation and that recirculation line relief valve AF-4035 was added to the component maintenance program to ensure periodic pressure testing.

b. Findings

Introduction: On December 16, 2005, the inspector identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," of very low safety significance (Green). Specifically, the licensee failed to promptly correct a condition adverse to quality, the potential for the AFW recirculation line to crimp during a DBE or DBT event. The licensee missed prior opportunities to correct the adverse condition: 1) as a result of the two Red findings related to the AFW system, the licensee reviewed the AFW system for the effects of high energy line break, DBE, and DBT events and identified crimping of the non-safety related portion of the common AFW recirculation line as a potential common mode failure; and 2) an external self-assessment in mid-2003 also concluded that crimping of the AFW recirculation line was credible and a potential common mode failure.

Discussion: As a result of two Red findings associated with the AFW system, the licensee's corrective action process identified several potential AFW system common mode failures. Among the potential common mode failures was crimping of the common AFW recirculation line from where it exited the seismic AFW pump room and traversed a portion of the nonseismic AFW pump room and traversed a portion of the nonseismic turbine building to the nonseismic condensate storage tanks. A subsequent detailed design review by a licensee's contractor team in mid-2003 also indicated that crimping of the AFW recirculation line was a potential common mode failure.

In Condition Evaluation CE011367, the licensee addressed the common mode failure concern of the AFW recirculation piping crimping and concluded, in-part:

> The possibility of blockage of the AFW minimum flow recirculation piping from either a seismic or tornado event is considered beyond the PBNP licensing and design basis. The likelihood of occurrence of this event is negligible. It has been determined that the turbine building will not fail in a seismic event. The likelihood of a tornado missile striking the recirculation line and causing a blockage is extremely low. Furthermore,

relief valve AF-4035 is capable of providing AFW recirculation flow in the highly unlikely event that the recirculation line is blocked.

During an NRC special inspection, from July 25 through August 24, 2005, the inspectors were concerned that, since the licensee was crediting the non-safety related AFW recirculation with performing a safety function (to pass required AFW pump recirculation flow) in a non-safety related area, crimping of the recirculation line should be evaluated as a failure mode unless it could be shown that the line was not susceptible to crimping. The licensee documented the inspectors' concern in CAP066199 and subsequently formally evaluated the issue, Engineering Evaluation 2005-0012, "Auxiliary Feedwater Recirculation Line Crimping Evaluation," dated August 17, 2005, and OPR000148, dated August 19, 2005. During their initial review of these documents, the inspectors identified several errors and several questions, which were communicated to the licensee and tracked as URI 05000266/ 205011-01; 05000301/2005011-01.

From December 12-16, 2005, the inspector reviewed the licensee's corrective actions associated with the URI and actions taken (OBD000274 corrective action plan) to restore the operable but degraded/non-conforming item (the potential for AFW pump recirculation line crimp resulting from a seismic or tornado event) to full qualification. These actions included:

- Reinforce the work control center (the operations building) south wall to withstand effects of a design basis earthquake.

  The inspector reviewed the associated design calculation and determined that the seismic reinforcement for the south wall was installed as analyzed in the design calculation.

- Revise Engineering Evaluation 2005-0012, "Auxiliary Feedwater Recirculation Line Crimping Evaluation."

  The inspector reviewed the evaluation assumptions, engineering judgements, and evaluations used to conclude that damage to the AFW pumps caused by crimping of the AFW recirculation line as a result of a DBE, DBT, or high energy line break was not credible. The evaluation identified the potential failure of the work control center or condensate storage tanks due to a DBT as the credible concerns (work control center south wall was reinforced to withstand a DBE) but concluded that "complete blockage of the AFW recirculation line is not possible." The inspector could not independently verify that complete recirculation line blockage was "not possible" due to the uncertainty associated with postulating structural failure mechanisms and structural interactions that could occur during a DBT.

- Replace AFW pump recirculation line relief valve AF-4035 and bench test.

  The inspector reviewed the licensee's test report and determined that the original AFW recirculation line relief valve would have opened as designed and would have provided the required AFW recirculation flowpath if the AFW recirculation line crimped during a DBE or DBT event.

The inspector also reviewed licensee documentation for the replacement AFW pump recirculation line relief valve AF-4035 and determined that the valve was procured to meet ASME Code Section VIII requirements, sized to provide the required AFW recirculation flowpath if the AFW recirculation line crimped during a DBE or DBT event, and pretested by the manufacturer to demonstrate that the relief valve would actuate at the specified set pressure.

- Add recirculation line relief valve AF-4035 to Component Maintenance Program CMP 2.3, "Relief/Safety Valves," and establish the appropriate testing frequency.

  The inspector reviewed licensee documentation to replace and test recirculation line relief valve AF-4035 including: Callup ID M-RELVLV for AF-04035, OBD000293 to revise the classification of AF-04035 in CMP 2.3, and CMP 2.3 testing frequency requirements and acceptance criteria for the engineering judgement valve classification. The inspector determined that the CMP 2.3 testing frequency requirements for the engineering judgement valve classification was consistent with the ASME OM Code for Class 2 and Class 3 valves.

The inspector determined that the licensee's commitment to periodically replace AFW recirculation line relief valve AF-4035 with a pretested valve provided reasonable assurance that AF-4035 would provide the required flowpath to protect the AFW pumps if the AFW recirculation line crimped during a DBE or DBT event.

The inspector also determined that the licensee failed to promptly identify that crimping of the AFW recirculation line during a DBE or DBT event was credible and to correct this condition adverse to quality. The licensee planned to supplement CAP066199 to address the inadequate corrective actions.

Analysis: The inspectors determined that a performance deficiency existed because it was reasonably within the licensee's control to have promptly identified that crimping of the AFW recirculation line during a DBE or DBT event was credible and to correct this condition adverse to quality.

The inspector determined that the finding is greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because it was associated with the equipment 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. Additionally, if left uncorrected, the finding would become a more significant safety concern. Specifically, if left uncorrected the AFW recirculation line relief valve could have deteriorated over time, failed to open as designed, and not provided the required recirculation line flowpath to protect the AFW pumps if the recirculation line crimped during a DBE or DBT event.

The inspector determined the finding was of very low safety significance (Green) because testing of the original AFW recirculation line relief valve demonstrated that the relief valve would have opened as designed and would have provided the required AFW recirculation flowpath if the AFW recirculation line crimped during a DBE or DBT event.

The inspectors also determined that a primary cause of this finding was related to the cross-cutting area of problem identification and resolution, because the licensee failed to correct the condition adverse to quality involving the potential for the auxiliary feedwater recirculation line to crimp during a DBE or DBT event which was first identified in 2003.

Enforcement: Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B, requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this, from mid-2003, when the licensee and the licensee's external self-assessment identified crimping of the non-safety related portion of the common AFW recirculation line as a potential common mode failure, until November 17, 2005, when the licensee had replaced and pressure tested the original AFW recirculation line relief valve and had commitments in place to periodically replace and pressure test the AFW recirculation line relief valve to provide the required flowpath to protect the AFW pumps if the AFW recirculation line crimped during a DBE or DBT event, the licensee failed to correct this condition adverse to quality.

Because this issue was of very low safety significance and because the issue will be entered into the licensee's corrective action program as a supplement to CAP066199, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2005013-07; 05000301/2005013-07). Unresolved Item URI 05000266/2005011-01; 05000301/2005011-01 is closed.

.2 RVCH and CRDM Housing Replacement (IP 71007)

The penetration nozzles on the original Unit 1 RVCH were fabricated from Inconel Alloy 600 material. These nozzles were welded to the RVCH with a partial penetration weld fabricated from Inconel Alloy 182 weld filler metal. In recent years, several pressurized water reactors have experienced pressure boundary leakage caused by primary water stress corrosion cracking of these materials.

The design of the Unit 1 replacement RVCH is similar to the original, with some notable exceptions as follows:

- New RVCH is constructed from a single piece forging which eliminates the dome-to-flange weld
- New CRDM housing design eliminates vents and seal welds
- New RVCH design eliminates the unused control rod penetrations
- Use of Inconel Alloy 600 was prohibited in fabrication of the new RVCH. For example, the penetration tube material was changed from Inconel Alloy 600 to Inconel Alloy 690 which is more resistant to primary water stress corrosion cracking.

a. Inspection Scope

From September 12 through October 14, 2005, the inspectors reviewed the licensee's design changes associated with the RVCH replacement efforts. This review included certified design specifications, certified design reports, ASME Code reconciliation reports, fabrication deviation notices, non-conformance reports, and design calculations to confirm that the replacement RVCH and CRDM housings were in compliance with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subsection NB (1998 Edition including addenda through 2000 Addendum). Specifically, the inspectors confirmed that the design specifications and design reports were certified by registered professional engineers competent in ASME Code requirements. The inspectors confirmed that adequate documentation existed to demonstrate the certifying registered professional engineers were qualified in accordance with the requirements of the ASME Code Section III (Appendix XXIII of Section III Appendices). The inspectors also confirmed that the replacement RVCH and CRDM housings were provided as Code NPT stamped components.

The inspectors also reviewed a sample of nondestructive examination and preservice examination records and weld repair records to confirm that the activities were performed in accordance with the ASME Code and NRC Order EA-03-009, issued February 11, 2003, establishing interim inspection requirements for reactor pressure vessel heads at pressurized water reactors. The sample included review of radiographic film and records for eddy current, ultrasonic, and liquid penetrant examinations.

b. Findings

No findings of significance were identified.

.3 Head Assembly Upgrade Package (HAUP) (71007)

During the fall 2005 Unit 1 refueling outage (U1R29), the licensee elected to install a reactor HAUP that integrated the design of various plant components and structures into the reactor head assembly. This integration involved the reuse of some plant components and the complete replacement of others including:

- New CRDM cooling system
- New integral reactor vessel missile shield
- Reactor vessel head lift rig
- CRDM seismic spacer plates
- Cable drawbridge with connector panels
- New RVLIS pipe supports
- New RCGVS pipe supports
- Handrail modifications and new ladders
- New cable supports on head platform and lift legs
- Reactor vessel head insulation

a. Inspection Scope

From September 12 through October 14, 2005, the inspector reviewed the licensee’s design documentation associated with the installation of the Unit 1 HAUP. Specifically, the inspectors reviewed the design specification and a representative sample of design calculations to confirm that HAUP structures and components were designed in accordance with the requirements of the HAUP design specification and the American Institute of Steel Construction (AISC) and ASME design codes.

b. Findings

No findings of significance were identified.

.4 Implementation of Reactor Vessel Closure Head and CRDM Housing Replacement During the Refueling Outage (IP 71007)

a. Inspection Scope

As part of the Unit 1 refueling outage, the licensee replaced the RVH and CRDM housings. The inspectors conducted inspections of these activities and in some cases performed inspections of specific implementation activities as samples of other inspection procedure modules. In addition, as part of this inspection the inspectors reviewed the following activities associated with the Reactor Vessel Closure Head replacement:

- Temporary modifications associated with this modification
- Applicable engineering design, modification, and analysis associated with RVH lifting and rigging, including: (1) crane and rigging equipment and full load testing; (2) RVH drop analysis; (3) safe load paths; (4) lay-down areas
- Controls and plans to minimize adverse impact on the operating unit and common systems
- Activities associated with lifting and rigging, which involved preparations and procedures for rigging and heavy lifting including any required crane and rigging inspections, testing, equipment modifications, lay-down area preparations, and training of crane and rigging personnel
- Documentation associated with the lifting equipment to handle the loads
- Establishment of the appropriate operating conditions for the various activities associated with the modification
- Testing programs for components which were reinstalled from the old RVH;
- Controls for excluding foreign materials
- Licensee's post-installation inspections and verifications program, including implementation
- Conduct of RCS leakage testing and reviewed test results
- Procedures for equipment performance testing required to confirm the design and to establish baseline measurements

b. Findings

No findings of significance were identified.

.5 Temporary Instruction 2515/161 - Transportation of Reactor Control Rod Drives In Type A Packages

a. Inspection Scope

The inspectors conducted interviews and reviewed shipment logs to verify that: (1) the licensee had undergone refueling activities since calendar year 2002; and (2) did not ship irradiated control rod drive mechanisms in Department of Transportation Specification 7A, Type A packages during the time frame of 2002 to the present.

b. Findings

No findings of significance were identified.

.6 Temporary Instruction 2515/163 - Operation Readiness of Offsite Power

Earlier in 2005, the inspectors reviewed the operational readiness of the licensee's offsite power systems. This review was documented in Section 4OA5.4 of Inspection Report 05000266/2005004; 05000301/2005004. As a followup to that effort, the inspectors reviewed with the licensee the compensatory actions the control room operators would perform if the offsite transmission system operator was not able to predict the post-reactor trip voltage at Point Beach for current grid conditions. This issue had been entered into the licensee's corrective action program as CAP065676, corrective actions which involved a revision of the communication and mitigation protocols for the interface between Point Beach and the offsite grid administrator. This information has been forwarded to the Office of Nuclear Reactor Regulation for review.

.7 Followup of April 21, 2004, Confirmatory Action Letter

a. Inspection Scope

The inspectors reviewed the licensee's efforts in implementing a commitment made as part of a April 21, 2004, Confirmatory Action Letter. The commitment was to perform a progress review of procedures to be developed or revised associated with an effort to improve the configuration management program. The commitment was designated as Step 16.A of Action Plan OP-14-001, one of several improvement Action Plans that were incorporated into the Confirmatory Action Letter.

b. Findings

The licensee's performance measure for successful completion of this step was to review and revise at least 40 procedures by the second quarter of 2005. The progress review identified that 45 of the 54 procedures within the scope of this Action Plan (83 percent) had been reviewed. Seventeen of the 45 were determined to not require revision, 6 were cancelled, and 22 had been revised as of May 2005.

The inspectors reviewed a sample of four procedure revisions and found that the revisions appeared to strengthen the configuration management controls at Point Beach. The inspectors concluded that the licensee completed the Action Plan Action Step as committed in the March 22, 2004, letter that was incorporated into the CAL. The inspectors did not identify any significant problems with actions taken to complete this step.

4OA6 Meetings

.1 Exit Meeting

On January 18, 2006, the resident inspectors presented the inspection results to Mr. D. Koehl, the Point Beach Site Vice-President, and members of his staff, who acknowledged the findings. The licensee did not identify any information, provided to or reviewed by the inspectors, as proprietary in nature.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Occupational radiation safety access control/ALARA program inspection during the licensee's Unit 1 refueling outage and the radiological environmental monitoring program with Mr. D. Koehl and other licensee staff on October 21, 2005
- ISI procedure (IP 71111.08) with Ms. F. Flentje and other members of the licensee's staff on November 23, 2005. The licensee confirmed that none of the potential report input discussed was considered proprietary.
- Reactor vessel head replacement procedure (IP 71007) and evaluation of changes, tests, and experiments procedure (IP 71111.02) with Mr. D. Koehl and other members of the licensee's staff on October 14, 2005. The licensee confirmed that the design documentation prepared by their contractors was considered proprietary. It was agreed that copies of all proprietary documents not returned to the licensee would be shredded.
- Emergency Preparedness inspection with Ms. M. Ray on November 29, 2005
- Public radiation safety radioactive waste processing and transportation program inspection with Mr. D. Koehl and other licensee staff on December 9, 2005
- Closure of URI 0500266/2005011-01; 05000301/2005011-01, "Potential Vulnerability of AFW Recirculation Line," with Mr. D. Koehl and other members of the licensee's staff on December 16, 2005. The licensee did not identify any information, provided to or reviewed by the inspector and likely to be included in the inspection report, as proprietary.

4OA7 Licensee-Identified Violations

Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated in specifications, drawings, procedures and instructions. Contrary to the design of the motor protection scheme, on July 15, 2005, the licensee identified a potential condition that during a design basis accident with the degraded safeguards bus voltage, the Unit 1 “A” SI pump motor could trip and lockout on overcurrent prior to the associated safeguards bus stripping on undervoltage. The SI pump lockout could then prevent an automatic start of the SI pump during emergency diesel generator load sequencing. A similar condition was identified for all SI pump motors in both units. This issue was documented in the licensee’s corrective action program as CAP065765 and CAP066104. The finding was considered to have very low safety significance (Green) using Phase II of the SDP for Loss of Offsite Power during a design basis accident because the SI pumps could be started manually and emergency operating procedures and training exist to start the pumps manually from the control room under these conditions.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

R. Amundson, Training Supervisor Operations
G. Casadonte, Fire Protection Coordinator
B. Cole, Internal Assessment Supervisor
G. Corell, Chemistry Manager
Y. Elsen, RRVCH Document Control Supervisor
F. Flentje, Senior Regulatory Compliance Engineer
S. Forsha, Engineer, Nuclear Oversight
T. Gemskie, Emergency Preparedness Supervisor
B. Grazio, Regulatory Affairs Manager
L. Hawki, Engineering Supervisor
C. Hill, Assistant Operations Manager
B. Jensen, Level III
C. Jilek, Maintenance Rule Coordinator
R. Johnson, Senior Emergency Preparedness Coordinator
K. Kappelman, Emergency Preparedness Instructor
T. Kendall, Engineering Senior Technical Advisor
D. Koehl, Site Vice-President
R. Ladd, Fire Protection Engineer
G. LeClair, Radwaste Supervisor
M. Lorek, Plant Manager
J. McCarthy, Director of Site Operations
J. McNamara, Engineering Supervisor
C. Onesti, Senior Health Physicist
G. Packard, Operations Manager
L. Peterson, Design Engineer Manager
M. Ray, Emergency Planning Manager
G. Ridder, Senior Engineer - RRVCH Project
D. Schuelke, Radiation Protection Manager
J. Schweitzer, Site Engineering Director
D. Shannon, General Supervisor, Radiation Support
G. Sherwood, Engineering Programs Manager
C. Sizemore, Training Manager
P. Smith, Licensed Operator Requalification Training Group Lead
W. Smith, Site Assessment Manager
J. Strharsky, Planning and Scheduling Manager
N. Stuart, Maintenance Manager
J. Tabat, Responsible Engineer, RRVCH Project
R. Turner, Inservice Inspection Coordinator
P. Wild, Design Engineering Projects Supervisor
R. Womack, Fleet Program Engineering Manager

Nuclear Regulatory Commission personnel

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000266/2005013-01; 05000301/2005013-01	NCV	Failure to Perform An Equivalency Evaluation on a Safety-Related Motor (Section 1R19)
05000266/2005013-02	NCV	Failure to Adequately Implement Procedures Related to Containment Debris near ECCS Sump (Section 1R20.1)
05000266/2005013-03; 05000301/2005013-03	NCV	Inadequate Design Verification Testing of SI 850 Valves (Section 4OA2.1b.(1))
05000266/2005013-04; 05000301/2005013-04	NCV	Inadequate Corrective Actions for Potential Boric Acid Corrosion of SI-850 Valves (Section 4OA2.1b.(2))
05000301/2005013-05	NCV	Incorrect Performance of As-Found Static Lift Test of Valve 2SI-850B (Section 4OA2.1b.(3))
05000266/2005013-06; 05000301/2005013-06	NCV	Multiple Examples of the Failure to Notify the NRC within 8 hours as Required by 10 CFR 50.72 (Section 4OA3.1)
05000266/2005013-07; 05000301/2005013-07	NCV	Failure to Promptly Correct Potential Crimping Vulnerability of AFW Recirculation Line (Section 4OA5.1)

Closed

05000266/2005004-06; 05000301/2005004-06	URI	Unanalyzed Condition Due to Appendix R Safe Shutdown Strategy Deficiency (Section 4OA3.5)
05000266/2005011-01; 05000301/2005011-01	URI	Potential Vulnerability of AFW Recirculation Line (Section 4OA5.1)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather

CAP068839; Outstanding Cold Weather Issues
CAP068818; U1 Facade SW piping - No insulation
CAP067207; FF Heat Trace WO completion will likely miss Oct. 1 PC49 Date
CAP067206; WO04085999, U2R27 WO, not complete affecting cold weather preps
CAP068719; Cold Weather Concern - 2MS-272 Blowdown Valve
PC 49; Cold Weather Preparations; Revision 5
OI 106; Facade Freeze Protection; Revision 21

Section 1R02: Evaluation of Changes, Tests, or Experiments

SCR 2005-0037; 10 CFR 50.59/72.48 Screening for MR 03-047 - Replacement of Unit 1 Reactor Vessel Closure Head; dated June 23, 2005
EVAL 2005-002; 10 CFR 50.59 Evaluation for MR 03-047 - Replacement of Unit 1 Reactor Vessel Closure Head; dated June 28, 2005
Westinghouse Letter LTR-MPG-05-17; Subject: NMC - Point Beach Unit 1 - Replacement RV Head - 10CFR50.59 Input - EVAL-05-24; dated March 28, 2005
SCR 2005-0060; 10 CFR 50.59/72.48 Screening for MR 03-046 - Install Jib Cranes in Unit 1 Containment; dated September 7, 2005
SCR 2005-0036; 10 CFR 50.59/72.48 Screening for MR 03-048 - Replacement of the Unit 1 Reactor Head Metallic Reflective Insulation; dated 1, 2005
SCR 2005-0059; 10 CFR 50.59/72.48 Screening for MR 03-049 - Reactor Vessel Head Assembly Upgrade Package - Unit 1; dated September 12, 2005
Westinghouse Letter LTR-MPG-05-18; Subject: NMC - Point Beach Unit 1 - Head Assembly Upgrade Package (HAUP) - 10CFR50.59 Evaluation, Accident Analysis Input - EVAL-05-25; dated May 2, 2005
SCR 2005-0042; 10 CFR 50.59/72.48 Screening for MR 03-050 - Unit 1 Containment Equipment Hatch Shield Wall Modification - Unit 1; dated August 29, 2005
SCR 2005-0055; 10 CFR 50.59/72.48 Screening for MR 03-051 - Analog Rod Position Indicator Cable and Connector Modification - Unit 1; dated August 26, 2005
SCR 2005-0056; 10 CFR 50.59/72.48 Screening for MR 03-052 - CRDM Cable and Connector Modification - Unit 1; dated August 31, 2005
SCR 2005-0012; 10 CFR 50.59/72.48 Screening for MR 03-053 - Core Exit Thermocouple System Mineral Insulated Cable Upgrade - Unit 1; dated February 14, 2005
10 CFR 50.59/72.48 Pre-Screening Review; Changes Due to Revision 29 of 1RMP 9096, Reactor Vessel Head Removal and Installation; dated September 20, 2005
10 CFR 50.59/72.48 Pre-Screening Review; Revision to TRM 3.9.4, Reactor Vessel Head Lift; dated September 23, 2005

Section 1R05: Fire Protection

Drawing PB 01EFPL06200100 Electrical Layout Fire Detection System Unit 1 Cont (EL 8' 0"), FPE - 001
Drawing PB 01EFPL06200200 Electrical Layout Fire Detection System Unit 1 Cont (EL 26' 0"), FPE - 001
Drawing PB 01EFPL06200300 Electrical Layout Fire Detection system Unit 1 Cont (EL 44' 0"), FPE - 003

Drawing PB 31 MFPL00015216 Fire Protection For Turbine Building, Aux Building & Containment Elev. 8' 0"
Drawing PB 31 MFPL00015309 Fire Protection For Turbine Building, Aux Building & Containment Elev. 26' 0"
Drawing PB 31 MFPL00015407 Fire Protection For Turbine Building, Aux Building & Containment Elev. 44' 0"
Drawing PB 31 MFPL00015605 Fire Protection For Turbine Building, Aux Building & Containment Elev. 66' 0"

Section 1R06: Flood Protection Measures
CAP062251; TS Bases Inconsistent With Design/License Basis for Facility
CAP062162; Flooding Concerns With AFW Pump Room
DBD-T-41 MODULE A, Hazards - Internal and External Flooding, Revision 4
NP 8.4.17 PBNP Flooding Barrier Control, Revision 3
AOP-9A Service Water System Malfunction, Revision 21

Section 1R08: Inservice Inspection Activities
Ultrasonic Calibration Record PBNP U1R29; RC-03-BP-10001-06; dated October 6, 2005
Ultrasonic Calibration Record PBNP U1R29; RC-03-BP-10001-07; dated October 6, 2005
Ultrasonic Calibration Record PBNP U1R29; RC-03-BP-10001-08; dated October 6, 2005
Ultrasonic Calibration Record PBNP U1R29; RC-03-BP-10001-09; dated October 6, 2005
Ultrasonic Calibration Record PBNP U1R29; RC-03-BP-10001-11; dated October 6, 2005
1007904; Steam Generator In Situ Pressure Test Guidelines; Revision 2
ISI-1131; ISI Isometric PBNP Unit 1 Pressurizer Safety; dated November 12, 1990
ISI-1135; ISI Isometric PBNP Unit 1 Pressurizer Power Operated Relief Valves; dated November 12, 1990
ISI-1136; ISI Isometric PBNP Unit 1 Loop “A” RTD Bypass; dated November 12, 1990
SG-SGDA-04-22; Steam Generator Condition Monitoring Assessment of Spring 2004 Inspection Results and Operational Assessment for Operation Cycle 29 and 30, Point Beach Unit 1, U1R28; Revision 0
SG-SGDA-05-36; Steam Generator Degradation Assessment for Point Beach Unit 1 U1R29; Revision 1
Point Beach Nuclear Plant NDE Procedure Qualification; NDE-451 - Visible Dye Penetrant Examination Temperature Applications 45° F to 125° F; dated March 12, 2002
MRS-TRC-1672; Use of Appendix H Qualified Techniques at Point Beach Unit 1 for the Fall 2005 S/G Inspection; Revision 1
ETSS #20510.1; Eddy Current Examination Technique Specification Sheet; Revision 5
SG-SGDA-05-36; Steam Generator Degradation Assessment for Point Beach Unit 1 U1R29; Revision 1
NDE-173; PDI Generic Procedure for the Ultrasonic Examination of Austenitic Piping Welds; Revision 7
NDE-451; Visible Dye Penetrant Examination Temperature Applications 45° F to 125° F; Revision 21
PTBCH-SG-006; In-Situ Pressure Test Using the Computerized Data Acquisition System; Revision 2
STD-400-173; Checkout and Operation of the Steam Generator Tube Standard In-Situ Pressure Test System; Revision 5

Point Beach Nuclear Plant Procedure NDE-757; Visual Examination For Leakage of Reactor Pressure Vessel Penetration; Revision 4
Point Beach Nuclear Plant Procedure NDE-173; PDI Generic Procedure for the Ultrasonic Examination of Austenitic Piping Welds; Revision 7
Indication Disposition Report 2005-026 Evaluation for AC-601R-2-H2-IWA, Weld 4; dated June 13, 2005
Indication Disposition Report 2005-029 Evaluation for Boric Acid Residue on RPV Supports; dated September 29, 2005
WO 0212695; Repair / Replace SI Accumulator Nozzles as Necessary to Disposition Indications per MR 02-035; dated September 25, 2002
Modification MR 02-035; Unit 1 SI Accumulator Nozzle Upgrade to Remove Nozzle Flaws; dated September 28, 2002
WO 0213476; Unit 1 CRDM Nozzle Repair per MR 03-041; dated October 30, 2002
Modification MR 03-041; Repair of Reactor Vessel Head CRDM Penetrations; dated May 12, 2004
Structural Integrity Associates Calculation PBCH-01Q-301; Accumulator Tank Repair Stress Analysis; Revision 0
WEP-91-150; Wisconsin Electric Power Company Point Beach Unit 1 Evaluation for Non-Hydraulic Expanded Tube; dated April 30, 1991
NMC-400-002; Multifrequency Eddy Current Examination of Non-Ferromagnetic Steam Generator Tubing, Point Beach Nuclear Power Plant, Revision 3
MNC-400-006; Eddy Current Independent QDA (Point Beach); Revision 3
NMC-400-003; Analysis of Bobbin Coil Eddy Current Data - Point Beach; Revision 4
NMC-400-004; Analysis of Rotating Coil Eddy Current Data; Revision 5
NP7.7.16; Steam Generator Program; Revision 6
WesDyne Procedure WDI-SSP-095; Liquid Penetrant Examination; Revision 2
WesDyne Procedure WDI-UT-013; Intraspect UT Analysis Guidelines; Revision 7
WesDyne Procedure WDI-SSP-079; Intraspect Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations, Time of Flight, Longitudinal Wave and Shear Wave, with Open Housing Scanner and/or Gap Scanner; Revision 2
WesDyne Procedure WDI-SSP-077; Intraspect Eddy Current Inspection Reactor Vessel Head Penetrations with Gap Scanner; Revision 2
WesDyne Procedure WDI-SSP-078; Intraspect Eddy Current Inspection of J-Groove Weld in Vessel Head Penetrations with Grooveman; Revision 3
Radiographic Film and Reader Sheets for Welds; WC-J212-20A, 30A; WC-J109-1A, 5A, 20A, 33A
Ultrasonic Examination Records for Welds; WC-J212-1A, 17A, 29A; WC-J109-3A, 20A, 25A, 41A
Liquid Penetrant Examination Records for Welds; WC-P109-21A, 20A, 35A; WC-J109-3A
Eddy Current Examination Records for Welds; WC-P109-10A, 15A, 20A, 35A
Mitsubishi Heavy Industries Weld Repair Record 2901-RVCH-10A R-1; dated April 28, 2004
Mitsubishi Heavy Industries Weld Repair Record 2901-RVCH-10F R-1; dated February 8, 2005
Mitsubishi Heavy Industries Weld Repair Record 2901-RVCH-80B R-1; dated August 16, 2004

Corrective Action Reports
CAP067694; ISI/NDE Finds Indication in Pressurizer Weld; dated October 7, 2005
CAP014516; Unit 1 “A” Steam Generator Manufacturer Defect; dated May 3, 1991
CAP065077; ISI Examination on AC-601R-2-H2_IWA; dated June 13, 2005

CAP067936; ISI Examination Indications on Unit 1 “B” Steam Generator Circumferential Weld; dated October 13, 2005
CAP067716; Observations from NRC During Section XI Inspection; dated October 7, 2005
CAP057877; Repair / Replacement Program May Not Contain Sufficient Detail; dated July 13, 2004
CAP059387; Potential for Missed Section XI Examinations; dated September 22, 2004
CAP062997; Steam Generator and RC Pump Supports; dated March 23, 2005
CAP063797; Section XI Requirement for Replacement Valve May be Exceeded; dated April 17, 2005

Corrective Action Reports Initiated as a Result of NRC Inspection
CAP067675; Questions from NRC During ASME Section XI Inspection; dated October 6, 2005
CAP067719; Questions from NRC During ASME Section XI Inspection; dated October 7, 2005

Section 1R11: Licensed Operator Qualifications
OP 4A; Filling and Venting Reactor Coolant System; Revision 63
OP 4B; Reactor Coolant Pump Operation; Revision 45
EPIP 1.3 Dose Assessment and Protective Action Recommendations; Revision 35
Simulator Exercise Guide PB-LOR-056-003S

Section 1R14: Operator Performance During Nonroutine Evolutions and Events
CAP068715; NRC Report Made
CAP064595; Degraded Coatings on SW Pipes Near Sump B Screens
CAP069331; Reactor Trip due to Circ water pump failure
CAP069334; 4 Hour Report Made to NRC
NP 5.3.3; Incident Investigation and Post-Trip Review Completed; December 14, 2005
PM Deferral for 1P-030B Circulating Water Pump
ODMI for Actions needed for start of 1P-30A Circulating Water Pump; dated December 13, 2005

Section 1R16: Operator Workarounds
CAP067181; Overload Concerns of Safety Related Equipment
CAP068544; 1P-002A Charging Pump Manifold Check Valve Degradation
CAP028609; Resolution Of CV a(1) Status Unknown
Operator Workaround Summary; November 22 and 29, 2005, and December 19, 2005
Operator Workaround Aggregate Impact; December 19, 2005
MRE000048; 2P-2C Charging Pump Trips on RTS at ~85 percent Speed Signal
MRE000003; Loss of Unit 2 Letdown due to 2P-2C Charging Pump Trip
OPR 154; Overload Concerns of Safety Related Equipment
Equipment Out-Of-Service Log; November 21, 2005
Traveling Water Screen WO Work Plan; September 15, 2004
NP 2.1.4; Operator Burdens; Revision 5

Section 1R19: Post-Maintenance Testing
CAP068075; Questions Associated with Refurbished Unit 1 “A” RHR Pump Motor; October 18, 2005
PB 638; Motor Repair Specification; Revision 1
NP 5.1.8; 10 CFR 50.59/72.48 Applicability, Screening and Evaluation; Revision 6

NP 5.3.1; Action Request Process; Revision 28
NP 9.3.3; Spare Parts Equivalency Evaluation; Revision 9
Purchase Order 304927; Revision 3
Westinghouse Electric Company, Acknowledgment Letter for 150 and 200HP Motor Refurbishments; LTR-PMOM-05-250; July 22, 2005
Westinghouse Electric Company Proposed Work Scope; NMC P.O. P304927; Revision 1; Customer Concurrence September 2, 2005
Motor Repair/Refurbishment Report for 200 HP RHR Pump Motor; Westinghouse Electric Company; NMC P.O. P304927
Procurement Deficiency Report - Quality Receipt (PDRQR) No. 2005-443; October 5, 2005
Engineering Evaluation 2005-022; Evaluation of LTR-PMOM-05-250; October 20, 2005
Spare Parts Equivalency Evaluation 2005-088; Stator rewind using equivalent materials and processes by Westinghouse; October 20, 2005
CAP068118; 1-P10A Rotated in Reverse; October 20, 2005
IT 03F; 1P-10A LHSI Pump Profile Test Mode 6 High Cavity Water Level Unit 1; Revision 1; performed October 20, 2005

Section 1R20: Outage
U1R29 Reduced-Inventory Orange Path Contingency Plan; Revision 1
OP4D; Draining the Reactor Coolant System; Revision 67
OP4F; Reactor Coolant System reduced Inventory Requirements Unit 1
CL1E; Containment Closure Checklist Unit 1; Revision 10
CL2C; Mode 5 to Mode 4 Checklist; Revision 10
CL2D; Mode 4 to Mode 3 Checklist; Revision 7
CAP068262; Large Amounts of Boric Acid in the Unit Keyway on Everything Including the Reactor Vessel
CAP068523, Cold Rod Drop Testing Anomalies
CAP067629; Problems With Establishing Conditions for U1 Head Lift
CAP068637; Issues Identified During NRC Walkdown of U1 Containment, Sump ‘B’ Area
CAP067925; Old WO Tag Found Hanging on Protected Equipment
ACE001970; Issues Identified During NRC Walkdown of U1 Containment, Sump ‘B’ Area
L 5C Spent Fuel Pool Cooling and Refueling Water Circulating Pump Normal Operation Valve Lineup
WO 0416468; Work Order for Bare Metal Visual Examination of Bottom of Reactor Vessel

Section 1R22: Surveillance Testing
CAP067248; Large Amount of Boric Acid on 8' U1 Containment
MRE000473; Maintenance Rule Evaluation, Large Amount of Boric Acid on 8 Ft U1 Containment
RMP 9314; “1(2) SI-850A/B Maintenance, Static Test and Adjustment”
IT 40; “Safety Injection Valves (Quarterly)” Unit 1
PBNP Inservice Background Valve Data Sheet 1 for SI-850A RHR Pump P-10A Suction B Isolation

Section 1R23: Temporary Plant Modifications
CAP051581 VNPSE Valves IST Acceptance Criteria Incorrect Not Conservative
TM 04-001; Temporary Replacement of Unit 1 Purge Supply/Return Valves

U1 WO 0414290 Remove Blind Flange (Temp Mod) and Install Valve to Support Refueling. Reinstall temp Mod for VNPSE-03212
U1WO 0414288 Remove Blind Flange (Temp Mod) and Install Valve to Support Refueling. Reinstall temp Mod for VNPSE-03244
SCR 2004-0056-02; Temporary Modification TM 04-001 Temporary Replacement of Unit 1 Purge Supply/Return Valves, Rev. 2.
TM 2005-016 "Unit 1 Reactor Cavity Portable Sump Pump Installation
SCR 2005-0238 Temporary Modification 2005-016, Unit 1 Reactor Cavity Portable Sump Pump Installation
SCR 2004-0056-02, 10 CFR 50.59/72.48 Screening for Temporary Modification TM 04-001
Eval 2005-010; 10CRR 50.59 Evaluation for Temporary Modification 2005-016

Section 1EP4: Emergency Action Level and Emergency Plan Changes
Point Beach Emergency Plan Appendix C; Revision 14
Point Beach Emergency Plan Appendix J; Revision 13
Point Beach Emergency Plan Appendix L; Revision 2

Section 2OS1: Access Control to Radiologically Significant Areas
HP 2.5; Radiation Work Permit; Revision 33
HP 2.14; Containment Keyway Personnel Access; Revision 11
Job File 168; Hydrogen Peroxide or Hydrazine Add to RCS; Revision 1
HPIP 3.80; Letdown Gas Striper Surveys; Revision 3
HP 3.2.8; Posting Requirements for Areas Affected by Fuel Movement; Revision 12
OP 7A; Placing Residual Heat Removal System In Operation; Revision 44
CAP067739; LHRA Posting Changed to Radiation Area During Fuel Handling; October 8, 2005
CAP064590; Loss of Contamination Control Due to RWP Violation; May 16, 2005

Section 2OS2: As-Low-As-Is-Reasonably-Achievable (ALARA) Planning And Controls
NP 4.2.1; ALARA Program; Revision 14
HPIP 4.40; TEDE ALARA Evaluation; Revision 1
Unit 1 Refueling Outage (R29) Dose Estimates and Daily Exposure Reports (Week of October 17, 2005)
RWP 05-167 (Revision 0); Associated ALARA Review, TEDE ALARA Evaluation and Radiation Surveys; Unit 1 Under-Vessel Boric Acid Clean-Up
RWP 05-157 (Revision 0); Associated ALARA Review, TEDE ALARA Evaluation and Radiation Surveys; Unit 1 Reactor Head Replacement Activities
RWP 05-133 & 05-164 (Revision 0); Associated ALARA Review, TEDE ALARA Evaluation and Radiation Surveys; Nozzle Dam Installation & Removal
RWP 05-140 (Revision 0) and Associated ALARA Review; Unit 1 Upper Cavity Decontamination
RWP 05-134 (Revision 0) and Associated ALARA Review; Steam Generator Eddy Current Testing
RWP 05-127 (Revision 0); Install/Remove Containment Scaffold
RWP 05-123 (Revision 0); Mechanical Valve Maintenance
In-Progress ALARA Review/Assessment; Unit 1 Reactor Vessel Head Replacement; dated October 10, 2005
In-Progress ALARA Review/Assessment; Install and Remove Steam Generator Nozzle Dams; dated October 7, 2005

In-Progress ALARA Review/Assessment; Steam Generator Manway Diaphragm Removal and Installation; dated October 7, 2005
In-Progress ALARA Review/Assessment; Remove and Install Steam Generator Lower Handhole Covers; dated October 10, 2005
In-Progress ALARA Review/Assessment; Sludge Lancing and Visual Inspection on “A” Steam Generator; dated October 10, 2005
In-Progress ALARA Review/Assessment; Remove “A” Reactor Coolant Pump Motor; dated October 15, 2005
Snap Shot Assessment Report SA 016037; RP Outage Access Control/ALARA
Nuclear Oversight Observation Report 2005-002-3-022; Radiological Protection
CAP066899; Personnel Exceeded Dose Rate Alarm Setpoint on Electronic Dosimetry; dated September 9, 2005
CAP064288; Amount of Assigned Workers not ALARA; dated May 3, 2005
CAP064372; Potential for Dose Overexposure/Failure to Properly Terminate Workers; dated May 5, 2005

Section 2PS2: Radioactive Material Processing and Transportation
CAP055279; Blowdown Evaporator System Repair/Replacement; dated January 29, 2004
CAP054962; Radioactive Waste Shipping Program Weakness; dated March 18, 2004
CAP055545; BDE Tube Leak Delays Waste Water Processing; dated April 09, 2004
CAP057357; Potentially Contaminated Control Side Oil Sample Release; dated June 11, 2004
CAP057824; Transportation Package Notification Requirement Changes; dated July 08, 2004
CAP058397; Certificate for Rad Material Shipping Package Issued by NRC; dated August 09, 2004
CAP058957; Dry Fuel Shipment Trainer Release Survey Detects Fixed Contamination; dated September 02, 2004
CAP060679; BDE Bottoms Loop Leak is Worsening; dated November 21, 2004
CAP061373; Additional Dose Received by RP and Ops Personnel During Resin Transfer; dated January 07, 2005
CAP063049; Westinghouse “Traveler” Shipping Package for New Fuel; dated March 25, 2005
CAP064995; Requirement of Pre-Shipment Inspection; dated June 8, 2005
CAP065076; Backup of Water in Decon Facility During Dewatering; dated June 13, 2005
CAP066899; Personnel Exceeded Dose Rate Alarm Setpoint on EPD; dated September 9, 2005
CAP067029; Examples of Focused Self-Assessments Not Scheduled/Performed as Expected (included RAM Shipping); dated September 16, 2005
CAP067599; RV Head Shipping Cover Plate Gasket Not Fully Compressed; dated October 5, 2005
CAP067861; Leak in Hose and Fittings During BDE Pumpout; dated October 12, 2005
CAP Data of Industry Operating Experience (OE) Relative to Radioactive Material Processing and Transport; Dated January 05, 2004 through November 20, 2005
ER-04-008; Duratek Engineering Report - Characterization of Point Beach Unit-2 Nuclear Power Plant Reactor Pressure Vessel Head; Revision 0
Focused Area Self-Assessment (Mid Cycle) PBSA-PAS-05-01; dated March 25, 2005
FO-OP-023; Bead Resin/Activated Carbon Dewatering Procedure for Duratek 14-215 or Smaller Liner; Revision 22
HPI-02-LP015; Radiation Protection Technologist Training Program, Basic Radiological Protection; Revision 2

Job File 160; Resin Transfers; dated July 8, 2005
Job File 180; Radiation Protection Old Reactor Head Transportation
MRS-SSP-1710 (Westinghouse) RVH Prepare for Disposal and Rig Out of Containment; Revision 1
NP 7.1.5; Abandoned Equipment; Revision 3
NP9.9.1; Warehouse Receiving; Revision 4
PCP; Process Control Program; Revision 4
RDW 14.4; Requirements for the Storage of Container in Outside Areas; Revision 3
RDW 15.0; Radioactive Material Shipping; Revision 6
RDW 15.1; Determining Shipment Type and Packaging Requirements; Revision 8
RDW 15.2.3; Packaging Type A Quantity Material for Shipment; Revision 7
RDW 15.2.4; Packaging Type B Quantity Material for Shipment; Revision 6
RDW 15.3; Radioactive Material (Greater Than Limited Quantity) Shipment via Non-Exclusive Use Vehicle; Revision 6
RDW 15.6; Reportable Quantity; Revision 3
RDW 15.15; Exempt Quantity Shipments; Revision 2
RDW 15.16; Packaging and Shipping of LSA and SCO Material via an Exclusive Use Vehicle; Revision 2
RDW 15.17; Packaging and Shipping of Radioactive Material Excepted Package, Limited Quantifies; Revision 3
RDW 16.1; Preparation, Transport and Storage of Radwaste; Revision 4
RDW 16.7; Dry Active waste Processing for Transport in Sea Land Vans; Revision 4
RDW 17.0; Liquid Radwaste Processing; Revision 2
RDW 17.1; Scheduling and Setup for Liquid Waste Processing; Revision 2
RDW 17.3; Processing Bead Resin by Dewatering; Revision 7
RDW 17.3.1 Dewatering Resin Liners During Discharge to Truckbay; Revision 0
RDW 17.8; Processing Evaporator Bottoms by Drying; Revision 2
RDW 18.1; Determining Activity and Radionuclide Content of Radwaste and Radioactive Material Packages; Revision 6
RDW 18.1.1; 10 CFR 61 Sampling Program; Revision 3
RDW 18.2; Radwaste Classification Shipment Type and Waste Stability Determination; Revision 2
Review of the Radiation Protection Program for the Year 2003; dated March 23, 2004
RWP 05-261; RU2 Old Rx Head Shipping Prep and Transfer Activities; Revision 0
OI 19; Spent Fuel Pool Demineralizer resin Flush and Recharge, U6: Revision 13
OI 20; Resin Transfer Cask and Resin Tank Storage Tank T-112; Revision 32
OI 21; Mixed Bed (HOH) Demineralizer Resin Flush and Recharge, 1U1A(B) and 2U1A(B); Revision 21
Nuclear Oversight Observation Reports, Miscellaneous; dated June 21, 2002 through July 2, 2004
Radwaste Training Roster, Point Beach Radiation Protection Technicians; dated September 2, 2005
Report of Analysis (Vendor/Teledyne), Login # L25300, Project ID # WI744-3PPointBeach (Analysis of 2004 Resin Comp); dated April 1, 2005
Report of Analysis (Vendor / Teledyne), Login # L25300, Project ID # WI744-3PPointBeach (Analysis of 2004 Blowdown Evaporator Bottoms Comp); dated April 1, 2005
Report of Analysis (Vendor/Teledyne), Login # L25300, Project ID # WI744-3PPointBeach (Analysis of Unit-1 Reactor Head); dated April 1, 2005

Report of Analysis/Certificate of Conformance (Vendor / Teledyne), LIMS # 22958, Project ID # WI744-3PPointBeach (Analysis of Smears / DAW, BDE Bottoms, and Resins); dated March 25, 2004
Snap-Shot Evaluation of the Radioactive Material Processing and Transportation Program - Draft Report; dated December 2, 2005
Shipping Package #2004-021, Temporary Demin Equipment; dated April 4, 2004
Shipping Package #2004-055, Blowdown Evaporator (BDE) Bottoms; dated July 22, 2004
Shipping Package #2005-011, Resin for Processing; dated March 15, 2005
Shipping Package #2005-028, RT-11 Source; dated April 30, 2005
Shipping Package #2005-032, Split Pin Equipment; dated May 6, 2005
Shipping Package #2005-033, Split Pin Equipment; dated May 6, 2005
Shipping Package #2005-036, Reactor Head Replacement Equipment; dated May 10, 2005
Shipping Package #2005-045, Unit-2 Reactor Vessel Head; dated July 18, 2005
Shipping Package #2005-068, Contaminated Laundry for Processing; dated October 13, 2005
Shipping Package #2005-070, Unit-1 Reactor Vessel Head; dated October 18, 2005
Surveillance PC21-5 for 2005 Year to Date

Section 2PS3: Radiological Environmental Monitoring and Radioactive Material Control Programs

Annual Monitoring Reports for 2003 and 2004; dated April 29, 2004, and May 2, 2005
CAP051212; Valve with Radioactive Material Tag Found in the Ready Stores Storeroom; October 18, 2003
CAP054693; Positive I-131 Airborne Concentration During 2003 REMP; March 11, 2004
CAP057357; Potentially Contaminated Controlled Side Oil Sample Release; June 11, 2004
CAP058092; Incomplete and Conflicting Information in the 2003 Annual Monitoring Report; June 18, 2004
CAP058264; Cesium-137 Found in Two Soil Samples for the Security Upgrade Project; August 4, 2004
CAP063815; Rubber Totes Possibly from RCA Found on Clean Side of Plant; April 18, 2005
CAP063893; Radioactive Material Found on Clean Side of Plant; April 20, 2005
CAP064202; Contaminated Strap from Warehouse 3; April 29, 2004
CAP064784; CAPs Assigned Low Significance Value; May 25, 2005
CAP065753; Weekly RM-14 Frisker Source Checks Were not Performed; July 15, 2005
CAP066099; REMP Sampling Errors; August 1, 2005
CAP067009; Cs-136/Co-60 in Dirt; September 19, 2005
CAP068104; Problem with Outside Temperature Display on MI-6206 not Resolved; October 19, 2005
Daily RP Instrument Checklist; dated October 20, 2005
Document Review and Approval, “Offsite Dose Calculation Manual,” for Revision 15, dated June 25, 2003; for Revision 16, dated October 14, 2004; and for Revision 17, dated August 19, 2005
Document Review and Approval, “Environmental Manual,” for Revision 18; dated September 22, 2005
Engineering Change Notice No. 2005-0154; dated October 18, 2005
Engineering Evaluation No. 2003-0004, “PBNP Meteorological Data Wind Rose Comparison for Various Time Periods”; dated October 28, 2003
Environmental Manual; Revision 18; dated September 28, 2005

HPCAL 1.33, “Maintenance and Calibration of Low Volume Air Samplers,” Revision 11, performed on sampler No. HP-LV-19 (March 14, 2006); No. HP-LV-53 (July 14, 2004, and June 9, 2005); No. HP-LV-54 (July 7, 2004); No. HP-LV-56 (July 14, 2004, and June 10, 2005); No. HP-LV-82 (July 28, 2004); No. HP-LV-85 (June 9, 2005); No. HP-LV-87 (July 8, 2004, and June 9, 2005); and No. HP-LV-91 (July 7, 2004, and March 14, 2006)
HPCAL 2.8, “Eberline PCM-1B Personnel Contamination Monitor Calibration Procedure,” Revision 15 and 16, Performed on Monitors No. 7737 (July 10, 2005), No. 7738 (April 7, 2005), and No. 7739 (May 7, 2005)
HPCAL 2.15, “Small Articles Monitor Type SAM-9/11 Calibration and Efficiency,” Revision 9, Performed on Monitors No. 2 (February 14, 2005), No. 4 (April 13, 2005), and No. 5 (February 14, 2005)
ICP 6.55; “Meteorological Instrument Calibration”; Revisions 12 and 13; completed on March 30, 2005, October 8, 2004, and March 26, 2004
IWP 05-005-01; Installation of Primary Meteorological Tower, Attachment A, “Test Procedure for Checking the Universal Interface Module Voltage Output”; completed October 18, 2005
IWP 05-005-02; Installation of Backup Meteorological Tower, Attachment A, “Test Procedure for Checking the Universal Interface Module Voltage Output”; performed September 22, 2005
IWP 05-005-03; Installation of Inland Meteorological Tower, Attachment A, “Operational Checkout and Calibration Procedure - Field”; performed October 17, 2005
NOS Observation Report No. 2004-002-3-063, “Radiological Environmental Monitoring Program”; completed June 28, 2004
NOS Observation Report No. 2004-002-3-064, “Radioactive Waste Processing and Program Support”; completed June 29, 2004
NOS Observation Report No. 2005-002-3-004, “Emergent Assessment”; completed May 2, 2005
NUPIC Joint Audit/Survey of “Environmental, Inc.”; prepared June 3, 2003
NUPIC Joint Audit/Survey of “Environmental, Inc.”; revised December 8, 2003
Procedure NP 4.2.25; “Release of Material, Equipment and Personal Items from Radiologically Controlled Areas”; Revision 14; dated March 17, 2004
RPM Policy Note No. 2001-02, “Release of Items from RCA Using a SAM”; signed by S. Thomas
Snapshot Report, AR Number SA016036, “REMP/RAM”; performed September 30, 2005
Snapshot Self-Assessment Report, “0TH012103 Equipment Snapshot Self-Assessment”; performed June 13, 2005

Section 4OA2: Identification and Resolution of Problems

Corrective Action Program Documents
CAP068535; Unit 2 Enters LCO 3.03 and Commenced TS Required Shutdown; dated November 2, 2005
CAP068534; Wax Found on Two Elevations on Floors in Unit 2 Containment; dated November 2, 2005
CAP068527; Degraded Coating Inventory for Unit 2 May Exceed OPR 161 Analysis Limit; dated November 1, 2005
CAP068526; Unit 1 and 2 Containment Coating Qualifications; dated November 1, 2005
CAP068444; U1R29 Containment Floor Wax Inspection; dated October 30, 2005
CAP068442; GL 98-04 Commitments; dated October 30, 2005

CAP068403; Enhancements - Containment Coating Program Assessment; dated October 28, 2005
CAP068402; Area for Improvement 3 - Containment Coating Program Assessment; October 28, 2005
CAP068400; Area for Improvement 2 - Containment Coating Program Assessment; dated October 28, 2005
CAP068399; Area for Improvement 1 - Containment Coating Program Assessment; dated October 28, 2005
CAP068182; Containment Coating Issues are not Identified in a Timely Manner; dated October 21, 2005
CAP068079; Damaged Level 1 Coatings; dated October 18, 2005
CAP068071; Floor Wax in SG/RCP Cubicles Needs to be Removed; dated October 18, 2005
CAP068067; Steel Pipe Supports in Containment with Unqualified Coating; dated October 18, 2005
CAP030250; 2SI-850B RHR Pump Suction from Sump B Failed its Stroke Time; dated November 27, 2002
CAP021179; Limit Switch Problem Impacts Valve Testing; dated April 25, 2002
CAP017098; Calc N92-071 ECCS Branch K Factors Improperly Models SI-850 A&B; dated April 25, 2002
CAP006127; RHR Pump B Suction Valve Failed Stroke Test; dated April 24, 2002
CAP001016; 1SI-850A Failed Surveillance; dated December 19, 2001
CAP021170; Limit Switch Problem Impacts Valve Testing; dated September 9, 1999
CA004283; Added 500 ml oil to 2SI-850B during IT-45; dated April 3, 2002
CA011194; Limit Switch Problem Impacts Valve Testing; dated April 26, 2002
CR01-2885; 1SI-00850A Opening Peak Hydraulic Pressure; dated October 1, 2001

Corrective Action Program Documents as a Result of NRC Inspection
CAP069113; NRC Sites Potential Criterion 3 Violation; dated December 1, 2005
CAP069111; NRC Sites Potential Criterion 16 Violation; dated December 1, 2005
CAP069101; ECCS Leakage Test Improvements; dated December 1, 2005
CAP069087; As-Found Testing for WO 02162 was Performed Incorrectly; dated November 30, 2005
CAP069078; Boric Acid Found on SI-850 Valves; dated November 29, 2005
CAP069050; Guidance for Detection of Passive Failure During Sump Recirculation; dated November 28, 2005
CAP069029; Original FFSAR Citations in EDMS are not Correct; dated November 26, 2005
CAP068997; RMP 9314 Indicates Training on The Procedure May be Necessary; dated November 23, 2005
CAP068923; Concern with the Potential of SI-850 Valves to Drift Closed During Sump Recirculation; dated November 18, 2005
CAP068902; Deficiencies in PBNP Coatings Program; dated November 17, 2005
CAP068882; Parts Installed Without Equivalency Evaluation; dated November 16, 2005
CAP068872; Maintenance Rule Evaluations not Generated for Coatings Operability; dated November 16, 2005
CAP068832; Containment Coatings Inspection Question 76; dated November 14, 2005
CAP068803; CLRT Attachment 2 Potentially Needs Revision to ECCS Leakage Basis; dated November 12, 2005
CAP068663; Walkdown of Tendon Gallery Combustible Loading; dated November 7, 2005

CAP068637; Issues Identified During NRC Walkdown of U1 Containment, Sump B Area; dated November 5, 2005
CAP068629; Tendon Gallery Inspection Results; dated November 4, 2005
CAP068447; Question with Ability of ECCS Sump Screens to Pass Required Flow; dated October 30, 2005
CAP068442; GL 98-04 Commitments; dated October 30, 2005
CAP068373; Containment Coatings Not Maintained Within Analyzed Limits; dated October 27, 2005
CAP068346; Potential Weakness in Containment Coating Program and Analyses; dated October 26, 2005
CAP067716; Observations from NRC During Section XI Inspection; dated October 7, 2005

Operability Recommendations (OPRs)
OPR000036; Non-Standard Rod Packing Configuration Installed on 2-SI-850B, dated December 4, 2002
OPR000161; Containment ECCS Suction Strainers; dated December 9, 2005
OPR000162; The RHR Pumps 1(2) P-10A(B); dated December 9, 2005
OPR000164; Floor Wax; dated November 5, 2005
OPR000165; 1&2 SI-00850A&B, RHR Pump Suction from Sump B; dated November 6, 2005

Other Documents
Spare Parts Equivalency Evaluation Document, Amoco Rykon Premium Oil 32 to Chevron Rykon Premium Oil ISO 32; dated November 9, 2000
Calculation 2005-0024; Evaluation of Containment Sump Screen Debris Buildup on EPRI Technical Report and Current Degraded Epoxy Inventories; Revision 1
Calculation 2001-0001, Hydraulic Pressures Associated with the SI-850 Valves, dated January 8, 2001
Licensee Event Report 05000266-2000-002-00; TS Surveillance Requirement to Verify ECCS Valve Position Not Fully Implemented; dated February 10, 2000
Stearns-Roger Drawing 546-J-81056-B; Containment Isolation Valve General Assembly and Parts List; Revision 3
Stearns-Rogers Drawing 546-J-81056-B; Lower Body Assembly; Revision 1
Manatrol Division Parker Hannifin Drawing 203-D-676; Directional Control Valve; Revision B
Point Beach Nuclear Plant Drawing —276; Containment Safety Injection Sump Requirements for Screens; Revision 2
NP 8.4.15; Protective Coating Program; Revision 4
NDE-802; Condition Monitoring and Assessment of Containment Coatings; Revision 0
Point Beach Nuclear Plant Drawing MSFK00000264; P&ID Auxiliary Coolant System; Revision 64
Point Beach Nuclear Plant Drawing MRHL183001-10; Auxiliary Coolant System to RHR Pumps P10A &B Suction and Discharge Outside Containment; Revision 10
Point Beach Nuclear Plant Drawing MSIL133005-07; Auxiliary Coolant System Penet. P-7 to RHR Pump 2P-10A &B Suction; Revision 7
Wisconsin Electric Letter NPL 97-0315; Supplement to TSs Change Request 192, Point Beach Nuclear Plant Units 1 and 2; dated June 3, 1997
CLRT Testing Program Basis Document, Revision 6

Procedures
CL 7A; Safety Injection System Checklist Unit 1; Revision 23

IT 510B; Leakage Reduction and Preventative Maintenance Program Test of Safety Injection Test Line (Refueling) Unit 1, Revision 20
IT 530A; Leakage Reduction and Preventative Maintenance Program Seat Leakage Test of The Train A RHR System (Refueling) Unit 1, Revision 15
IT 531; Leakage Reduction and Preventative Maintenance Program Test of Containment Sump B Suction Line Mode 5, 6, or Defueled Unit 1
IT 530E; Leakage Reduction and Preventative Maintenance Program Test of the LHSI and RHR System Unit 1, Revision 6
IT 535E; Leakage Reduction and Preventative Maintenance Program Test of the LHSI and RHR System Unit 2, Revision 5

Procedures - Completed
CL 1B, Containment Barrier Checklist Unit 1; dated November 21, 2005
RMP 9314; 1(2)SI-850A/B Maintenance, Static Test, and Adjustment; dated March 3, 2005
IT-531; LRPM Program Test of Containment Sump B Suction Line Mode 5,6 or Defueled U-1; dated October 9, 2005
IT-536; Leakage Reduction and Preventative Maintenance Program Test of Containment Sump B Suction Line Mode 5, 6, or Defueled Unit 2; dated April 28, 2005
IT-40; Safety Injection Valves Quarterly Unit 1; dated May 25, 2005

Work Orders
WO 9704782; P-10A RHR Pump Sump B Suction Operator; dated July 16, 1997
WO 9704783; P-10B RHR Pump Sump B Suction Operator; dated August 4, 1997
WO 9604943; P-10A RHR Pump Sump B Suction Operator; dated December 4, 1996
WO 9604942; P-10B RHR Pump Sump B Suction Operator; dated December 3, 1996
WO 0216212; P-10B RHR Pump Sump B Suction Operator, dated November 27, 2002

Section 4OA5: Other Activities
AOP-13C; Abnormal Operating Procedure: Severe Weather Conditions; Revision 16
Calculation No. 05Q0537-C-001; Reinforce Operations Office South Block Wall; Revision 0
Callup ID: M-RELVLV; Equipment ID: AF-4035; Replace Relief Valve with Pretested Valve Per RMP 9377-1; due date October 27, 2015
CA053119; Perform New Review of AFW System to Support Recirculation AOV Safety Function Upgrade; dated October 11, 2003
CA064089; Evaluation of AFW Common Recirculation Line Requires Additional Effort; dated August 9, 2005
CAP065676; Transmission System Documentation and Agreement Issues; dated July 12, 2005
CAP066199; Evaluation of AFW Common Recirculation Line Requires Additional Effort; dated August 5, 2005
CE011367; Condition Evaluation: Requirements for Seismic Induced Blockage of Pipe Not Clear; dated March 24, 2003
CMP 2.3; Component Maintenance Program: Relief/Safety Valves; Revision 7
Corrective Action Plan; OBD000274, Potential for Auxiliary Feedwater Pump Recirculation Line Crimp Resulting from a Seismic Event or Tornado; dated November 17, 2005
DBD-01; Design Basis Document: Auxiliary Feedwater System; Revision 14
Drawing M217, Sheet 1; Piping & Instrumentation Diagram, Auxiliary Feedwater System, Units 1 & 2; Revision 82

Drawing P-103; Emergency Feedwater Pumps to Main Feedwater Lines, Units 1 & 2; Revision 12
Drawing P-159; Auxiliary Feedwater from Heating Boiler Condensate Return & Pump Recirculation to Condensate Storage Tank; Revision 10
Engineering Evaluation No. 2005-0012; Auxiliary Feedwater Recirculation Line Crimping Evaluation; Revision 1
OBD000274; Evaluation of Potential for Auxiliary Feedwater Recirculation Line Crimp; dated October 12, 2005
OBD000293; Revise Classification of AF-04035 in CMP 2.3; dated November 17, 2005
OTH024554; Update AFW DBD with Pertinent Information from EE 2005-0012; dated September 2, 2005
Purchase Order No. P305228; Valve, Relief, Crosby Model 2J3 JLT-JOS-E-15-C, Set Pressure 50 psig, Capacity 268 gpm; Revision 0
Receipt Inspection Data Sheet; PO No. P305228, Relief Valve; dated October 1, 2005
SPEED No. 2005-062; Spare Parts Equivalency Evaluation Document: Replacement Valve for Crosby Style JLT-JOS Containing a 2" Inlet, 3" Outlet, a "J" Orifice and Metal Seats; Revision 0
Tyco Valves & Controls; Valve Assembly & Test Instruction Form; PO No. P305228, Crosby Relief Valve, Model JLT-JOS-E-15-C; dated August 29, 2005
WO 0511210; Work Order: Test AF-4035 (Aux Feedwater Pumps, Mini Recirculation Relief) to Validate Proper Operation and Setpoints; dated November 14, 2005
WO 0511256; Work Order: Install New Support for Operations Building South Wall; dated October 9, 2005

Replacement Reactor Vessel Closure Head (71007)
Modification No. 03-047; Point Beach Unit 1, Reactor Vessel Head Replacement Project; Revision 0
Design Specification No. 414A83; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head (RRVCH); Revision 2
Design Specification No. 414A86; Point Beach Unit 1, Control Rod Drive Mechanism (CRDM) Model L106A; Revision 3
Design Specification No. 414A86 Addendum 1; Point Beach Unit 1, Control Rod Drive Mechanism (CRDM) Model L106A; Revision 3
Design Specification No. NSD-ENG-DS-327; Point Beach Units 1 and 2, Core Exit Thermocouple Nozzle Assembly (CETNA); Revision 0
Document L5-01DP510; Point Beach Unit 1, Replacement Reactor Vessel Closure Head, Design Report; Revision 2
WCAP-16345-P, Addendum 2; Point Beach Unit 1, Replacement Reactor Vessel Closure Head - Design Report, Revision 0; dated March 2005
WCAP-16345-P, Addendum 1; Point Beach Unit 1, Replacement Reactor Vessel Closure Head - Design Report, Revision 0; dated March 2005
WCAP-16345-P; Point Beach Unit 1, Replacement Reactor Vessel Closure Head - Design Report; Revision 0
Calculation Note No. CN-RCDA-03-140; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - Design Transients for Use in Stress Analysis; Revision 0
Calculation Note No. CN-RCDA-03-135; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - Analysis Procedure; Revision 2
Calculation Note No. CN-RCDA-04-63; Point Beach Units 1 and 2, RRVCHs, ASME Section XI Code Reconciliation; Revision 2

Calculation Note No. CN-RCUMP-04-3; Point Beach Unit 1, Replacement Reactor Vessel Closure Head - Flange Model, Thermal/Structural Analysis; Revision 0
Calculation Note No. CN-RCUMP-04-4, Point Beach Unit 1, Replacement Head Project - Closure Head Flange ASME Code and Leakage Evaluation; Revision 0
Calculation Note No. CN-RCDA-03-118; Kewaunee, Replacement Reactor Vessel Closure Head, Finite Element Model of CRDM Head Adapters; Revision 0
Calculation Note No. CN-RCDA-03-119; Kewaunee, Replacement Reactor Vessel Closure Head, ANSYS Thermal and Structural Analyses of CRDM Head Adapters; Revision 1
Calculation Note No. CN-RCDA-03-120; Kewaunee, Replacement Reactor Vessel Closure Head, CDRM Head Adapter ASME Code Evaluation; Revision 0
Calculation Note No. CN-RCDA-04-44; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - CRDM Head Adapter ASME Code Evaluation; Revision 1
Calculation Note No. CN-RCDA-04-22; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - Closure Head Adapter Bimetallic Weld Analysis; Revision 0
Calculation Note No. CN-RCDA-04-78; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - Closure Head Lifting Lug Stress Analysis; Revision 3
Calculation Note No. CN-RCDA-03-121; J. M. Farley Units I & 2, Replacement Reactor Vessel Closure Head, Finite Element Model of Vent Pipe; Revision 0
Calculation Note No. CN-RCDA-03-122; J. M. Farley Units I & 2, Replacement Reactor Vessel Closure Head, ANSYS Thermal and Structural Analysis of Vent Pipe; Revision 0
Calculation Note No. CN-RCDA-04-42; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - Vent Pipe ASME Code Evaluation; Revision 1
Calculation Note No. CN-PAFM-05-6; Point Beach Unit 1, Replacement Reactor Vessel Closure Head, Fracture Evaluation; Revision 0
Design Report No. DAR-CI-04-18; Point Beach Units 1 and 2, Core Exit Thermocouple Nozzle Assembly (CETNA) Design Report; Revision 0
Calculation Note No. CN-CI-03-56; Canopy Seal-less Core Exit Thermocouple Nozzle Assembly (CETNA) Stress Evaluation; Revision 0
Document No. L5-01DP501; Point Beach Unit 1, Basic Sizing Report of Reactor Vessel Closure Head; Revision 3
Document No. L5-01DP505, Justification for Nonconformance Reports of Replacement Reactor Vessel Closure Head; Revision 2
Document No. L5-01DP506; Point Beach Unit 1, Replacement Reactor Vessel Closure Head, Additional Reconciliation of Applicable Documents for the Design Report; Revision 1
Document PB-KCS-04-0009; Point Beach Units 1 and 2, Control Rod Drive Mechanism, Design Report PB-KCS-04-0002, Revision 2, Addendum; Revision 3
Document PB-KCS-04-0002; Point Beach Units 1 and 2, Control Rod Drive Mechanism, Design Report; Revision 2
WCAP-16267-P, Addendum 4; Point Beach Units 1 and 2, Replacement Control Rod Drive Mechanism, Design Report, Revision 0; dated March 2005
WCAP-16267-P, Addendum 3; Point Beach Units 1 and 2, Replacement Control Rod Drive Mechanism, Design Report, Revision 0; dated March 2005
WCAP-16267-P, Addendum 2; Point Beach Units 1 and 2, Replacement Control Rod Drive Mechanism, Design Report, Revision 0; dated December 2004
WCAP-16267-P, Addendum 1; Point Beach Units 1 and 2, Replacement Control Rod Drive Mechanism, Design Report, Revision 0; dated November 2004
WCAP-16267-P; Point Beach Units 1 and 2, Replacement Control Rod Drive Mechanism, Design Report; Revision 0

Calculation Note No. CN-ENG-04-6; Point Beach Units 1 and 2, CRDM - Analysis Procedure; Revision 1
Calculation Note No. CN-ENG-04-37; Point Beach, CRDM - ASME Section XI Reconciliation; Revision 1
Calculation Note No. CN-ENG-04-33; Point Beach, CRDM - Tentative Pressure Thickness Calculations Per NB-3324; Revision 0
Calculation Note No. CN-ENG-04-20; Point Beach, CRDM - Pressure Housing ASME Qualification; Revision 1
Document No. PB-KCS-04-0005; Point Beach Units 1 and 2, Control Rod Drive Mechanism, Design Data Report; Revision 1
Document No. PB-KCS-04-0008; Point Beach Units 1 and 2, Control Rod Drive Mechanism, Additional Reconciliation of Design Report with Latest Drawing Revision; Revision 4
Document No. PB-KCS-05-0001; Point Beach Unit 1, Justification for Nonconformance Reports of Replacement Control Rod Drive Mechanism; Revision 2
MHI Drawing L5-01DP101; Replacement Reactor Vessel Closure Head, Closure Head General Assembly 1/2 Revision 2
MHI Drawing L5-01DP102; Replacement Reactor Vessel Closure Head, Closure Head General Assembly 2/2; Revision 1
MHI Drawing L5-01DP103; Replacement Reactor Vessel Closure Head, Closure Head Welding 1/2; Revision 1
MHI Drawing L5-01DP104; Replacement Reactor Vessel Closure Head, Closure Head Welding 2/2; Revision 0
MHI Drawing L5-01DP107; Replacement Reactor Vessel Closure Head, Closure Head Penetration Position 1/2; Revision 0
MHI Drawing L5-01DP108; Replacement Reactor Vessel Closure Head, Closure Head Penetration Position 2/2; Revision 0
MHI Drawing L5-01DP109; Replacement Reactor Vessel Closure Head, Closure Head and Adapter Housing Assembly; Revision 1
MHI Drawing L5-01DP110; Replacement Reactor Vessel Closure Head, Instrumentation Port Head Adapter 1/2; Revision 1
MHI Drawing L5-01DP111; Replacement Reactor Vessel Closure Head, Instrumentation Port Head Adapter 2/2; Revision 0
MHI Drawing L5-01DL124; Replacement Reactor Vessel Closure Head, Closure Head Name Plate; Revision 0
MHI Drawing L5-01DP151; Replacement Reactor Vessel Closure Head, Closure Head Marking Procedure Drawing; Revision 0
MHI Drawing L5-01DP171; Replacement Reactor Vessel Closure Head, As-Built Drawing (RV Closure Head) 1/3; Revision 1
MHI Drawing L5-01DP172; Replacement Reactor Vessel Closure Head, As-Built Drawing (RV Closure Head) 2/3; Revision 1
MHI Drawing L5-01DP173; Replacement Reactor Vessel Closure Head, As-Built Drawing (RV Closure Head) 3/3; Revision 1
MHI Drawing L5-03BJ101; Reactor Vessel Closure Head, General Assembly Drawing; Revision 4
MHI Drawing L5-03BJ102; Control Rod Drive Mechanism, General Assembly; Revision 2
MHI Drawing L5-03BJ103; Control Rod Drive Mechanism, Pressure Housing Assembly; Revision 2
MHI Drawing L5-03BJ211; Control Rod Drive Mechanism, Rod Travel Housing; Revision 3

MHI Drawing L5-03BJ212; Control Rod Drive Mechanism, One-Piece Latch Housing; Revision 4
MHI Letter MHI-NMC-05-P100; Subject: Typographical Error in Point Beach Unit 1 CRDM CDS; dated October 3, 2005
Westinghouse Letter LTR-RCDA-05-751; Subject: Transmittal of 414A86 Rev 3 Addendum 1 - Applicable for Point Beach Unit 1 Replacement CRDMs; dated October 7, 2005
Reactor Vessel Closure Head and Control Rod Drive Mechanisms; ASME Normal Pressure and Temperature Component Certification; Mitsubishi Heavy Industries, Ltd; dated March 18, 2005

Head Assembly Upgrade Package (71007)
Modification No. 03-049; Point Beach Unit 1; Install Head Assembly Upgrade Package (HAUP); Revision 0
Design Specification No. 418A27; Point Beach Units 1 and 2, Head Assembly Upgrade Package (HAUP); Revision 7
Design Specification No. 953206; Control Rod Drive Mechanism Seismic Support Design Specification, ASME Section III Code Class I Component Supports; Revision 1
Calculation Note No. CN-RVCHP-04-2; Point Beach Units 1 and 2, HAUP Airflow Analysis; Revision 2
Calculation Note No. CN-RVCHP-04-3; Point Beach HAUP Plenum Stress Qualification; Revision 8
Calculation Note No. CN-RVCHP-04-4; Point Beach Units 1 and 2, HAUP Radiation Shield Structural Analysis; Revision 3
Calculation Note No. CN-RVCHP-04-6; Point Beach Head Assembly Upgrade Package (HAUP) Cable Bridge Structural Analysis; Revision 3
Calculation Note No. CN-RVCHP-04-9; Point Beach HAUP Head Lift Rig, Leg Extension and Spreader Evaluation; Revision 4
Calculation Note No. CN-RVCHP-04-10; Point Beach Units 1 and 2, HAUP - Missile Impact Analysis; Revision 2
Calculation Note No. CN-RVCHP-04-16; Point Beach Units 1 and 2, HAUP Structural Analysis of the RVLIS and RCGVS Supports; Revision 5
Calculation Note No. CN-RVCHP-04-17; Point Beach HAUP - CRDM Spacer Plate Analysis; Revision 1
Calculation Note No. CN-RVCHP-04-38; Point Beach Units 1 and 2, RVLIS Analysis; Revision 1
Calculation Note No. CN-RVCHP-04-44; Point Beach Units 1 and 2, HAUP Cooling Shroud Structural Analysis; Revision 6
Calculation Note No. CN-RVCHP-04-102; Seismic Support Platform Adjustment Plate Assembly; Revision 2
Calculation Note No. CN-RVCHP-05-15; Point Beach Units 1 and 2, HAUP Miscellaneous Hardware Seismic Evaluation; Revision 1
Calculation Note No. CN-RVCHP-05-22; Point Beach Units 1 and 2, HAUP Ladder Support Evaluation; Revision 1
Calculation Note No. CN-RVCHP-05-30; Point Beach Units 1 and 2, HAUP Radiation Shield Connection and Support Ring Evaluation; Revision 0
Drawing No. 1C51883; Point Beach Units 1 and 2, Head Assembly Upgrade Package, Cruciform Support Structure; Revision 0
Drawing No. 1C51884; Point Beach Units 1 and 2, Head Assembly Upgrade Package, Cruciform Support Structure; Revision 0
Drawing No. 1C52657; Point Beach Units 1 and 2, Head Assembly Upgrade Package, Angle; Revision 0

Drawing No. 10040C12; Point Beach Units 1 and 2, Head Assembly Upgrade Package, Ladder Support Weldment; Revision 1
Drawing No. 10017E59, Sheet 1; Point Beach Unit 2, Head Assembly Upgrade Package, Cooling System Power / Instrumentation Cable Routing Assembly; Revision 2
Drawing No. 10017E59, Sheet 6; Point Beach Unit 2, Head Assembly Upgrade Package, Cooling System Power / Instrumentation Cable Routing Assembly; Revision 2
Drawing No. 10021E90; Point Beach Units 1 and 2, Head Assembly Upgrade Package, Thermocouple Routing; Revision 0
Drawing No. 6474E44; Point Beach Units 1 and 2, Head Assembly Upgrade Package; Cable Support Assembly; Revision 0
MHI Drawing L5-01DP002; Replacement Reactor Vessel Closure Head, Closure Head Outline Drawing 2/2 (Basic Design Drawing); Revision 1
MHI Drawing L5-01DP101; Replacement Reactor Vessel Closure Head, Closure Head General Assembly 1/2; Revision 2
MHI Drawing L5-01DP112; Replacement Reactor Vessel Closure Head, Vent Pipe; Revision 1
MHI Drawing L5-01DP121; Replacement Reactor Vessel Closure Head, Ventilation Shroud Support Structure 1/3; Revision 1

Miscellaneous Modifications (71007)
Modification No. 03-046; Auxiliary Cranes to Support Unit 1 Reactor Vessel Head Assembly and Disassembly; Revision 0
Modification No. 03-048; Point Beach Unit 1, Replacement Reactor Head Insulation; Revision 0
Modification No. 03-050; Point Beach Unit 1, Containment Equipment Hatch Shield Wall Modification; Revision 0
Modification No. 03-051; Point Beach Unit 1, Analog Rod Position Indicator Cable and Connector Modification; Revision 0
Modification No. 03-052; Point Beach Unit 1, CRDM Cables and Connector Modification; Revision 0
Modification No. 03-053; Point Beach Unit 1, Core Exit Thermocouple System Mineral Insulated Cable Upgrade; Revision 0
Westinghouse Engineering Report DAR-ME-04-27; Qualification Summary for the Core Exit Thermocouple Cable and Connector Upgrade at Point Beach Units 1 and 2; Revision 0
Westinghouse Drawing No. E-PTBEACH-155-011; CET Cable Support Structure Assembly; Revision 1
Westinghouse Letter LTR-ME-05-10; Subject: Point Beach Unit 1, Core Exit Thermocouple Upgrade, CET Cable Support Structures Assembly Design Details; dated February 23, 2005

Corrective Action Reports
CA064699; Calculation CN-RVCHP-04-9 Acceptance Criteria Question; dated September 22, 2005
CAP067385; Errors in Calculation CN-HAUP-04-2, Rev. 2; dated September 29, 2005
CAPs-ACA-05-182-M025; Westinghouse Apparent Cause Analysis Report; Revision 1

CAPs Initiated as a Result of NRC Inspection
CAP066983; Westinghouse Design Specification 414A86 Correction; dated September 14, 2005
CAP067319; "B" RCP Access Ladder Hinges Clamped; September 27, 2005
CAP067349; EOP-0 and AOP-22 Diesel Loading Concerns During a Design Base Accident; September 28, 2005

CAP067350; NRC Identified ACE Weaknesses; September 28, 2005
CAP067352; NRC Questions Regarding Identification of Assumptions in Vendor Calculations; September 28, 2005
CAP067353; Missing Calculation Information and Page in EDMS; September 28, 2005
CAP067357; NRC Approves License Amendment Request (LAR) 239 - Battery Chargers; September 28, 2005
CAP067359; Errors Found in 1RMP9096; 10 CFR 50.59 Pre-Screening; Revision 29; September 28, 2005
CAP067397; Root Cause Investigation Technique Utilization; September 28, 2005
CAP067402; Calculation Process Improvement Opportunity; September 29, 2005
CAP067405; Control Room Temperature Effects on Instrument Uncertainties; September 29, 2005
CAP067407; Briefing for CA063225 Not Provided to All Required Personnel; September 29, 2005
CAP067413; Adequacy of Corrective Actions - NRC Observation; September 29, 2005
CAP067414; NRC Security Inspector Observations on Weapons Familiarization Training; September 29, 2005
CAP067415; Procedural Errors Being Tracked by Non-CAP Action Requests; September 29, 2005
CAP067416; NRC Inspector Observation on Drills for Personnel Who Support Security; September 29, 2005
CAP067418; Inadequate Resolution of NCVs; September 29, 2005
CAP067428; Review Search Process for Emergency Vehicles; September 30, 2005
CAP067445; NRC Information Notice Question (34) During PI&R Inspection; September 30, 2005
CAP067455; NRC Comments on Non-CAP Action Requests; September 30, 2005
CAP067603; TS 3.3.4.A - Placing Channel in Trip; October 5, 2005
CAP067613; Calculations and Specification Revisions Required for Additional Clarification; October 5, 2005
CAP067619; Consider Revising the Extent of Condition of ACE 1883, “Unplanned TSAC Entry Due to Air Leak on 2P-29 AFWP Mini Recirc Valve”; October 5, 2005
CAP067656; Timeliness in Determining Regulatory Reportability; October 6, 2005
CAP067675; Questions from NRC During ASME Section XI Inspection; October 6, 2005
CAP067693; Discrepancy Between TRM 4.8 and TS 5.5.8 Regarding SG “Repairs”; October 7, 2005
CAP067716; Observations from NRC During ASME Section XI Inspection; October 7, 2005
CAP067719; Observations from NRC During ASME Section XI Inspection; October 7, 2005
CAP067883; Question Regarding Scaffolding for SW-315 “C” CCW HX Outlet Valve; October 12, 2005
CAP067925; Old Work Order Tag Found Hanging on Protected Equipment; October 13, 2005
CAP067934; Original Signed Copy of OPR 155 Unavailable; October 13, 2005
CAP067947; Calculation Review Comments; October 14, 2005
CAP068070; Westinghouse Documents Contain Incorrectly Designated Non-Proprietary Info; October 18, 2005
CAP068075; Questions Associated with Refurbished Unit 1 “A” RHR Pump Motor; October 18, 2005
CAP068101; Evaluate Including Definitions of Frequency Notations in Environmental Manual; October 19, 2005

CAP068131; Air Sampler from Environmental Site E-04 Flow Rates Greater Than 32 LPM; October 20, 2005
CAP068146; Incorrect Reference in OPR000155; October 20, 2005
CAP068155; Meteorological Test Value Determined to be Out of Tolerance; October 20, 2005
CAP068159; Improvements to Radiation Work Permits; October 20, 2005
CAP068160; Environmental Air Sampling Filter Removed Incorrectly from Sample Head; October 20, 2005
CAP068161; Use of Rad Conditions in TEDE ALARA Reviews and RWPs; October 20, 2005
CAP068162; Stop Work Criteria and Radiological Controls; October 20, 2005
CAP068163; In-Progress ALARA Reviews; October 20, 2005
CAP068164; Documentation of Planned PCEs; October 20, 2005
CAP068165; ALARA and Work Planning; October 20, 2005
CAP068175; Routine Survey of Clean Side Tools/Equipment for Contamination; October 21, 2005
CAP068189; 2004 Annual Monitoring Report Submittal to NRC; October 21, 2005
CAP068193; ALARA Plan Improvements; October 21, 2005
CAP068194; Routine Survey Program Improvements; October 21, 2005
CAP068323; Inadequate Ladder Control in Containment - 8 foot level; October 26, 2005

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
AFW	Auxiliary Feedwater
AISC	American Institute of Steel Construction
ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
ASNT	American Society for Nondestructive Testing
CAP	Corrective Action Program Document
CFR	Code of Federal Regulations
CL	Checklist
CMP	Component Maintenance Program Procedure
CRDM	Control Rod Drive Mechanism
DAW	Dry Active Waste
DBD	Design Basis Document
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DOT	Department of Transportation
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EM	Environmental Manual
EPRI	Electric Power Research Institute
ET	Eddy Current Examination
FSAR	Final Safety Analysis Report
HAUP	Head Assembly Upgrade Package
HRA	High Radiation Area
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NP	Nuclear Plant Procedures Manual
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OPR	Operability Recommendation (Operability Evaluation)
PBNP	Point Beach Nuclear Plant
PI	Performance Indicator
PMT	Post-Maintenance Testing
psig	Pounds Per Square Inch Gage
PT	Dye Penetrant Examination
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCGVS	Reactor Coolant Gas Ventilation System
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RMP	Routine Maintenance Procedure

RHR	Residual Heat Removal
RP	Radiation Protection
RRVCH	Replacement Reactor Vessel Closure Head
RV	Reactor Vessel
RVCH	Reactor Vessel Closure Head
RVLIS	Reactor Vessel Level Indication System
RWP	Radiation Work Permit
SDP	Significance Determination Process
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SPEED	Spare Parts Equivalency Evaluation
SRO	Senior Reactor Operator
SW	Service Water
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
TRM	Technical Requirements Manual
TS	Technical Specifications
URI	Unresolved Item
UT	Ultrasonic Examination
VHP	Vessel Head Penetration
VHRA	Very High Radiation Area
WO	Work Order