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

February 10, 2010

EA-06-178

Mr. Larry Meyer
Site Vice-President
NextEra Energy Point Beach, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, NRC INTEGRATED INSPECTION REPORT 05000266/2009005; 05000301/2009005 AND STATUS OF CONFIRMATORY ORDER EA-06-178

Dear Mr. Meyer:

On December 31, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Point Beach Nuclear Plant, Units 1 and 2. The enclosed report documents the inspection results, which were discussed on January 6, 2010, with Mr. C. Trezise and members of your staff. The report also documents the status of Confirmatory Order EA-06-178, as it relates to your Point Beach Nuclear Plant.

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

Based on the results of this inspection, two NRC-identified and three self-revealed findings of very low safety significance were identified. Of these findings, four involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this Inspection Report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Point Beach Nuclear Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector Office at the Point Beach Nuclear Plant. The information that you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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 Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael Kunowski, Chief
Branch 5
Division of Reactor Projects

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

Enclosure: IR 05000266/2009005; 05000301/2009005
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServe

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-266; 50-301
License Nos:	DPR-24; DPR-27
Report No:	05000266/2009005; 05000301/2009005
Licensee:	NextEra Energy Point Beach, LLC
Facility:	Point Beach Nuclear Plant, Units 1 and 2
Location:	Two Rivers, WI
Dates:	October 1, 2009, through December 31, 2009
Inspectors:	S. Burton, Senior Resident Inspector R. Ruiz, Senior Resident Inspector (Acting) M. Thorpe-Kavanaugh, Resident Inspector (Acting) J. Jandovitz, Project Engineer J. Cassidy, Senior Health Physicist R. Jickling, Senior Emergency Preparedness Inspector D. Jones, Reactor Inspector D. McNeil, Senior Operations Engineer R. Edwards, Reactor Engineer J. Gilliam, Reactor Engineer E. Sanchez-Santiago, Reactor Engineer N. Feliz Adorno, Reactor Engineer
Approved by:	M. Kunowski, Chief Branch 5 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000266/2009005, 05000301/2009005; 10/01/2009 – 12/31/2009; Point Beach Nuclear Plant, Units 1 & 2; Maintenance Effectiveness, Operability Evaluations, Plant Modifications, Outage Activities, and Other Activities.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Also discussed is the status of Confirmatory Order EA-06-178. Five Green findings were either self-revealed or identified by inspectors in this inspection period. Four of the findings had associated Non-Cited Violations of NRC requirements, and one finding had no associated violation of regulatory requirements. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance for the failure to meet a commitment made in the Generic Letter (GL) 89-13 program. Specifically, the program states that biocide treatments at Point Beach are performed at least annually and are directly applied to the service water system for mussel control and eradication to prevent fouling of safety-related heat exchangers. However, the 2008 biocide treatment for mussel control was deferred until 2009. After the treatment in 2009, greater than expected tube blockage and reduced flow to safety-related heat exchangers due to mussels was identified. In response, the licensee adjusted flow through the affected heat exchangers and opened and cleaned the heat exchangers to remove mussels that caused the tube blockage. The licensee took corrective actions to ensure that future annual biocide treatments would be conducted annually.

  This finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," Table 4a, for the Mitigating Systems Cornerstone, dated January 10, 2008. The finding was determined to be of very low safety significance because the issue did not result in the actual loss of a safety function. This finding did not involve a violation of NRC regulatory requirements. The inspectors determined this performance deficiency was not indicative of current performance; therefore, no cross-cutting aspect was identified. (Section 1R12.1)

- Green. The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the failure to update the Safe Load Path Manual for the Unit 2 turbine building (SLP-3) as part of the mid-1990's modification that added the G-03 and G-04 emergency diesel

generators. Specifically, it was identified that SLP-3 allowed unrestricted load lifts over the Unit 2 turbine building truck bay area based upon a 1980's evaluation, and was not updated to reflect a modification that added safety-related cables for emergency diesel generators under the Unit 2 truck bay. Due to the close proximity of the “A” train cables to the “B” train cables, a loss of both trains of emergency alternating current (AC) power could result if the underground cables were disabled by a dropped load of sufficient magnitude. The licensee addressed the immediate concern by installing temporary steel plates over the affected area of the truck bay to provide adequate protection for upcoming heavy load lifts. Additionally, the licensee revised SLP-3 to require additional risk mitigation measures be taken prior to heavy load lifts in that area.

The finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of design control and adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," Table 4a, for the Mitigating Systems Cornerstone, dated January 10, 2008. The finding was determined to be of very low safety significance because the issue did not result in the actual loss of a safety function. This finding had a cross-cutting aspect in the area of problem identification and resolution, corrective action program component, because the staff did not take appropriate corrective actions to address safety issues in a timely manner, commensurate with their safety significance. Specifically, in 2008, when questions were raised by licensee staff regarding the adequacy of SLP-3, the SLP was not revised (P.1(d)). (Section 1R18.1)

- Green. A self-revealed finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for performing an Instrumentation and Control (I&C) procedure that was inappropriate to the circumstances, and resulted in the momentary loss of all available channels of reactor vessel level indication in the control room. As part of the immediate corrective actions, the licensee suspended the performance of the procedure and sent an operator into containment to verify reactor vessel level via the local standpipe level indicator and to ensure level indication was reestablished. Additionally, the licensee applied a work planning logic-tie to this activity to ensure the reactor was de-fueled prior to performing this calibration and was currently evaluating the need for revisions to the procedure.

  The finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of procedure quality and adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors assessed the significance of the finding in accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," and determined that this issue required a Phase 2 analysis since the finding increased the likelihood of a loss of reactor coolant system inventory. The inspectors and a senior reactor analyst determined through the analysis that this issue is best characterized as a finding of very low safety significance. This finding had a cross-cutting aspect in the area of human performance, work control component, in that the licensee did not appropriately coordinate work activities for the existing plant conditions to ensure the

operational impact on reactor vessel level indication while at a water level above reduced inventory was fully understood (H.3(b)). (Section 1R20.1)

Cornerstone: Barrier Integrity

- Green. A self-revealed finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for the failure to ensure adequate control of foreign material in accordance with the requirements of procedure NP 8.4.10, "Exclusion of Foreign Material from Plant Components and Systems." Specifically, on October 17, 2009, foreign material was discovered inside the 2SI-897B valve after the valve failed to properly stroke during the performance of procedure IT-215, "SI Valves - Cold Shutdown." The licensee took prompt corrective actions to repair the valve and perform an extent-of-condition review. Additionally, upon entering the issue into its corrective action program, the licensee performed a causal evaluation to determine any additional corrective actions.

  The finding was more than minor because it was associated with the Barrier Integrity Cornerstone attribute of human performance and adversely affected the associated cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, due to the interference caused by the foreign material inside the 2SI-897B valve, the valve would not have been able to perform its safety function to close during the initiation of the post-LOCA (loss of coolant accident) sump-recirculation phase of safety injection. The inspectors determined the finding could be evaluated in accordance with IMC 0609, “Significance Determination Process," Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings," Table 4a, dated January 10, 2008. The finding was determined to be of very low safety significance because the issue did not represent a degradation of the radiological barrier function provided for the control room, the auxiliary building, or the spent fuel pool; represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; represent an actual open pathway in the physical integrity of reactor containment (valves, airlocks, containment isolation system (logic and instrumentation)), and heat removal components; or involve an actual reduction in function of hydrogen ignitors in the reactor containment. No cross-cutting aspect was identified because the foreign material was determined to have been introduced into the system in the past and was not considered indicative of current performance. (Section 1R15.1)

Cornerstone: Public Radiation Safety

- Green. A self-revealed finding of very low safety significance and associated Non-Cited Violation of 10 CFR 20.1101(b) was identified for the failure to adequately control radioactive material to prevent its migration outside the radiologically controlled area (RCA), as required by licensee procedures. On May 21, 2009, a contract worker performing inspections of the main electrical transformers located outside the RCA picked-up a wadded-ball of debris (unmarked tape) and placed it in his front pants pocket. The debris was later found to be radioactively contaminated when the worker alarmed the protected area exit radiation monitors a few hours later as he attempted to leave the site. The tape was likely used to cover contaminated hoses that were previously used within the Point Beach RCA, but had escaped the licensee's control and migrated (blew) into the transformer area outdoors where it was found by the worker.

The licensee's storage of radioactive material in an outdoor satellite RCA and/or the licensee's radioactive material control practices during refueling outages when the containment building equipment hatch was open to the environment led to the escape of the material outside the RCA. The contractor's assigned work duties should not have involved exposure to radioactive material; consequently, the worker was unnecessarily exposed to radiation from the contaminated tape. A dose evaluation completed by the licensee's consultant determined that the effective dose equivalent to the worker's thigh from exposure to the contaminated ball of tape was approximately one mrem.
The licensee's corrective action called for expanded radiation protection oversight during movement of material in outdoor areas. Procedures were revised to include a post-outage walkdown of outdoor areas near the RCA yard. Additionally, the licensee planned to construct an enclosure so that storage/transfer of contaminated materials could be performed indoors.

The finding was more than minor because it impacted the program and process attribute of the Public Radiation Safety Cornerstone and adversely affected the cornerstone objective of ensuring adequate protection of public health and safety from exposure to radiation, in that, unnecessary radiation exposure was received by an individual from inadequately controlled radioactive material. The finding was determined to be of very low safety significance because: (1) it involved a radioactive material control problem that was contrary to NRC requirements and the licensee's procedure; and (2) the dose impact to a member of the public (the contract worker) within the licensee's restricted area was less than 5 millirem total effective dose equivalent. The cause of the radioactive material control problem involved a cross-cutting component in the human performance area for inadequate work control, in that, job site conditions including environmental conditions (high winds, night time work, etc.) impacted human performance and consequently, radiological safety, during movement of material/equipment in outdoor areas (H.3.(a)). (Section 4OA5.1)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 1 was at 100 percent power throughout the entire inspection period with the exception of a planned reduction in power during routine auxiliary feedwater (AFW) testing and an unplanned down-power to approximately 45 percent power on November 17, 2009, due to a lake grass influx and subsequent condenser cleaning evolution.

Unit 2 was at 100 percent power at the beginning of the inspection period, shut down to commence a refueling outage (U2R30) on October 15, 2009, restarted on December 5, and returned to 100 percent power on December 11. Unit 2 remained at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. The inspectors walked down accessible portions of risk-significant equipment and systems susceptible to cold weather freezing prior to the onset of severe cold weather. The inspectors walked down all accessible portions of the Units 1 and 2 facade buildings, which enclosed the reactor containments, and certain safety-related plant equipment inside the protected area. The inspectors reviewed the corrective action documents and work orders (WOs) written for identified problems. The inspectors also walked down areas that had a history of freeze problems to ensure that previous corrective actions were implemented. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one winter seasonal readiness preparations sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed a partial system walkdown of the spent fuel pool cooling system.

The inspectors selected this system based on its risk significance relative to the Reactor Safety Cornerstones at the time it was inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Final Safety Analysis Report (FSAR), Technical Specification (TS) requirements, outstanding WOs, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the system to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted one partial system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

During the Unit 2 refueling outage (U2R30), the inspectors performed a complete system alignment inspection of the residual heat removal (RHR) system to verify the functional capability of the system. This system was selected because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

These activities constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- fire zone 245 – Unit 1 electrical equipment room;
- fire zone 318 – cable spreading room;
- fire zone 775 – G-04 emergency diesel generator (EDG); and
- fire zone 301 – Unit 2 turbine building basement.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded, or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan.
The inspectors selected fire areas based on their overall contribution to internal fire risk and their potential to impact equipment that could initiate or mitigate a plant transient. The inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; fire detectors and sprinklers were unobstructed; transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted four quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On December 10, 2009, the inspectors observed a fire brigade activation in the north service building in response to a simulated electrical fire in the warehouse storeroom. Based on this observation, the inspectors completed an annual evaluation of the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient firefighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of

the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre-planned strategies; (9) adherence to the pre-planned drill scenario; and (10) drill objectives. Documents reviewed are listed in the Attachment to this report.

These activities constituted one annual fire protection inspection sample as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected important-to-safety plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the FSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the CAP to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area to assess the adequacy of flood protection and mitigation features, verify drains and sumps were clear of debris and were functional, and verify that the licensee complied with its commitments. Documents reviewed are listed in the Attachment to this report.

- G-01 and G-02 EDG rooms.

This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08P)

From November 2 through November 6, 2009, the inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS), steam generator (SG) tubes, AFW systems, risk-significant piping and components, and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, 1R08.3, 1R08.4, and 1R08.5 below constituted one ISI sample as defined in IP 71111.08-05.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors observed and reviewed records of the following nondestructive examinations mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications and defects detected were detected, and to determine if these were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- ultrasonic examination of steam generator shell-to-head circumferential weld SG-A-5R1 (Report No. 2009UT-22);
- liquid penetrant examination of reactor closure head peripheral control rod drive mechanism housings 28 and 32 welds (Report No. 2009PT-001); and
- ultrasonic examination of the reactor coolant system pressurizer surge nozzle inside radius section weld (Report No. 2009UT-057).

The inspectors reviewed records of the following nondestructive examinations conducted as part of the licensee's industry initiative inspection program for primary water stress corrosion cracking to determine if the examinations were conducted in accordance with the licensee's augmented inspection program, industry guidance documents, and associated licensee examination procedures, and if any indications and defects were detected, to determine if these were dispositioned in accordance with approved procedures and NRC requirements.

- visual examination of SG outlet nozzle-to-safe end weld RC-36-MRCL-AII-01A (Report No. 2009VT-031);
- visual examination of SG safe-end to "A" S/G inlet nozzle weld RC-34-MRCL-AI-05 (Report No. 2009VT-030);
- visual examination of SG "A" cold leg vent nozzle, (Report No. 2009VT-029); and
- visual examination of SG "A" hot leg vent nozzle, (Report No. 2009VT-028).

There were no examinations completed during the previous outage with relevant or recordable conditions or indications accepted for continued service. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed the following pressure boundary weld repairs completed on risk-significant systems since the beginning of the last refueling outage (RFO) to verify that the welding and any associated non-destructive examinations were performed in accordance with the Construction Code and ASME Code, Section XI. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure(s) were qualified in accordance with the requirements of Construction Code and the ASME Section IX Code.

- Work Order 00352831, “Replacement of an ASME Section III, Class 1, Excess Letdown Heat Exchanger (ELHX) 2HX-4 Outlet Drain Valve 2CV-D-11;” and
- Work Order 00352519, “Replacement of an ASME Section III, Class 1, RCS to P-10A/B Residual Heat Removal (RHR) Pump Suction Header Drain Valve 2RH-D-9.”

b. Findings

No findings of significance were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 2 reactor vessel head, a bare metal visual examination was required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D).

The inspectors reviewed records of the visual examination conducted on the Unit 2 reactor vessel head at penetrations 16, 32, and 40 to determine if the activities were conducted in accordance with the requirements of ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). In particular, the inspectors confirmed that:

- the required visual examination scope/coverage was achieved in accordance with the licensee's procedures; and
- the criteria for visual examination quality and instructions for resolving interference and masking issues were adequate.

No indications of potential through-wall leakage were identified by the licensee. Therefore, no NRC review was completed for this IP attribute.

The licensee did not perform any welded repairs to vessel head penetrations since the beginning of the preceding outage for Unit 2. Therefore, no NRC review was completed for this IP attribute.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

The inspectors observed and reviewed records of the licensee's initial BACC visual examinations and verified whether these visual examinations emphasized locations where boric acid leaks could cause degradation of safety-significant components.

The inspectors reviewed the following licensee evaluations of RCS components with boric acid deposits to determine if degraded components were documented in the CAP. The inspectors also evaluated corrective actions for any degraded RCS components to determine if they met the component Construction Code, ASME Section XI Code, and/or NRC-approved alternative.

- boric acid evaluation No. 09-219, 2SC-953 boric acid indications; and
- boric acid evaluation No. 09-173B, 2P-116, 2T-6C BA tank recirculation pump.

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to determine if the corrective actions completed were consistent with the

requirements of the ASME Section XI Code and 10 CFR Part 50, Appendix B, Criterion XVI.

- Work Order Package 0035658301, “Replace Pump Mechanical Seal;” and
- Work Request No. 00039792, “Adjust Packing to Last Value During AOV [air operated valve] Diagnostics.”

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

For the Unit 2 SGs, no examination was required pursuant to the TSs during the current RFO, U2R30. Therefore, no NRC review was completed for this IP attribute.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG-related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI/SG-related problems;
- the licensee had taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate 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.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On December 1, 2009, the inspectors observed a crew of licensed operators in the plant's simulator during just-in-time training for the Unit 2 startup to verify that operator

performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

.2 Annual Operating Test Results (71111.11B)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the individual Job Performance Measure operating tests, and the simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee from August 10 through October 1, 2009, as part of the licensee's operator licensing requalification cycle. These results were compared to the thresholds established in IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," and IP 71111.11, "Licensed Operator Requalification Program." Documents reviewed are listed in the Attachment to this report.

Completion of this section constituted one biennial licensed operator requalification inspection sample as defined in IP 71111.11B.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Containment Accident Fan Cooler Units (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant system:

- containment accident fan cooler units.

The inspectors reviewed events, such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems, and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly maintenance effectiveness sample as defined in IP 71111.12-05.

b. Findings

Failure to Meet Generic Letter (GL) 89-13 Program for Mussel Control

Introduction: The inspectors identified a Green finding for the failure to meet a GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," program commitment. Specifically, the licensee committed to implement mussel control methods to prevent fouling of safety-related heat exchangers. The 2008 annual biocide treatment for mussel control was not conducted and excessive tube blockage and reduced flow to safety-related heat exchangers due to mussels was identified after treatment in 2009.

Description: In response to GL 89-13, Point Beach developed a program documenting GL 89-13 commitments made to the NRC. Among those commitments was one to implement a biofouling program for mussel control and eradication to prevent fouling of safety-related components.

In 1999, the plant experienced significant mussel blockage events after not performing a biocide treatment in the previous year. In 2000, a licensee review of the mussel control strategy determined that two biocide treatments per year should be implemented so that mussels did not grow to a size that would block heat exchanger tubes when the shells detach from the piping. However, since that time, the plant performed only one biocide treatment per year, which empirically appeared adequate.

In August 2008, the annual mussel biocide treatment was deferred due to concerns by operations that the treatment would impact the operation of safety-related components. The decision, however, was made without consulting the GL 89-13 program engineer or the service water (SW) system engineer. It was possible to defer the treatment with minimal reviews since the WO was inappropriately categorized as a low Priority 4, "other," task.

The missed biocide treatment was documented in the CAP as Action Request (AR) 1133110, and corrective actions were implemented. None of the corrective actions discussed rescheduling the biocide treatment in 2008. Instead, the decision was made to perform the SW system biocide treatments for Unit 1 in spring and fall 2009, and for Unit 2 in fall 2009, just prior to the RFO. This schedule resulted in the Unit 2 SW system not being treated for about two years.

The Unit 2 mussel biocide treatment was completed on October 8, 2009. The following day, Unit 2 entered an unexpected Technical Specification Action Condition (TSAC) due to low flow in containment fan cooler (CFC) 2HX-15D. Flow was promptly increased by operations, and the TSAC was exited. Subsequently, during the Unit 2 outage (within a month of the biocide treatment) the component cooling water heat exchangers (CCWHXs), 2HX-12D and 0HX-12C (those affected by the Unit 2 biocide treatment), and the Unit 2 CFCs, 2HX-15A, 2HX-15C, and 2HX-15D, were opened for inspection. The CCWHXs acceptance criterion for the number of tubes blocked was 160 tubes. In 2HX-12D, 828 tubes were found blocked and in 0HX-12C, 507 tubes were blocked by mussel shells. The CFCs acceptance criterion for blocked tubes is 25 tubes. The plant identified 46, 107, and 77 tubes blocked by mussel shells in 2HX-15A, 2HX-15C, and 2HX-15D respectively. The 2HX-15B CFC was found acceptable. All heat exchangers were cleaned and mussel shells removed from the tubes. The inspectors reviewed the licensee's evaluation of past operability for the unacceptable CCWHXs, which concluded they had been operable during power operations, and found no issues.

Analysis: The inspectors determined that the failure to prevent fouling of safety-related heat exchangers in accordance with GL 89-13 commitments was a performance deficiency. Specifically, the deferral of the 2008 biocide treatment allowed mussels to grow to sufficient size that they would no longer pass through the heat exchanger tubes and the licensee could have reasonably been expected to prevent this based on past experience. The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of equipment performance and adversely affected the associated cornerstone objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform the 2008 biocide treatment affected the operability and design requirements of the CCWHXs and the CFCs.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04,

"Phase 1 - Initial Screening and Characterization of Findings," Table 4a, for the Mitigating Systems Cornerstone, dated January 10, 2008. The finding was determined to be of very low safety significance (Green) because the issue did not result in the actual loss of a safety function or loss of a single train for greater than its allowed TS time, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather initiating events. The inspectors determined this performance deficiency was not indicative of current performance and therefore no cross-cutting issue was identified.

Enforcement: No violation of regulatory requirements occurred because this issue represents a failure to implement an NRC commitment. This finding was entered into the licensee's CAP as AR 01158115 (FIN 05000266/2009005-01; 05000301/2009005-01).

In response to this issue, the licensee adjusted flow through the affected heat exchangers to address the immediate low flow conditions in addition to opening and cleaning all affected heat exchangers to remove mussel shells. In addition, the licensee raised the priority of future annual biocide treatments by designating them as preventive maintenance tasks. This re-designation will require more extensive reviews and approvals if a plan to defer an annual treatment arises.

.2 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant system:

- gas turbine system.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly maintenance effectiveness sample as defined in IP 71111.12-05.

b. Findings

No findings of significance were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- week of November 16, 2009, following the circulating water grass intrusion event and inverter trouble.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

These maintenance risk assessments and emergent work control activities constituted one sample as defined in IP 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Valve 2SI-897B Failure to Operate

a. Inspection Scope

The inspectors reviewed AR 01158812, written due to the failure of the 2SI-897B valve to operate during test procedure IT 215, "SI Valves - Cold Shutdown."

The inspectors selected this potential operability issue based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS past-operability and system functionality

were properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and FSAR to the licensee's evaluations to determine whether the components or systems were operable or functional. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted one sample as defined in IP 71111.15-05.

b. Findings

Failure to Ensure Adequate Control of Foreign Material in Safety-Related Systems

Introduction: A self-revealed finding of very low safety significance (Green) and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified for the failure to ensure adequate control of foreign material in accordance with the requirements of procedure NP 8.4.10, "Exclusion of Foreign Material from Plant Components and Systems."

Description: On October 17, 2009, foreign material was discovered inside the 2SI-897B valve after the valve failed to properly stroke closed during the performance of test procedure IT-215, "SI Valves - Cold Shutdown." Due to the tight clearances in the valve internals, once the foreign material became lodged in the valve trim cage, the valve plug became stuck while it was being stroked. Upon retrieval of the material by the licensee, it was discovered to be a pliable, black nylon material about ½-inch wide by 5-inches long, and appeared to be a cable-tie of unknown origin or variety. The licensee performed a boroscope inspection of the upstream and downstream piping for additional fragments of the material and none were found. The licensee performed a Condition Evaluation, AR 01158812, to determine the most likely source of the material.
The licensee concluded that the material most likely was introduced into the Unit 2 refueling water storage tank (RWST) where it flowed through a single-stage containment spray pump during testing to the safety injection (SI) pump test recirculation line.
The licensee also concluded that due to the pliable nature of the material, it was highly unlikely that the material would have damaged any pumps in its possible flow path.

Valve 2SI-897B is one of two normally-open, redundant, AOVs in series with valve 2SI-897A on the common Unit 2 SI pumps' test recirculation line (minimum-flow) to the RWST. Together, these normally-open valves perform the safety function to remain open during the SI injection phase to provide a minimum flow recirculation path to prevent damage to the SI pumps as a result of operating in a low flow or dead-headed condition. Since these valves were open, as designed, during modes in which the SI system was required to be operable, this safety-function, and the operability of the SI pumps was not impacted by this foreign material event.

The SI-897A and B valves also have a safety function to manually close during the transition from the injection phase of SI to the sump recirculation phase to prevent the flow of recirculation coolant into the RWST and potentially release radioactivity via the RWST's open vent. During a small-break loss of coolant accident scenario, the

RHR pumps would take suction from the containment sump during the recirculation phase and may be required to supply the SI pumps. If both SI-897A and B could not close at that time, containment sump water would be lost to the RWST via the minimum-flow line from the SI pumps, and radioactivity could be released to atmosphere. It was this safety function that was affected when the foreign material caused the mechanical binding of the 2SI-897B valve's internals and caused the valve to bind when 75 percent shut during the performance of IT-215 on October 17. However, since the 2SI-897A valve stroked satisfactorily on October 17, the safety function was maintained by this redundant valve. The last time that the 2SI-897B valve was successfully stroked was May 3, 2008, during the previous performance of IT-215. Additionally, once these valves are required to shut during an accident scenario, there are no sequences in which the valves would be required to re-open.

Analysis: The inspectors determined that the failure to ensure adequate control of foreign material in safety-related systems was contrary to the requirements of procedure NP 8.4.10, "Exclusion of Foreign Material from Plant Components and Systems," and was a performance deficiency.

The finding was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of human performance and adversely affected the associated cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, due to the interference caused by the foreign material inside the 2SI-897B valve, the valve would not have been able to perform its safety function to close during the initiation of the post-LOCA sump-recirculation phase of safety injection.

The inspectors determined the finding could be evaluated in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” Table 4a, containment barrier column, dated January 10, 2008. The finding was determined to be of very low safety significance (Green) because the issue did not represent a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool; represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; represent an actual open pathway in the physical integrity of reactor containment (valves, airlocks, containment isolation system (logic and instrumentation)), and heat removal components; nor involve an actual reduction in function of hydrogen ignitors in the reactor containment. No cross-cutting aspect was identified because the foreign material was determined to have been introduced into the system in the past and was not considered indicative of current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Specifically, procedure NP 8.4.10, required, in part, that maintenance activities preclude the introduction of foreign material into the SI system.

Contrary to this, prior to October 17, 2009, the licensee failed to accomplish activities affecting the quality of the SI system in accordance with the documented instructions and procedures associated with the exclusion of foreign material from safety-related plant equipment and systems, an activity affecting quality. Specifically, during a

previous work activity involving an open safety-related fluid system boundary, such as the RWST, the licensee failed to adequately control foreign material in accordance with procedure NP 8.4.10. Because this violation was of very low safety significance and was entered into the licensee's CAP as AR 011588112, "2SI897B Failed to Operate," this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000301/2009005-02).

In response to this issue, the licensee took prompt corrective actions to repair the valve and perform an extent-of-condition review, including a boroscope inspection of the upstream and downstream piping. Additionally, upon entering the issue into its CAP, the licensee performed a causal evaluation to determine the most probable location through which the foreign material entered and to develop appropriate corrective actions.

.2 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- AR 01161636; New Auxiliary Feedwater Line in Contact with Service Water Pipe;
- AR 01160262; 1HX-I5C CFC Flow Out-of-Limit Low per TS-33;
- AR 01158549; U2R20 Mode 3 UT [ultrasonic testing] Results – GL 08-01; and
- AR 01159784; Spent Fuel Pool Pump Suction Isolation Valve Stem Contacting Adjacent Pipe Insulation.

The inspectors selected this potential operability issue based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability and system functionality were properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and FSAR to the licensee's evaluations to determine whether the components or systems were operable or functional. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted four samples as defined in IP 71111.15-05.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification:

- modifications in Unit 2 turbine building to facilitate installation of new feedwater heaters.

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the FSAR, and the TSs, as applicable, to verify that the modification did not affect the operability or availability of the affected systems. The inspectors also compared the licensee's information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee's decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one temporary modification sample as defined in IP 71111.18-05.

b. Findings

Failure to Update Safe Load Path Manual to Include Safety-Related Cable Locations

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for the failure to ensure that the safe load path (SLP) and rigging manual for the Unit 2 turbine building crane (SLP-3), was updated as part of the major safety-related modification that added the G-03 and G-04 EDGs in 1995 and 1996.

Description: On October 14, 2009, the licensee generated AR 1158472, which captured an NRC-identified concern regarding the adequacy of SLP-3 with respect to the G-03 and G-04 modifications. Specifically, it was identified that SLP-3 allowed unrestricted load lifts over the U2 turbine building truck-bay area based upon evaluations performed in the early 1980s in response to NRC GL 81-07 "Control of Heavy Loads," and was not updated to reflect changes to the design of the facility when the G-03 and G-04 EDGs were installed and a modification added safety-related, risk-significant, cables under the Unit 2 truck bay in 1995 and 1996. These cables included the 4160-volt AC output cables from the train “B” EDGs (G-03 and G-04), and the 480-volt AC power cables to the train “A” EDGs’ (G-01 and G-02) fuel oil transfer pumps. Due to the close proximity of “A” and “B” train cables, a loss of both trains of emergency AC power could result if the underground cables were disabled by a postulated dropped load of sufficient

magnitude, such as a drop of the spare low pressure turbine rotor from the 66-foot elevation.

On September 30, 2009, the inspectors initially queried the licensee about upcoming Unit 2 feedwater (FW) heater replacement activities, with heavy load lifts scheduled for the Unit 2 truck bay during the fall 2009 RFO. Specifically, the inspectors inquired about the underground cables and whether or not the licensee had accounted for them in the preparations for the FW heater removals and installations with regard to potential load drop effects. When the inspectors asked for the licensee's justification for why a load drop analysis had not been performed, the licensee stated that it was unnecessary because SLP-3 allowed for unrestricted load lifts in that area.

When the inspectors examined the basis for SLP-3, it was noted that the plan for that area had remained essentially unchanged since its initial creation in the early 1980s, before the installation of the G-03 and G-04 EDGs in 1995 and 1996. It became evident to the inspectors that the SLP-3 had not been sufficiently revised to account for the existence of the risk-significant cables under the Unit 2 truck bay.

As a result of these discussions, the licensee determined that a 2 inch-thick layer of steel plates would be temporarily installed under the FW heater load lift area to provide adequate protection for the cables in the event of a load drop.

Analysis: The inspectors determined that the failure to update the SLP-3 as a part of the engineering change process when the diesel generator modification was implemented was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency.

The finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of design control and adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). In accordance with NRC IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated January 10, 2008, the inspectors conducted a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of system safety function or loss of a single train for greater than its allowed technical specification time, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather initiating events.

This finding has a cross-cutting aspect in the area of problem identification and resolution, CAP, because the staff did not take appropriate corrective actions to address safety issues in a timely manner, commensurate with their safety significance. Specifically, when AR 1122278 from February 2008 raised similar questions regarding the adequacy of SLP-3, no revision to the SLP resulted, despite one being drafted at the time. That AR was closed in April 2009 to no actions taken. Inspectors viewed that AR as a missed opportunity for the site to resolve the SLP-3 issue (P.1(d)).

Enforcement: 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 into specifications, drawings, procedures, and instructions. Contrary to this, from initial in-service installation of the G-03 and G-04

EDGs, to the point when SLP-3 was corrected in October 2009, the licensee failed to ensure that the design bases changes to the EDG system were correctly translated into specifications, drawings, procedures, and instructions. Because this violation was of very low safety significance and was entered into the licensee's CAP, as AR 1158472, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000301/2009005-03).

The licensee's corrective actions addressed the immediate concern by installing temporary steel plates over the affected area of the truck bay to provide adequate protection for upcoming heavy load lifts. Additionally, the licensee revised SLP-3 to require additional risk mitigation measures be taken prior to any future heavy load lifts in that area.

.2 Permanent Plant Modifications

a. Inspection Scope

The following engineering design packages were reviewed and selected aspects were discussed with engineering personnel:

- GSI 191 (Generic Safety Issue) modifications EC [Engineering Change] 13601 – RCP [Reactor Coolant Pump] S/G, and RCS Loops Piping Insulation Replacement - Unit 2, and EC 12601 - Additional Sump Strainer Modules - Unit 2; and
- EC 11542; Unit 2 Main Generator Circuit Breaker Addition.

These documents and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and proper update of relevant procedures, design, and licensing documents. The inspectors observed ongoing and completed work activities to verify that installation was consistent with the design control documents. The first sample was for the modification that replaced the Unit 2 main generator circuit breaker, and the other sample was for GSI-191 modifications in the Unit 2 containment that replaced insulation and added additional sump strainer modules. Documents reviewed are listed in the Attachment to this report.

Specifically, the inspectors conducted a walkdown of the strainer assemblies during the fall 2009 RFO for Unit 2. The engineering design packages were associated with the licensee's response to GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors." The licensee's implementation of commitments documented in its initial responses to GL 2004-02 was previously reviewed in accordance with temporary instruction (TI) 2515/166, "Pressurized Water Reactor Containment Sump Blockage." The closure of this TI in the summer 2008, documented in NRC Inspection Report (IR) 05000266/2008003; 05000301/2008003, indicated that the licensee had received approval for an extension for GL 2004-02 corrective actions.

In July 2008, after the establishment of an industry head loss test protocol, the licensee conducted additional testing using the revised test methodologies. During this testing, the licensee determined that the original containment sump strainer modification of

11 strainer modules per train, which had already been installed, did not meet test acceptance criteria. As a result, the licensee installed three additional strainers modules per train, added debris interceptors, removed fibrous insulation in the fall 2008 RFO for Unit 1, and planned similar modifications for the fall 2009 RFO for Unit 2. The purpose of the modification was to obtain additional net positive suction head margin for the residual heat removal pumps. However, prototypical testing of the debris interceptors in January 2009 indicated that the efficiency of the debris interceptors was not as high as required. In order to address this issue and recent concerns regarding the assumed destruction zone of influence for fibrous insulation, the licensee planned to remove additional fibrous insulation and revise the debris generation and transport analyses accordingly. Specifically, the licensee developed an additional modification that reduced the amount of fibrous insulation debris by replacing the existing insulation with metallic reflective insulation on reactor coolant pumps bowl assemblies, portions of steam generators, and portions of reactor coolant system loop piping.

The licensee requested and received NRC approval for an extension for GL 2004-02 corrective actions to June 30, 2010, for Unit 1, and June 20, 2011, for Unit 2. Since the closure of TI 2515/166, the licensee has completed the following actions:

- installation of an additional three strainer modules per train to increase the overall surface area in Units 1 and 2;
- installation of debris interceptors to reduce the quantity of suspended debris that could be transported to the screen surface in Unit 1;
- structural reinforcement of the strainer assemblies to accommodate an increased differential pressure in Unit 2;
- extension of the refueling cavity drain line away from the strainers in order to prevent water from spilling on or near the strainers and potentially causing air ingestion in Units 1 and 2; and
- initiated the fibrous insulation reduction effort in Units 1 and 2.

The outstanding actions are:

- complete the fibrous insulation reduction effort during the spring 2010 RFO for Unit 1 and the spring 2011 RFO for Unit 2; and
- update the licensing bases as required.

This inspection constituted two permanent plant modification samples as defined in IP 71111.18-05.

b. Findings

Potential Failure To Adequately Evaluate Seismic II/I Concerns For Units 1 And 2 "B" Containment Sump Strainers

Introduction: The inspectors identified an unresolved item (URI) regarding the "B" containment sump strainers for Units 1 and 2. Specifically, the inspectors questioned

whether the ventilation ducts located above containment sump strainers were adequately evaluated with respect to seismic II/I considerations.

Description: On October 27, 2009, the inspectors performed a walkdown of the containment sump strainers of Unit 2 and noted a ventilation duct located above the "B" containment sump strainer. The inspectors were concerned that during a seismic event the structure could collapse and affect the strainer's ability to fulfill its accident mitigating function. Specifically, if the ventilation duct and its support structure collapsed, the structural integrity of the sump strainer could be compromised or the failed duct and support could block the strainers. The sump strainers are relied upon to simultaneously maintain an adequate post-loss-of-coolant-accident suction source while preventing debris from entering the emergency core cooling system.

The licensee's immediate documentation search on the seismic evaluation of the ventilation duct was unsuccessful. The licensee initiated AR 01159937. The licensee also determined that the same condition existed in Unit 1 and performed a prompt operability determination for the Unit 1 "B" strainer.

The licensee later determined that the installation modification documentation for Unit 1, Engineering Change (EC) 1602, indicated that the modification did not require analysis of non-seismic components located over or adjacent to seismic components because there was no evidence of a potential seismic II/I concern at the time the modification was completed. Specifically, a seismic interaction walkdown was required in the installation work plan prior to the installation of the strainers. The walkdown was completed by two civil engineers who were Seismic Qualification Users Group (SQUG) qualified.
The licensee determined, through discussions with the engineers who performed the walkdown, that the ventilation ducts were reviewed. Based on these facts, the licensee concluded that: (1) the ventilation ducts were seismically evaluated; (2) the evaluation determined that there are no seismic II/I concerns; and (3) that this is a documentation issue. The same conclusions applied to Unit 2.

However, the inspectors were concerned with the use of SQUG methodology to evaluate the seismic II/I interactions with respect to the duct ventilation and the strainer. Specifically, the inspectors questioned whether this methodology could be applied to ventilation ducts because this type of structure did not appear in the equipment classes of the implementing procedure for SQUG. As a result of the inspectors' questions, the licensee performed a prompt operability determination, in accordance with EN-AA-203-1001 that determined the Unit 1 "B" sump strainer was operable. The basis for this conclusion was documented in EC 14790. This EC performed a structural analysis that concluded that the ventilation duct support structure would be able to support loads induced by a seismic event. Again, this evaluation applied to Unit 2.

In addition, the inspectors noted that the FSAR, Appendix A5.6, stated that "Modified, new, or replacement equipment classified as Seismic Class I may be seismically designed and verified (after installation) for seismic adequacy using seismic experience data in accordance with a methodology developed by the SQUG." It was not clear whether this statement applied for all new modifications or to the replacement of previously SQUG-qualified equipment with similar equipment.

The inspectors were also concerned with the level of documentation maintained by the licensee for the walkdowns performed using the SQUG methodology. Specifically, the

inspectors noted that the documentation did not provide the necessary details to permit independent auditing of the inferences or conclusions.

This issue is unresolved pending further NRC review of the licensing basis for the use of SQUG methodology and determination of further NRC actions to resolve the issues (URI 05000266/2009005-04; 05000301/2009005-04).

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing (PMT)

a. Inspection Scope

The inspectors reviewed the following PMT activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- auxiliary feedwater and containment spray systems post-weld testing;
- TS-82 monthly EDG run PMT for annual maintenance and failed level switch in sump tank;
- RHR pump 2P-10B PMT after oil leak repair; and
- Unit 2 polar crane PMT following cable replacement.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TSs, the FSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted four post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan and contingency plans for the Unit 2 RFO, conducted October 15 - December 5, 2009, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the Outage Safety Plan for key safety functions and compliance with the applicable TS when taking equipment out-of-service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and Outage Safety Plan requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls, including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Refueling activities, including fuel handling and activities to detect fuel assembly leakage.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to RFO activities.

This inspection constituted one refueling outage sample as defined in IP 71111.20-05.

b. Findings

Momentary Loss of Unit 2 Reactor Vessel Level Indication in the Control Room

Introduction: A finding of very low safety significance and associated Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when the licensee performed an Instrumentation and Control (I&C) procedure that was inappropriate to the circumstances and caused the momentary loss of all available channels of reactor vessel level indication in the control room.

Description: On October 19, 2009, operators were maintaining reactor vessel inventory at 70 percent in preparation for head disassembly and placed the reduced inventory reactor vessel level transmitters (LT), LT-447 and LT-447A, into service for level indication. Subsequently, the operators authorized maintenance to perform I&C procedure 2ICP 04.023-1, "Reactor Vessel Level Outage Calibration." The purpose of the procedure was to calibrate reactor vessel wide and narrow range level transmitters, 2LT-494, 2LT-495, 2LT-496, and 2LT-497.

During the performance of this procedure, following calibration of 2LT-494, the technician valved in the transmitter. This allowed a flow path to exist between the variable and common reference legs of all the reactor vessel level indicators, which caused a perturbation on the level indication for 2LT-447 and 2LT-447A, and subsequent momentary loss of reactor vessel level indication in the control room. The operators took immediate action to suspend the performance of the I&C procedure and sent an operator into containment to verify reactor vessel level via the local standpipe level indicator (LI), LI-447B, and to ensure level indication was reestablished.

The I&C procedure contained instructions to notify the control operator that perturbations on the reactor vessel level indicators 2LT-447 and 2LT-447A may occur and required operators to verify reduced inventory conditions were not in effect. However, the procedure did not contain cautions or prerequisite conditions for the given conditions of being at 70 percent inventory and time-to-boil (TTB) of 17 minutes, essentially the same TTB as a reduced inventory condition. No additional barriers were in place to prevent the procedure from being performed at the same time as preparations for head disassembly.

Analysis: A performance deficiency was identified when the licensee performed an I&C procedure that was inappropriate for the circumstances of reactor vessel level at 70 percent and a TTB of 17 minutes; thereby, causing a loss of all available channels of reactor vessel level indication in the control room. The finding was more than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

In accordance with NRC IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 3, dated May 25, 2004, the inspectors conducted a Phase 1 SDP screening and determined that the finding required a Phase 2 analysis since the finding increased the likelihood of a loss of RCS inventory based on loss of reactor vessel level indication in the control room (Sections II(A)(2) II(B)(3) of Checklist 3).

A Region III senior reactor analyst (SRA) performed the assessment using Appendix G, Attachment 2, "Phase 2 Significance Determination Process Template for PWR During Shutdown," dated February 28, 2005. The SRA determined this to be a precursor to an initiating event (a loss of level control precursor - LOLC). The plant operating state (POS) was determined to be "POS 1" (vessel head on and RCS closed). The initiating event likelihood for LOLC using Table 1, "Initiating Event Likelihood (IELs) for LOLC Precursors" was "1," since the time to RHR loss was greater than two hours and action to recover RHR could be identified and performed within half of the time to RHR loss. The SRA considered this to be an overly conservative value considering that there was

no actual loss of RCS inventory, only momentary loss of indication. To better estimate the IEL, the SRA performed an analysis using the SPAR-H Human Reliability Analysis Method, NUREG/CR-6883, September 2004.

For diagnosis of potential loss of level control, the analyst assumed available time to be expansive. For action, the analyst assumed stress to be high. All other performance shaping factors were assumed to be nominal. The resultant value of 3E-3 was assumed as the initiating event likelihood.

Using Appendix G, Attachment 2, Worksheet 1, "SDP for a PWR Plant - Loss Level Control in POS 1 (RCS Closed)," the SRA evaluated the remaining mitigating capability credit to reflect equipment availability and the time available to complete tasks prior to core damage. The most significant core damage sequences involved loss of steam generator cooling and failure of RCS injection and bleed before core damage. The combined sequences had a risk significance of about 3E-8. Therefore, the SRA determined that this issue is best characterized as a finding of very low safety significance (Green).

The finding had a cross-cutting aspect in the area of human performance, work control aspect, in that the licensee did not appropriately coordinate work activities for the existing plant conditions to ensure the operational impact on reactor vessel level indication while at a water level near reduced inventory (H.3(b)).

Enforcement: Title 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and be accomplished in accordance with these instructions, procedures or drawings. Contrary to this, the licensee performed an I&C procedure that was inappropriate to the circumstances. Specifically, I&C procedure 2ICP 04.023-1, disabled all control room reactor vessel level indication while the reactor coolant system was at 70 percent reactor vessel level. As a result, the indication of reactor water level in the reduced inventory range was momentarily lost in the control room, which was not appropriate for the current plant condition. Because this violation was of very low safety significance and it was entered into the licensee's CAP (AR 01158914), this violation is being treated as an NCV consistent with section VI.A.1. of the NRC Enforcement Policy (NCV 05000301/2009005-05).

The licensee took immediate action to suspend the performance of the I&C procedure and sent an operator into containment to verify reactor vessel level via the local standpipe level indicator (LI-447B) to ensure level indication was reestablished. Additionally, the licensee has applied work planning logic to this activity to ensure the reactor is defueled prior to beginning the calibration and is evaluating necessary revisions to the I&C procedure.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- Unit 2 ORT 3A/B EDG loss of offsite power loss of coolant accident routine test;
- OSHA [Occupational Safety and Health Administration] polar crane inspection; and
- Unit 2 turbine-driven AFW pump and valve inservice test.

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of ASME Code Section XI, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;

- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted two routine surveillance testing samples and one inservice testing sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System (ANS) Evaluation (71114.02)

.1 ANS Evaluation

a. Inspection Scope

The inspectors reviewed documents and conducted discussions with Emergency Preparedness (EP) staff and management regarding the operation, maintenance, and periodic testing of the ANS in the Point Beach Plant's plume pathway Emergency Planning Zone. The inspectors reviewed monthly trend reports and the daily and monthly operability records from October 2007 through November 2009. Information gathered during document reviews and interviews was used to determine whether the ANS equipment was maintained and tested in accordance with Emergency Plan commitments and procedures. Documents reviewed are listed in the Attachment to this report.

This alert and notification system inspection constituted one sample as defined in IP 71114.02-05.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation Testing (71114.03)

.1 ERO Augmentation Testing

a. Inspection Scope

The inspectors reviewed and discussed with plant EP management and staff the emergency plan commitments and procedures that addressed the primary and alternate methods of initiating an ERO activation to augment the on-shift ERO as well as the provisions for maintaining the station's ERO qualification and team lists. The inspectors reviewed reports and a sample of CAP records of unannounced off-hour augmentation tests and pager test, which were conducted between March 2008 and September 2009,

to determine the adequacy of the drill critiques and associated corrective actions. The inspectors also reviewed a sample of the EP training records of approximately 37 ERO personnel, who were assigned to key and support positions, to determine the status of their training as it related to their assigned ERO positions. Documents reviewed are listed in the Attachment to this report.

This emergency response organization augmentation testing inspection constituted one sample as defined in IP 71114.03-05.

b. Findings

No findings of significance were identified.

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

.1 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

Since the last NRC inspection of this program area, emergency action level and Emergency Plan revisions were implemented based on the licensee's determination, in accordance with 10 CFR 50.54(q), that the changes resulted in no decrease in effectiveness of the Plan, and that the revised Plan as changed continues to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. Revisions to the emergency action levels and emergency plan reviewed by the inspectors included: 1) EP 2.0, Revision 46; 2) EP 6.0, Revisions 51 and 52; 3) Appendix M, Revision 2; and 4) EPIP 1.2.1, Revision 3. The inspectors conducted a sampling review of the Emergency Plan changes and a review of the Emergency Action Level changes to evaluate for potential decreases in effectiveness of the Plan. However, this review does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety.

This emergency action level and emergency plan changes inspection constituted one sample as defined in IP 71114.04-05.

b. Findings

No findings of significance were identified.

1EP5 Correction of EP Weaknesses and Deficiencies (71114.05)

.1 Correction of EP Weaknesses and Deficiencies

a. Inspection Scope

The inspectors reviewed a sample of Nuclear Oversight 2008 and 2009 audits of the Point Beach EP program to determine that the independent assessments met the requirements of 10 CFR 50.54(t). The inspectors also reviewed critique reports and samples of CAP records associated with the 2008 biennial exercise, as well as various EP drills conducted in 2007, 2008, and 2009, in order to determine whether the licensee fulfilled drill commitments and to evaluate the licensee's efforts to identify and resolve

identified issues. The inspectors reviewed a sample of EP items and corrective actions related to the facility's EP program and activities to determine whether corrective actions were completed in accordance with the site's CAP. Documents reviewed are listed in the Attachment to this report.

This correction of emergency preparedness weaknesses and deficiencies inspection constituted one sample as defined in IP 71114.05-05.

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 Review of Licensee Performance Indicators (PIs) for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's Occupational Exposure Control Cornerstone PI to determine whether the conditions resulting in any PI occurrences had been evaluated and whether identified problems had been entered into the licensee's CAP for resolution.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit (RWP) Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas, and airborne radioactivity areas in the plant to determine if radiological controls including surveys, postings, and barricades were acceptable:

- Auxiliary Building;
- Containment Building;
- Spent Fuel Pool.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed the RWPs and work packages used to access these areas and other high radiation work areas. The inspectors assessed the work control instructions and control barriers specified by the licensee. Electronic dosimeter alarm setpoints for

both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. The inspectors interviewed workers to verify that they were aware of the actions required if their electronic dosimeters noticeably malfunctioned or alarmed.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors walked down and surveyed (using an NRC survey meter) these areas to verify that the prescribed RWP, procedure, and engineering controls were in place; that licensee surveys and postings were complete and accurate; and that air samplers were properly located.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed RWPs for airborne radioactivity areas to verify barrier integrity and engineering controls performance (e.g., high-efficiency particulate air ventilation system operation) and to determine if there was a potential for individual worker internal exposures in excess of 50 millirem committed effective dose equivalent (EDE). There were no airborne radioactivity work areas during the inspection period.

Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and had provided appropriate worker protection.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors assessed the adequacy of the licensee's internal dose assessment process for internal exposures in excess of 50 millirem committed EDE. There were no internal exposures greater than 50 millirem committed EDE.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors also reviewed the licensee's physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) stored within the spent fuel pool or other storage pools.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed a sample of the licensee's self-assessments, audits, Licensee Event Reports (LERs), and Special Reports related to the access control program to verify that identified problems were entered into the CAP for resolution.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed corrective action reports related to access controls and any high radiation area radiological incidents (issues that did not count as PI occurrences identified by the licensee in high radiation areas less than 1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being 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 effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation/consideration of risk significant operational experience feedback.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors evaluated the licensee's process for problem identification, characterization, and prioritization and verified that problems were entered into the CAP and resolved. For repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, the inspectors verified that the licensee's self-assessment activities were capable of identifying and addressing these deficiencies.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if any of these PI events involved dose rates in excess of 25 R/hr at 30 centimeters or in excess of 500 R/hr at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. Unintended exposures exceeding 100 millirem total EDE (or 5 rem shallow dose equivalent or 1.5 rem lens dose equivalent) were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following three jobs that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas for observation of work activities that presented the greatest radiological risk to workers:

- insulation activities;
- reactor coolant pump activities; and
- core barrel movement activities.

The inspectors reviewed radiological job requirements for these activities, including RWP requirements and work procedure requirements, and attended As-Low-As-Is-Reasonably-Achievable (ALARA) job briefings.

This inspection constituted one sample as defined in IP 71121.01-5.

Job performance was observed with respect to the radiological control requirements to assess whether radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings. The inspectors evaluated the adequacy of radiological controls, including required radiation, contamination, and airborne surveys for system breaches; radiation protection job coverage, including any applicable audio and visual surveillance for remote job coverage; and contamination controls.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed radiological work in high radiation work areas having significant dose rate gradients to evaluate whether the licensee adequately monitored exposure to personnel and to assess the adequacy of licensee controls. These work areas involved areas where the dose rate gradients were severe, thereby increasing the necessity of providing multiple dosimeters or enhanced job controls.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.5 High Risk Significant, High Dose Rate, High Radiation Area, and Very High Radiation Area Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager concerning high dose rate, high radiation area, and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to assess whether any procedure modifications substantially reduced the effectiveness and level of worker protection.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors discussed with radiation protection supervisors the controls that were in place for special areas of the plant that had the potential to become very high radiation areas during certain plant operations. The inspectors assessed if plant operations required communication beforehand with the radiation protection group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors conducted plant walkdowns to assess the posting and locking of entrances to high dose rate, high radiation areas, and very high radiation areas.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation safety work requirements. The inspectors evaluated whether workers were aware of any significant radiological conditions in their workplace, of the RWP controls and limits in place, and of the level of radiological hazards present. The inspectors also observed worker performance to determine if workers accounted for these radiological hazards.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed radiological problem reports for which the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. Problems or issues with planned or completed corrective actions were discussed with the Radiation Protection Manager.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation protection technician performance with respect to radiation safety work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed radiological problem reports for which the cause of the event was radiation protection technician error to determine if there was an observable pattern

traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

.1 Radiological Work Planning

a. Inspection Scope

The inspectors compared the results achieved (including dose rate reductions and person-rem used) with the intended dose established in the licensee's ALARA planning for GSI-191 insulation removal activities. Reasons for inconsistencies between intended and actual work activity doses were reviewed.

This inspection constituted one required sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

.2 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The licensee's process for adjusting exposure estimates or re-planning work (when unexpected changes in scope, emergent work, or higher than anticipated radiation levels were encountered) was evaluated. This included determining whether adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles or whether they resulted from failures to adequately plan or to control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process.

This inspection constituted one required sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolutions

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and Special Reports related to the ALARA program since the last inspection to determine if the licensee's overall audit program's scope and frequency for all applicable areas under the Occupational Radiation Safety Cornerstone met the requirements of 10 CFR 20.1101(c).

This inspection constituted one required sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, and Occupational Radiation Safety

4OA1 PI Verification (71151)

.1 Mitigating Systems Performance Index (MSPI) - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Heat Removal System PI for Unit 1 and Unit 2 for the third quarter 2008 through the second quarter of 2009. To determine the accuracy of this PI data, definitions and guidance contained in the Nuclear Energy Initiative (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator narrative logs, corrective action reports, event reports, MSPI derivation reports, and NRC integrated IRs for October 2008 through June 2009 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's CAP database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI heat removal system samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 MSPI - RHR System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI Index - RHR System PI Unit 1 and Unit 2 for the third quarter 2008 through the second quarter of 2009. To determine the accuracy of the PI data, definitions and guidance contained in NEI 99-02, Revision 5, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports, and NRC Integrated IRs for October 2008 through June 2009, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's

issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI RHR system sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Drill/Exercise Performance

a. Inspection Scope

The inspectors sampled licensee submittals for the Drill/Exercise PI for the fourth quarter 2008 through third quarter 2009. To determine the accuracy of the PI data, definitions and guidance contained in NEI 99-02, Revision 5, were used. The inspectors reviewed the licensee's records associated with the PI to verify that the licensee accurately reported the PI in accordance with relevant procedures and the NEI guidance. Specifically, the inspectors reviewed licensee records and processes including procedural guidance on assessing opportunities for the PI, and assessments of PI opportunities during pre-designated control room simulator training sessions, performance during the 2008 biennial exercise, and performance during other drills. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes one drill/exercise performance sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.4 ERO Drill Participation

a. Inspection Scope

The inspectors sampled licensee submittals for the ERO Drill Participation PI for the fourth quarter 2008 through third quarter 2009. To determine the accuracy of the PI data, definitions and guidance contained in NEI 99-02, Revision 5, were used. The inspectors reviewed the licensee's records associated with the PI to verify that the licensee accurately reported the indicator in accordance with relevant procedures and the NEI guidance. Specifically, the inspectors reviewed licensee records and processes including procedural guidance on assessing opportunities for the PI; performance during the 2008 biennial exercise and other drills; and revisions of the roster of personnel assigned to key emergency response organization positions. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes one ERO drill participation sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.5 Alert and Notification System

a. Inspection Scope

The inspectors sampled licensee submittals for the ANS PI for the fourth quarter 2008 through third quarter 2009. To determine the accuracy of the PI data, definitions and guidance contained in NEI 99-02, Revision 5, were used. The inspectors reviewed the licensee's records associated with the PI to verify that the licensee accurately reported the indicator in accordance with relevant procedures and the NEI guidance. Specifically, the inspectors reviewed licensee records and processes including procedural guidance on assessing opportunities for the PI and results of periodic ANS operability tests. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes one ANS sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.6 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences PI for the third quarter 2008 through the third quarter 2009. To determine the accuracy of the PI data, definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6 (issued October 2009), were used. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational radiological occurrences sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.7 Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrences PI for the third quarter 2008 through the third quarter 2009. The inspectors used PI definitions and guidance contained in NEI 99-02, Revision 6, to determine the accuracy of the PI data. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates between the third quarter 2008 and the third quarter 2009 to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one radiological effluent technical specification/offsite dose calculation manual radiological effluent occurrences sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Items Entered into the CAP

a. Inspection Scope

As part of the various baseline IPs discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the attached List of Documents Reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings of significance were identified.

.2 Daily CAP Reviews

a. Inspection Scope

To assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the six-month period of July through December 2009, although some examples extended beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule evaluations. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted a single semi-annual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 (Closed) URI 05000266/2009004-01; 05000301/2009004-01, Failure to Control Radioactive Material Within the Radiologically Controlled Area Resulting in Unnecessary Dose to Worker

a. Inspection Scope

The inspectors reviewed additional information, including the licensee's dose assessment, for an incident on May 21, 2009, that involved a contract worker who received unnecessary radiation exposure while performing inspections of the licensee's electrical transformers. The inspection was completed through in-office review of documents generated by the licensee. The review included discussions with various members of the licensee's staff, both in person and by teleconference. A dose assessment completed by the licensee's consultant was reviewed and independently validated by NRC staff. Documents reviewed are listed in the Attachment to this report. This URI is closed.

b. Findings

Introduction: A self-revealed finding of very low safety-significance (Green) and an associated NCV of 10 CFR 20.1101(b) was identified for the failure to adequately control radioactive material and prevent its inadvertent migration outside the RCA, as required by licensee procedure.

Description: On May 21, 2009, a contract worker alarmed the security gatehouse portal radiation monitors while attempting to exit the protected area following completion of transformer inspections. The transformers are located outside the RCA but within the protected area. Investigation by the licensee disclosed that the worker picked-up debris (pieces of unmarked tape wadded-together to the size of a billiard ball) found lying near one of the transformers, placed the debris in the front trouser pocket, and approximately two hours later, after completing assigned work duties, alarmed the radiation monitors upon attempted egress from the protected area. The ball of tape was subsequently identified by the licensee to be radioactively contaminated, primarily with cobalt-60. The licensee's radiation measurements of the wadded tape ball using portable survey instruments identified contact gamma and beta dose rates of about 6 mrem/hour and 500 mrad/hour, respectively. Low levels of contamination were also identified on the worker’s clothing, some personal items, and left hand. No contamination or other contaminated debris was identified during follow-up surveys in/near the transformers.

The licensee performed an apparent cause evaluation (ACE) that determined the tape was likely used to cover the ends of piping or contaminated hoses because one of the pieces of tape had a two-inch diameter circular marking. During RFOs, the yard area outside the facade access into the containment building was the transfer point for materials/equipment into and out of the containment building. The containment building equipment hatch was sometimes opened to the environment to facilitate movement of equipment and supplies. The licensee surmised that since outage equipment/material was transferred from the containment building at night and during windy conditions and at times when portions of the outdoor RCA barrier fence was removed, the material could have escaped the licensee's control without notice and blown into the transformer area.

The contract worker frequented the site on an approximate monthly basis or less, spending a few hours to inspect and perform minor maintenance on the licensee's main power transformers. The individual had not entered an RCA while onsite that day, the work was not governed by a RWP, and the individual was not provided dosimetry. The worker's assigned duties did not involve exposure to radiation and the individual should not have come into contact with any radioactive material. The individual completed the licensee's Plant Access Training required for unescorted access into the protected area but not Radiation Worker Training required for access into RCAs. The licensee had classified the worker as a member of the public, as provided in its Plant Access Training, because the individual had no need to enter RCAs and the worker's dose was expected to be well within the public dose limits of 10 CFR 20.1301. Consequently, the NRC concluded that the dose received by the contractor from exposure to the contaminated tape was deemed to be "public dose" as defined in 10 CFR 20.1003.

A dose evaluation completed by the licensee's consultant determined that the EDE to the worker's thigh from exposure to the contaminated ball of tape was approximately one mrem. The evaluation was independently reviewed by NRC staff and found to be technically adequate and consistent with guidance provided in NRC Regulatory Issue Summary 2003-04, "Use of Effective Dose Equivalent in Place of the Deep Dose Equivalent in Dose Assessments." The licensee's corrective action called for expanded radiation protection staff oversight during movement of material in/out of the containment building during outages and for any movement of radioactively contaminated materials in outdoor areas. Also, a radiation protection procedure was revised to require a post-outage walkdown of outdoor RCA boundaries to ensure no material escaped. Additionally, the licensee planned to construct an enclosure so that storage/transfer of contaminated materials could be performed indoors.

Analysis: The inspectors determined that the failure to adequately control radioactive material and prevent its migration outside the RCA was a performance deficiency. The inspectors concluded that the cause of the performance deficiency was reasonably within the licensee's ability to foresee and correct and should have been prevented.

The finding was not subject to traditional enforcement since the incident did not have a significant or potentially significant safety consequence, did not impact the NRC's ability to perform its regulatory function, and was not willful.

In accordance with IMC 0612, the inspectors determined that the finding was more than minor because it impacted the program and process attribute of the Public Radiation Safety Cornerstone and adversely affected the associated cornerstone objective of ensuring adequate protection of public health and safety from exposure to radiation. Specifically, contaminated material with measured dose rates distinguishable from background escaped the licensee's control outside the RCA and resulted in unnecessary radiation exposure to a member of the public that was approximately one percent of the public dose limit. The finding was assessed using the Public Radiation Safety-Significance Determination Process and determined to be of very low safety significance because: (1) it involved a radioactive material control problem that was contrary to NRC requirements and the licensee's procedure; and (2) the dose impact to a member of the public (the contract worker) was less than 5 mrem total EDE.

The licensee conducted a "why staircase" analysis as part of its ACE that focused on why contaminated equipment was transferred/stored in outdoor areas (a contributor to

the problem) instead of why material control was compromised in this instance (the fundamental cause). Given that the licensee elected to transfer equipment outdoors during potentially unfavorable environmental conditions without adequate controls in place, the cause of the radioactive material control problem was determined to involve a cross-cutting component in the human performance area for inadequate work control. Specifically, the licensee did not plan/coordinate work activities consistent with safety in that job site conditions, including environmental conditions (high winds, night time work, etc.), impacted human performance and consequently radiological safety during movement of contaminated material and equipment (H.3.(a)).

Enforcement: Title 10 CFR 20.1101(b) requires that each licensee use to the extent practical procedures based on sound radiation protection principles to achieve occupational and public doses as low as is reasonably achievable. Licensee procedure NP 4.2.25, Revision 14, "Release of Material, Equipment and Personal Items From Radiologically Controlled Areas," implements 10 CFR 20.1101(b) and was established to ensure that licensed material is controlled and that dose to the public is minimized. Sections 2.1, 2.4, and 4.1 of the procedure require that radioactive material remain in RCAs, and that contaminated items be monitored by qualified radiation protection personnel to determine they are free from detectable radioactive contamination prior to release. Contrary to these requirements, on May 21, 2009, radioactively contaminated debris escaped the licensee's control, migrated outside the RCA, and was picked-up by an individual resulting in unnecessary radiation exposure. Since the failure to control radioactive material was of very low safety significance, corrective actions were proposed as described above, and the issue was entered into the licensee's CAP as AR 01150045, the violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000266/2009005-06; 05000301/2009005-06).

.2 (Closed) NRC TI 2515/175, "Emergency Response Organization, Drill/Exercise Performance Indicator, Program Review"

The inspectors performed TI 2515/175, ensured the completeness of the TI's Attachment 1, and then forwarded the data to NRC Headquarters.

.3 (Open) NRC TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01)"

a. Inspection Scope and Documentation

On October 27, 2009, the inspectors conducted a walkdown of normally inaccessible portion of piping of the RHR system in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspectors' independent walkdown (TI 2515/177, Section 04.02.c.3).

In addition, the inspectors verified that the licensee had isometric drawings that described the RHR system configurations. Specifically, the inspectors verified the following, related to the isometric drawings:

- high point vents were identified;
- high points that do not have vents were acceptably recognizable;

- other areas where gas can accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves, were acceptably described in the drawings or in referenced documentation;
- horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified;
- all pipes and fittings were clearly shown; and
- the drawings were up-to-date with respect to recent hardware changes and that any discrepancies between as-built configurations and the drawings were documented and entered into the CAP for resolution.

The inspectors noted that the isometric drawings were not accurate with respect to small bore piping (TI 2515/177, Section 04.02.a). Specifically, the inspectors found two vent valves and one small relief valve that were not shown in the isometric drawings. Subsequently, the inspectors were informed by the licensee that the drawings were developed to record dimensions and configurations necessary to perform pipe stress analyses and that the scope of that effort excluded piping with a diameter less than 2.5 inches. Although these specific examples did not present an adverse impact to plant safety at the time of the inspection, the inspectors questioned if the level of detail of the isometric drawings was appropriate with regard to the Gas Accumulation Management Program. The licensee captured the issue in its CAP as AR 01159839.

In addition, the inspectors verified that Piping and Instrumentation Diagrams (P&IDs) accurately described the subject systems, that they were up-to-date with respect to recent hardware changes, and any discrepancies between as-built configurations, the isometric drawings, and the P&IDs were documented and entered into the CAP for resolution (TI 2515/177, Section 04.02.b).

Documents reviewed are listed in the Attachment to this report.

This inspection effort counts towards the completion of TI 2515/177, which will be closed in a later IR.

b. Findings

No findings of significance were identified.

.4 Confirmatory Order EA-06-178 Actions (92702)

a. Inspection Scope

In a letter dated January 3, 2007, (ADAMS Accession Number ML063630336), the NRC issued a Confirmatory Order to the licensee as part of a settlement agreement through the NRC's Alternative Dispute Resolution (ADR) program. The NRC investigated an alleged violation of 10 CFR 50.7, "Employee Protection," to determine whether a senior reactor operator was the subject of retaliation for raising a nuclear safety concern in the licensee's CAP. This issue was resolved through the NRC's ADR program and was being tracked as Apparent Violation (AV) 05000266/2006013-05; 05000301/2006013-05 pending continuing NRC review and inspection of the licensee's completion of the items specified in the Confirmatory Order.

The Order had been issued to the Nuclear Management Company (NMC), the previous operator of the Point Beach plant.

From December 14 through 18, 2009, the inspectors utilized IP 92702, "Followup On Traditional Enforcement Actions Including Violations, Deviations, Confirmatory Action Letters, Confirmatory Orders, And Alternative Dispute Resolution Confirmatory Orders," to assess the licensee's completion of the items contained in the Order. The inspectors interviewed site personnel, observed training conducted in response to the Confirmatory Order, performed document reviews, and reviewed some of the applicable corrective actions the licensee had taken in response to the Confirmatory Order. An Office of Enforcement Specialist assisted the inspectors.

In addition, the inspectors also assessed the results of the licensee's independent assessment of the corrective actions taken in response to the licensee's 2004, 2006, and 2008 culture surveys. This independent assessment was requested by the NRC Region III Office in the March 4, 2009, Annual Assessment Letter.

The modifications to the facility license as a result of the Confirmatory Order included the following items, in part:

1. By no later than nine (9) months after the issuance of this Confirmatory Order, the Nuclear Management Company (NMC) agrees to review, revise, and communicate to NMC employees and managers its policy relating to the writing of CAP reports, and provide training to NMC employees and managers to clarify management's expectation regarding the use of the program with the goal to ensure employees are not discouraged, or otherwise retaliated or perceived to be retaliated against, for using the CAP.

2. By no later than June 30, 2007, NMC agrees to communicate its safety culture policy (including safety-conscious work environment (SCWE)) to NMC employees, providing employees with the opportunity to ask questions in a live forum.

3. By no later than nine (9) months after the issuance of this Confirmatory Order, NMC agrees to train its employees holding supervisory positions and higher who have not had formal training on SCWE principles within the previous two years of the Confirmatory Order. NMC agrees to use a qualified training instructor (internal or external) for such training. NMC shall review and enhance, if necessary, its refresher SCWE training consistent with NMC's refresher training program and provide such refresher training to its employees. New employees holding supervisory positions and higher shall be trained on SCWE principles within nine (9) months of their hire dates unless within the previous two years of their hire dates, they've had the same or equivalent SCWE training.

4. By no later than March 30, 2007, NMC shall develop action plans to address significant issues identified as needing management attention in the NMC 2004 and 2006 Comprehensive Cultural Assessments at the Point Beach Nuclear Plant (PBNP); to conduct focus group interviews with Priority 1 & 2 organizations to understand the cause of the survey results; and to review and, as appropriate, reflect nuclear industry best practices in its conduct of focus groups and action plans to address the issues at PBNP. As part of the development of the action plans, NMC shall also assess and address any legacy issues identified in prior safety

culture assessments (i.e., CAP report 0510074 and Synergy Safety Culture Assessment) that impact the safety culture at PBNP. The executive summary, analysis, and contemplated action plans shall also be submitted to the NRC.

5. By no later than December 31, 2008, NMC shall perform another survey at PBNP comparable to the 2004 and 2006 surveys to assess trends of the safety culture at the site and the overall effectiveness of corrective actions taken in response to prior year assessments (i.e., CAP report 0510074 and 2006 Synergy survey).

6. By no later than 3 months after the receipt of the next cultural survey results at PBNP, NMC shall submit the executive summary, analysis of the results, and the contemplated corrective actions to the NRC.

7. NMC shall continue to implement a process which ensures that adverse employment actions are in compliance with NRC employee protection regulations and principles of SCWE.

8. In the event of the transfer of the operating license of any NMC operated facility to another entity, the commitments shall survive for the NMC fleet generally and PBNP specifically.

b. Observations and Findings

The NRC performed the first inspection of the Confirmatory Order items in June 2007 and documented observations in IR 05000266/2007003; 05000301/2007003, Section 4OA2.3. Inspectors reviewed the licensee's completion of Order Items 1, 2, and 3 and identified several observations, which the licensee subsequently entered into the CAP as AR 01096862.

The second NRC inspection was performed in June 2008 and documented in IR 05000266/2008003; 05000301/2008003, Section 4OA5.2. Inspectors verified the licensee's corrective actions taken in response to the previous NRC observations, documented in AR 01096862; reviewed the SCWE refresher and new supervisor training program as required by Order Item 3; and reviewed the licensee's actions in response to Order Item 4. No issues were identified with the actions taken for Order Items 1 and 2, and those two items were considered complete. Two Green findings (NCV 05000266/2008003-11; 05000301/2008003-11 and FIN 05000266/2008003-12; 05000301/2008003-12) were identified by the inspectors for Order Items 3 and 4, those items were not considered complete.

In July 2007, the PBNP operating license was transferred from the NMC to Florida Power and Light (FPL) Energy Point Beach, LLC. In April 2009, FPL Energy Point Beach, LLC changed its name to NextEra Energy Point Beach, LLC. Therefore, NextEra Energy Point Beach, LLC assumed responsibility for compliance with the Order.

The status of the remaining open Order items is summarized below. Note that an item status of complete refers to the status of the NRC review and inspection. Order Items 3, 7, and 8 contain ongoing actions that require continued implementation by the licensee.

(Complete) Order Item 3: The licensee continued implementation of Order Item 3, which required, in part, that the licensee provide SCWE training to its employees holding

supervisory positions and higher. The inspectors reviewed AR 01129565, initiated for NCV 05000266/2008003-11; 05000301/2008003-11, issued in 2008 when the NRC inspection identified four individuals who did not meet the SCWE training requirement. The four individuals who had exceeded the nine month requirement specified in the Order were subsequently trained by the licensee. In the current inspection, no additional supervisors were identified that missed the required training. The inspectors attended SCWE training for supervisors and found the 2009 training satisfactory. The inspectors reviewed the licensee procedures and the Learning Management System and determined they were satisfactory to track personnel for the required SCWE training, although the licensee recently identified several issues that required additional corrective actions. The inspectors determined that these issues, while not performance deficiencies, demonstrated that continued emphasis by the licensee was warranted to preclude future performance issues. Some additional oversight was provided by the plant training advisory board where, at the monthly meetings, individual supervisors who required SCWE training were tracked.

(Complete) Order Item 4: The licensee has completed Order Item 4 concerning actions resulting from the NMC 2004 and 2006 Comprehensive Cultural Assessments. On March 29, 2007, the licensee submitted to the NRC an analysis of the 2006 culture survey and the contemplated action plans (ML070890434). The inspectors verified that the licensee conducted the focus group interviews with Priority 1 and 2 organizations to understand the cause of the survey results, and that nuclear industry best practices were reflected in the conduct of focus groups and action plans to address the issues at Point Beach.

The inspectors reviewed the actions and status of the four "quick hitter" plans that were identified as not complete in the 2008 NRC inspection and the basis for Finding 05000266/2008003-12; 05000301/2008003-12. The licensee addressed this deficiency in AR 01129659 and the inspectors verified these "quick hitter" plans were complete.

The inspectors sampled several of the long-term actions plans and verified the licensee completed those individual actions. However, the inspectors noted that the results of the 2008 safety culture survey (Order Item 5) revealed the overall composite site nuclear safety culture rating remained low and the ratings from 2004 to 2008 showed minimal improvement. Based on the NRC findings issued in 2008 and the results of the 2008 safety culture survey, the inspectors were concerned there was a lack of management attention and priority to the action plans prior to the 2008 survey and that licensee management did not recognize many of the actions taken were either not effective or could not sustain improvements, especially in the departments which consistently had the lowest survey result scores in the 2004, 2006 and 2008 surveys. Licensee actions taken in response to the 2008 safety culture survey are discussed in the summary for Order Item 5.

(Complete) Order Item 5: The licensee has completed Order Item 5, to perform another survey at PBNP comparable to the 2004 and 2006 surveys. In June 2008, the licensee's contractor conducted a survey at Point Beach and submitted the results of the survey to the NRC on December 22, 2008, (ML083660387). As previously noted in the Order Item 4 discussion, the survey results did not show a marked improvement from the 2004/2006 surveys, and Point Beach continued to have an overall low nuclear safety culture rating.

As a result of the 2008 survey, and because the licensee had exceeded three assessment periods with a substantive cross-cutting issue in problem identification and resolution, the licensee was requested by the NRC in the March 4, 2009, Annual Assessment Letter to perform an independent assessment of the corrective actions taken in response to the 2004, 2006, and 2008 culture surveys. The independent assessment was performed from June 23 through June 25, 2009. The inspectors determined that the assessment team, which consisted of four individuals, was independent from the plant staff, with two members from FPL corporate, one member from another utility company, and one member from a consultant company. The inspectors noted that the assessment included personnel interviews, meeting attendance, and document reviews. The licensee's assessment concluded overall that the corrective actions taken for the 2008 survey results were more effective than those taken for the 2004 and 2006 culture surveys, and provided assurance that the progress could be sustained. However, the inspectors noted that the report did not include any detailed analysis or quantitative data as the basis for the assessment's conclusions; therefore, the inspectors could not evaluate the assessment team's conclusions. The licensee's assessment contained six observations and recommendations for improvements which were:

- an over-reliance on senior management's actions to establish expectations and demonstrate desired safety culture behaviors; therefore, the team recommended those behaviors be driven down to the department managers and line organization;
- while there is a high level of confidence in the CAP among licensee staff when dealing with safety-related, industrial safety, or plant reliability issues, the same confidence level does not exist with lower level issues, especially those which are closed to trend; therefore, the team recommended supplemental trending measures needed to be developed prior to the establishment of a fleet-trending program;
- while the managers interviewed understood safety culture, those same managers could not clearly articulate a consistent picture of an excellent nuclear safety culture; therefore, the team recommended that additional actions be taken to ensure the management team could clearly articulate the description of an excellent nuclear safety culture;
- the safety culture effectiveness assessments were currently compliance-focused with regard to the completion of corrective actions taken in response to the culture surveys; therefore, the team recommended an effectiveness assessment be performed to reevaluate the expectations provided in September 2008 and to promote the day-to-day implementation of the core nuclear safety culture values;
- the organization had difficulty separating day-to-day work place issues from nuclear safety culture issues; therefore, the team recommended addressing day-to-day work place issues in a different forum; and
- one of the major focus areas from the 2008 culture survey was achieving a better balance between workload and available resources, with the extended power uprate project adding additional workload to the plant; therefore, the team recommended the extended power uprate project should look for more effective means of implementation, to avoid unnecessary disruptions of the normal plant work schedule.

The independent assessment recommendations were entered into the CAP system as AR 01152228.

The inspectors also reviewed a sample of the corrective actions taken for the weaknesses identified in the 2008 safety culture survey and interviewed personnel in the groups having the lowest ratings in the survey. Many of the licensee personnel interviewed in December 2009 were interviewed during the 2007 and/or 2008 NRC inspections. The inspectors observed that many of the actions were recently completed and some groups made significant improvement, while other groups have shown marginal improvement, if any. However, the inspectors noted that the Point Beach Nuclear Safety Culture Improvement Team (NSCIT) developed and issued SCWE performance indicators for all work groups and that those indicators reflected that some groups remained as outliers (needed improvement). Those indicators aligned with the NRC observations from day-to-day resident inspections and interviews conducted with licensee personnel during this inspection.

In addition, the inspectors reviewed the results of other surveys performed on aspects of safety culture by FPL in late 2008 and one performed by an independent organization made up of external utility representatives in early 2009. While the inspectors concluded that those surveys were not comparable to the licensee's safety culture surveys previously discussed, the inspectors noted that both surveys contained positive results related to the nuclear safety culture and safety conscious work environment at Point Beach, indicative of some improvement since the 2008 safety culture survey.

Therefore, the inspectors concluded that the safety culture environment has shown some improvement and further monitoring by the plant NSCIT and continuing actions from the safety culture surveys and independent assessment team recommendations would be needed to continue this trend.

(Complete) Order Item 6: The licensee completed Order Item 6 when the licensee submitted the 2008 Safety Culture Survey executive summary, analysis of the results, and the contemplated corrective actions to the NRC on December 22, 2008, (ML083660387). The inspectors verified these submittals were complete within the timeframe contained in the Order.

(Complete) Order Item 7: The licensee continued implementation of Order Item 7 to implement a process that ensured adverse employment actions were in compliance with NRC employee protection regulations and principles of SCWE. The FPL Nuclear Division Policy, NP-413, was put in effect on May 15, 2008, and replaced the NMC procedure CP-0087. However, the inspectors observed that the FPL procedure was not as detailed as the original NMC procedure, and a follow-up inspection would be needed to look at specific adverse action cases. The licensee captured the inspectors' observations in condition report AR 01163410.

In a follow-up inspection, the inspectors reviewed a sample of adverse actions taken at PBNP since policy NP-413 was implemented to ensure the Order requirements were maintained. The inspectors also reviewed a new FPL Policy, "HR-AA-01, Involuntary Termination or Other Significant Employment Actions Affecting Nuclear Division Employees," issued as a result of the inspectors' previous observations. This new policy contained the employee protection criteria that were missing from the previous policy. During review of a sample of 10 adverse actions, the inspectors identified that in one

case the licensee had not completed an independent review of the personnel action by the Human Resources Department as required by the policy. The licensee entered this performance deficiency into the CAP as AR 01165164, performed the independent review, and determined there were no employee protection issues involved. The inspectors agreed with this determination and concluded the failure to implement the FPL Policy was considered a minor violation, in accordance with the NRC's Enforcement Policy.

(Complete) Order Item 8: For Order Item 8, the inspectors verified that after the transfer of the operating license of PBNP from NMC to NextEra Energy (formerly FPL), PBNP continued to follow the Order commitments.

No findings of significance were identified during this inspection.

Based on the results of this inspection and the actions documented in IRs 05000266/2007003; 05000301/2007003 and 05000266/2008003; 05000301/2008003, the inspectors concluded that the licensee had implemented all the actions required by the Confirmatory Order (EA-06-178). Therefore, the inspectors considered the associated Apparent Violation 05000266/2006013-05; 05000301/2006013-05, "Confirmatory Order EA-06-178," closed.

.5 Plant Modifications in Support of Extended Power Uprate (EPU) (71004)

a. Inspection Scope

From November 30 through December 18, 2009, the inspectors reviewed the following completed plant modifications during a baseline inspection for Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications. The following two modifications were completed for the Extended Power Uprate project, hence may be also be credited as samples towards completion of IP 71004, "Power Uprate." Additional details of these samples are included in IR 05000266/2009007; 05000301/2009007.

- Mechanical tie-ins to the SW and AFW systems for the new Unit 2 motor-driven AFW pump. Specifically, the inspectors reviewed a sample of the associated engineering change documentation, including the 10 CFR 50.59 screening, design calculations, work orders, engineering change requests, and corrective action documents, to assure the installed plant change was consistent with the design and licensing bases. The inspectors walked down the mechanical tie-ins to the SW and feedwater systems to verify the installed piping configurations were consistent with the design and installation documentation.
- Electrical and instrumentation tie-ins installed during the refueling outage for the new Unit 2 motor-driven AFW pump per EC-13401. The inspectors walked down changes to the Unit 2 control room panels with the SQUG engineer.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 5, 2010, the inspectors presented the inspection results to Mr. C. Trezise, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The Occupational Radiation Safety access control to radiologically significant areas and ALARA program inspection results to Mr. L. Meyer and other members of the licensee staff on October 30, 2009. This included closure of URI 05000266/2009004-01; 05000301/2009004-01 documented in Section 4OA5.
- TI 2515/177 inspection results to Mr. L. Meyer and other members of the licensee staff on October 30, 2009. The licensee acknowledged the issues presented.
- The ISI inspection results to Mr. L. Meyer and other members of the licensee staff on November 6, 2009. The licensee acknowledged the issues presented.
- The Verification of the Public Radiation Safety Performance Indicator inspection results with Mr. J. Pierce on December 4, 2009.
- The results of the Emergency Preparedness program inspection with Mr. C. Trezise on December 11, 2009.
- The licensed operator requalification training program inspection results with the Training Operations Supervisor, Mr. R. Amundson, on December 15, 2009.
- The annual review of Emergency Action Level and Emergency Plan changes with the licensee's Emergency Preparedness Manager, Mr. R. Johnson, via telephone on December 15, 2009.
- The Confirmatory Order (EA-06-178) inspection results to Mr. L. Meyer and other members of the licensee staff on December 18, 2009. The licensee acknowledged the conclusions and observations presented.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Aerts, Accounting Manager (NSCIT Leader)
B. Castiglia, Performance Improvement Manager
J. Costedio, Nuclear Licensing Manager/Regulatory Affairs Manager
R. Farrell, Radiation Protection Manager
R. Freeman, Emergency Preparedness Manager
R. Harrsch, Operations Site Director
L. Hawkeye, Engineering PI Manager
C. Hill, Work Control Center Manager
P. Holzman, GL 89-13 Program Engineer
L. Meyer, Site Vice-President
J. Schroeder, SW System Engineer
C. Trezise, Engineering Director/Acting Site Vice-President
T. Vehec, Plant Manager
G. Vickery, Work Management Manager

Nuclear Regulatory Commission

M. Kunowski, Chief, Division of Reactor Projects, Branch 5
J. Poole, Point Beach Project Manager, Office of Nuclear Reactor Regulations

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000266/2009005-01; 05000301/2009005-01	FIN	Failure to Meet GL 89-13 Program for Mussel Control (Section 1R12.1)
05000301/2009005-02	NCV	Failure to Ensure Adequate Control of Foreign Material in Safety-Related Systems (Section 1R15.1)
05000301/2009005-03	NCV	Failure to Update Safe Load Path Manual to Include Safety-Related Cable Locations (Section 1R18.1)
05000266/2009005-04; 05000301/2009005-04	URI	Potential Failure to Adequately Evaluate Seismic II/I Concerns for Units 1 and 2 'B' Containment Sump Strainers (Section 1R18.2)
05000301/2009005-05	NCV	Momentary Loss of Unit 2 Reactor Vessel Level Indication in the Control Room (Section 1R20.1)
05000266/2009005-06; 05000301/2009005-06	NCV	Failure to Maintain Proper Control of Radioactive Material Within the Radiologically Controlled Area (Section 4OA5.1)

Closed

05000266/2009005-01; 05000301/2009005-01	FIN	Failure to Meet GL 89-13 Program Requirement for Mussel Control (Section 1R12.1)
05000301/2009005-02	NCV	Failure to Ensure Adequate Control of Foreign Material in Safety-Related Systems (Section 1R15.1)
05000301/2009005-03	NCV	Failure to Update Safe Load Path Manual to Include Safety-Related Cable Locations (Section 1R18.1)
05000301/2009005-05	NCV	Momentary Loss of Unit 2 Reactor Vessel Level Indication in the Control Room (Section 1R20.1)
05000266/2009004-01; 05000301/2009004-01	URI	Failure to Control Radioactive Material Within the Radiologically Controlled Area Resulting in Unnecessary Dose to Worker (Section 4OA5.1)
05000266/2009005-06; 05000301/2009005-06	NCV	Failure to Maintain Proper Control of Radioactive Material Within the Radiologically Controlled Area (Section 4OA5.1)
05000266/2006013-05; 05000301/2006013-05	AV	Confirmatory Order EA-06-178 (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- AR 00509533; O&MR 379; Revision 1 Freezing Of Instrumentation Piping
- AR 00509586; Reduced Pump Seal Life Because Of Improper Venting
- AR 01075828; PAB HV Steam Exhaust Stack Drain Line Is Frozen
- AR 01140416; Ice In Sealtite For Many Security Components
- AR 01140633; Beach Drains Frozen
- AR 01141214; Ice On Floor In Unit 1 Facade
- AR 01141395; EC 12789 Facade Freeze Upgrade DRB Action items
- AR 01141687; Verify Cold Weather Preps Remains In A Working Stat
- AR 01142302; U2 Facade Sump Piping Heat Trace Alarms
- AR 01142711; Inadequate 2X04 Cable Drip Tray Causes Ice Buildup In Facade
- AR 01142806; Changes To OI 106 To Incorporate EC 12789 Facade Freeze Mod
- AR 01143674; Faulted MUX Causing MET Tower Data To Be Frozen
- AR 01143775; Frozen Drain Line In Unit 2 Facade
- AR 01146740; Cold Weather Checks UNSAT – Heat Lamp GFI Tripped
- AR 01148041; Heat Trace Drawing Needs Updating
- AR 01148221; Facade Heat Trace Panel reliability Unsatisfactory
- AR 01148314; Heat Trace Drawing Needs Updating/More Information
- AR 00149677; Massive Formation Of Ice Has Collected On Cable Tray In Southwest Corner Of Unit
- AR 01154068; Heat Trace For RS-SA-003 Installed Incorrectly
- AR 01155627; PC 49.5 Cold Weather Checklist, WH4 Heaters Broke
- AR 01155718; Heat Trace Not Installed Per Manufacturer Recommendations
- AR 01155829; Cold Weather Preps
- AR 01156747; Cold Weather Preps May Not Get Completed As Scheduled
- AR 01156940; Facade Freeze Tent
- AR 01156958; HV Piping Leak Downstream Of HV-990
- AR 01157478; CWPH MOD EC 11174 Requires Cold Weather Procedure Update
- AR 01158201; Cold Weather Issue – Primary And Backup Circuit In Alarm
- AR 01158202; Cold Weather Issue – Primary And Backup Circuit In Alarm
- AR 01158203; Cold Weather Issue – Vent To Atmosphere For RWST
- AR 01158938; Cold Weather Readiness System Engineering Reviews
- AR 01159535; BDE May Need Cold Weather Shutdown
- AR 01154813; Facade Freeze Protection Work Not Ready
- AR 01154683; Section Of Facade Heat Tracing Is Missing
- AR 01155829; Cold Weather Preps
- ICI 32; Facade Freeze Control Panel Settings; Revision 1
- IE Bulletin 79-24; Frozen Lines; September 27, 1979
- ISA-S67.02; Nuclear Safety-Related Instrument Sensing Line Piping And Tubing Standards For Use In Nuclear Power Plants
- OP-AA-102-1002; Seasonal Readiness; Revision 0
- OM 3.39; Degraded Equipment / Adverse Condition Monitoring Procedure; Revision 2

- 0-SOP HT-1B01; Unit 1 Non-Vital Train A Heat Trace Panels; Revision 0
- OM 3.9; Watchstation Status Checks And Watchstander Turnover Guides; Revision 15
- OI 106; Facade Freeze Protection; Revision 26
- OP-AA-102-1002; Seasonal Readiness; Revision 0
- PC 49; Cold Weather Preparations; Revision 7
- PC 49 Part 5; Cold Weather Checklist: Outside Areas And Miscellaneous; Revision 25
- WO #353856-07; Install 2FF-07-02C Heat Trace Cable On RS-SA-003 And Test
- WO #366472; 2VNTB-04802A Damper Not Fully Closing
- Drawing 019193; Electrical Layout Facade Area E-142; Unit 1
- Drawing 55805; Wiring Diagram Heat Tracing Panel "AH"; Auxiliary Building; Units 1 And 2
- Drawing 325073; Facade Freeze Protection Control Panel 1FFCP-02B; Secondary Distribution Breaker Panelboard 1FFDP-02-5; Unit 1
- Generic Letter 88-20; Supplement 5; Individual Plant Examination Of External Events For Severe Accident Vulnerabilities

1R04 Equipment Alignment

- 2-SOP-RH-002; Residual Heat Removal System Operation; Revision 3
- CL 7A; Safety Injection System Checklist Unit 2; Revision 30
- CL 7B; Safety Injection System Checklist Unit 2; Revision 27
- IT 04F; 2P-10A LHSI Pump Profile Test Mode 6 High Cavity Water Level Unit 2; Revision 4
- O-TS-EP-001; Weekly Power Availability Verification; Revision 11
- OP 7A; Placing Residual Heat Removal System In Operation; Revision 45
- Drawing ISI-2122; Residual Heat Removal Suction From Loop "A"; Unit 2
- Drawing ISI-2123; Residual Heat Removal Suction Header; Unit 2
- Drawing ISI-2125; Residual Heat Removal To Loop "B"; Unit 2
- Drawing ISI-2204; Residual Heat Removal Heat Exchangers HX-11A And HX-11B; Unit 2
- Drawing ISI-2228; Residual Heat Removal Pump Discharge; Unit 2
- Drawing ISI-2231; Residual Heat Removal Heat Exchanger Bypass; Unit 2
- Drawing ISI-PRI-2131; Residual Heat Removal To RPV; Unit 2
- Valve And Component Map; Pipeway 3; EL 8" PAB; Revision 0
- Valve And Component Map; Unit 2 RHR Heat Exchanger Cubicle; EL 5' PAB; Revision 4
- Valve And Component Map; Pipeway 3; Hallway Outside; EL 8' PAB; Revision 2
- Valve And Component Map; U2C – 46'; Revision 2
- Valve And Component Map; Unit 2 Containment; 10' And 21' Elevation; Revision 2
- Valve And Component Map; U2 A S/G Handhole Level; Area 2C-8; Revision 2
- Valve And Component Map; Unit 2 Containment "A" 10' Platform; Revision 3

1R05 Fire Protection

- FEP 4.0; Fire Emergency Plan; Revision 5
- FEP 4.20; Site; Revision 7
- FEP 4.26; North Service Building; Revision 3
- FHAR FZ245; Fire Area A01-E; Electrical Equipment Room – Unit 1; Fire Zone Data
- FHAR FZ775; Fire Area A71; G-04 Diesel Room; Fire Zone Data
- FOP 1.1; Brigade Training; Revision 9
- NP 1.9.14; Fire Protection Organization; Revision 10
- PC 74; Conducting And Evaluating Fire Drills; Revision 10
- Drawing 290590; Fire Protection For Turbine Building, Auxiliary Building And Containment; Elevation 44' – 0"
- Shift Staffing Report; Station Log; Mid-Shift; December 10, 2009

1R06 Flood Protection Measures

- AOP-9A; Service Water System Malfunction; Revision 24
- FSAR Appendix A.7; Plant Internal Flooding
- NP 8.4.17; PBNP Flooding Barrier Control; Revision 10

1R08 Inservice Inspection Activities

- AR 01142750; U2R30 Inservice Inspection
- AR 01160164; Delays In RPV Examinations
- AR 01153595; EPRI Issued Document For Dissimilar Metal Weld UT Exams
- AR 01144460; WCAP-15666-A (Reactor Coolant Pump Flywheel Examinations)
- AR 01125657; OE26445 – Nondestructive Examination Results Affect Core
- NDE-163; Manual Ultrasonic Examination Of Ferritic Pressure Vessel Welds Greater Than 2 Inches In Thickness; Revision 14
- NDE-109; Manual Ultrasonic Examination Using Longitudinal-Wave Straight-Beam Techniques; Revision 8
- NDE-171; Manual Ultrasonic Examination Of Nozzle Inside Radius Sections; Revision 13
- NDE-451; Visible Dye Penetrant Examination Temperature Applications 45°F TO 125°F; Revision 25
- NDE-753; Visual Examination (VT-2) Leakage Detection Of Nuclear Power Plant Components; Revision 15
- NDE-757; Visual Examination For Leakage Of Pressure Vessel Penetrations; Revision 7
- AM 3-31; Alloy 600 Management Program; Revision 4
- Work Order Package 00352519; Replacement Of An ASME Section III, Class 1, RCS To P-10A/B Residual Heat Removal Pump Suction Header Drain Valve 2RH-D-9
- Work Order Package 00352831; Replacement Of An ASME Section III, Class 1, Excess Letdown Heat Exchanger 2HX-4 Outlet Drain Valve 2CV-D-11

1R11 Licensed Operator Requalification Program

- OP 1B; Reactor Startup; Revision 60
- OP 1C; Startup To Power Operation Unit 2; Revision 15
- Results Of Licensed Operator Annual Operating Tests; 2009

1R12 Maintenance Rule Implementation

- AM 3-4; Implementation Of The Maintenance Rule At PBNP; Revision 7
- AR 01157305; Delayed Inspection Raises GL 89-13 Program And CCW Questions
- AR 01159196; 2HX-015D Containment Fan Coolers Blocked With Mussels
- NAP-407; Equipment Reliability; Revision 5
- NP 7.7.4; Scope And Risk Significant Determination For The Maintenance Rule; Revision 17
- NP 7.7.5; Maintenance Rule Monitoring; Revision 21
- NP 7.7.7; Maintenance Rule Periodic Evaluation; Revision 4
- SEM 4.2; Component Maintenance Program Guideline; Revision 4
- Evaluation Of Exceeding The Assumed Value For Partially Blocked Flowpaths For The Inspection Of Containment Fan Cooler 2HX-015A (EC14792)
- Evaluation Of Exceeding The Assumed Value For Partially Blocked Flowpaths For The Inspection Of Containment Fan Cooler 2HX-015C And 2HX-015D (EC14793)
- Point Beach GT System Corrective Action Plan; Revision 0 and 1
- Point Beach SE-0401 Action Tracking Data; Gas Turbine AR/CAPs

- Point Beach Gas Turbine System Health Report – Third Quarter 2009
- Point Beach Third Quarter System Matrix; July 1 – September 30, 2009
- Point Beach Fourth Quarter System Matrix; October 1 – December 31, 2009
- Point Beach Smart System Status Report; Gas Turbine System; December 15, 2007
- Point Beach Smart System Status Report; Gas Turbine System; January 17, 2008
- Point Beach Smart System Status Report; Gas Turbine System; February 28, 2008
- Point Beach Smart System Status Report; Gas Turbine System; May 2, 2008
- Point Beach Smart System Status Report; Gas Turbine System; August 1, 2008
- Point Beach Smart System Status Report; Gas Turbine System; January 1, 2009
- Point Beach Smart System Status Report; Gas Turbine System; February 23, 2009
- Point Beach Nuclear Plant Maintenance Rule (a)(1) Action Plan Timeline Data
- Point Beach Nuclear Plant Maintenance Rule Unavailability Data Sheet; June 1, 2009 – November 1, 2009
- GL 89-13 Program Document; Revision 8
- Procedure AM 3-19; Biofouling Control Program; Revision 4
- Procedure OI 155; Chemical Treatment of Service Water for Mussels; Revision 27
- Calculation 2002-0008; CCW HX Plugging Limit; Revision 3
- AR 01158115; Unexpected TSAC Entry due to low accident cooler SW flow
- AR 01158344; 2HX-12D CC HX Found to be Approximately 66% blocked
- AR 01159196; 2HS-1015D Containment Fan Cooler Blocked with Mussels
- AR 01159293; Significant Number of Blocked Tubes on 2HX-15C CFC
- AR 01159787; HX-12C CCW HX Exceeds Allowed Blocked Tubes
- AR 01159890; Tubes Blocked in 2HX-12D
- AR 01160350; U2 "A" CFC Exceeded Plugging Limits per Calculation 2002-0003
- EC14794; Evaluation of the Effect of the Blocked Flowpaths Found during the Inspection of HX-12C and 2HX-12D; December 10, 2009
- HX-12C BIO/SILT Fouling Inspection Program Form, Inspection dated October 27, 2009
- HX-12D BIO/SILT Fouling Inspection Program Form, Inspection dated October 28, 2009
- HX-15C-6 BIO/SILT Fouling Inspection Program Form, Inspection dated October 22, 2009

1R13 Maintenance Risk Assessments and Emergent Work Control

- AR 01161450; Availability Of AFW Piping For Mode 5 Steam Generator Availability
- AR 01161450; Availability Of AFW Piping For Mode 5 Steam Generator Availability
- IN 95-35; Degraded Ability Of Steam Generators To Remove Decay Heat By Natural Circulation
- NP 10.3.6; Shutdown Safety Review And Safety Assessment; Revision 30
- Control Room Log Entries Report; November 15 – 17, 2009
- Drawing 25494-200-M0K-0000-06061; Weld Map For FE-4036 Assembly
- Drawing 342215; ISI Isometric Auxiliary Feedwater To Steam Generator "B"
- Drawing 342217; ISI Isometric Auxiliary Feedwater To Steam Generator "A"

1R15 Operability Evaluations

- AR 01147224; Spent Fuel Pool Cooling Pump Was Rendered Non-Functional
- AR 01148036; P-12B Spent Fuel Pool Pump's RV Not Indicative Of True Performance
- AR 01156117; EPU Spent Fuel Pool Cooling Calculation Issues
- AR 01159196; 2HX-015D Containment Fan Coolers Blocked With Mussels
- AR 01160033; Apparent SW Leak; Unit 2 CFC HX-015A1-A4 Coils
- AR 01161636; New AFW Line In Contact With SW Pipe
- AR 01160007; Evidence Of Leakage From HX15A1-4

- AR 01160262; 1HX-15C CFC Flow Out Of Limit Low Per TS-33
- AR 01160350; U2 "A" CFC Exceeded Plugging Limits Per Calculation 2002-0003
- AR 01160443; Found Washer Between Gasket And Flange Face On 2HX-15A3
- AR 01162022; Spent Fuel Pool Cooling System Incorrectly Classified
- AR 1159784; Spent Fuel Pool Pump Suction Isolation Valve Stem Contacting Adjacent Pipe Insulation
- AR 1160262; 1HX-15C CFC Flow Out Of Limit Low Per TS-33
- CL 5C; Spent Fuel Pool Cooling And Refueling Water Circulating Pump Normal Operation Valve Lineup
- DG-M09; Design Requirements For Piping Stress Analysis; Revision 2
- EN-AA-203-1001; Operability Determinations/Functionality Assessments; Revision 1
- NP 8.4.10; Exclusion Of Foreign Material From Plant Components And Systems; Revisions 7 and 24
- TS 33; Containment Accident Recirculation Fan-Cooler Units (Monthly); Unit 1; Revision 31
- Causal Evaluation; 2SI-897B Failed To Operate (AR 1158812, AR1158797/WO 379810); October 22, 2009
- Drawing 018993; Auxiliary Cooling System; Unit 1; Revision 44
- Drawing 018995; P&ID Service Water; Unit 1
- Point Beach Nuclear Plant A-46 Final Report; Introduction And Seismic Verification Methodology; Revision 1
- Point Beach Nuclear Plant A-46 Final Report; Appendix A; Seismic Design For Structures and Equipment

<u>1R18 Plant Modifications</u>

- 07 Calculation 2009-0022; Air Entrainment for Containment Sump Screens; 2009
- AR 01122278; Safe Load Paths For Turbine Building Crane
- AR 01145715; SLP 3 Revision 11 For Precautions Needed Over U2 Truck Bay
- CA 0112278; Safe Load Paths For Turbine Building Crane
- 10 CFR 50.59/72.48 Screening For CA 0112278; Safe Load Paths For Turbine Building Crane
- AR 01159514; 5B FWTR Heater Contacted And Damaged Component
- AR 01162492; ACE 01157505 Failed To Meet Minimum Requirements
- EC 11542; Unit 2 Main Generator Circuit Breaker Addition
- 10 CFR 50.59 Evaluation of EC 11542; Unit 2 Main generator Circuit Breaker Addition
- EC 12601; Additional Sump Strainer Modules – Unit 2; October 1, 2009
- EC 13601; GSI-191 RCP, S/G, and RCS Loops Piping Insulation Replacement – Unit 2; February 11, 2008
- EC 14790; Validation of SSCs above the Unit 1 Sump B Suction Strainers; November 15, 2009
- EN-AA-203-1001; Operability Determinations/Functionality Assessments; October 8, 2009
- FSAR Appendix A.3; Control Of Heavy Loads
- MDB 3.2.5 1B30; 480 V AC Motor Control Centers; Unit 1; Revision 2
- MDB 3.2.6 2B30; 480 V AC Motor Control Centers; Unit 2; Revision 1
- OI 35B; Electrical Equipment General Information; Revision 14
- PASS 002452; Electrical Raceways – Unit 2 Containment 8ft; November 4, 2009
- PBNP Engineering Planning And Management Cable Schedule Data; Train "A" Cables
- PBNP U2R30 Draft Schedule (Fall 2009); September 2, 2009
- PBNP U2R30 Production Schedule; 72-Hour Look Ahead; October 18, 2009
- SCR 2009-0127-01; GSI-191 RCP, S/G, and RCS Loops Piping Insulation Replacement – Unit 2; September 8, 2009
- SFS-PB2-GA-00; Sure-Flow Strainer Recirc Sump System Layout; February 18, 2009

- SFS-PB2-GA-01; Sure-Flow Strainer General Notes; March 3, 2009
- SLP 3; Turbine Building Main Crane; Revisions 11 And 12-Draft A
- Bechtel Power Corporation Correspondence; Interim Load Paths For Safety-Related Handling Devices; October 8, 1981
- Drawing 19739; Lighting Schedule Panel 7L; Revision 22
- Drawing 080034; P&ID Service Water; Unit 1; Revision 65
- Drawing 6704-E-151001; Diesel Generator Building Yard Area Grading Plan; Revision 4
- Drawing 6704-E-353403; Yard Area Diesel Generator Duct Bank Plan; Revision 5
- Drawing 82607-G1.0; Old FWH 5A And 5B Removal; Revision 1
- Drawing M-2007; Equipment Location – Plan; Ground Floor North; Revision 19
- Hatch Area Study Design; Truck Bay, Gantry Track, Door Position And Opening, A/B Train Duct Banks
- Hatch Area Study With FWHTR Design; Truck Bay, Gantry Track, Door Position And Opening, A/B Train Duct Banks Feedwater Heater With Plates
- Hatch Area Study With Plates Design; Truck Bay, Gantry Track, Door Position And Opening, A/B Train Duct Banks With Plates

1R19 Post-Maintenance Testing

- AR 01159648; 2P-010B, Residual Heat Removal Pump Oiler Level Consumption
- AR 01160385; Bechtel Identification Of Precursors To EPC Contract
- AR 01160661; Failed Radiographs On Welds For EC11683
- AR 01161009; Failure Investigation Process Established Due To Repetitive Failure During Radiographic Testing Of AFW Welds Associated With EC 133400
- AR 01161191; Bechtel Corrective Action Report Not Written As Required
- AR 01159839; Some Vent Valves Not Identified On Isometric Drawings (NRC-Identified)
- AR 01159862; Acceptance Criteria For Gas Voids May Be Incomplete (NRC-Identified)
- AR 01159937; Sump Strainer Ii/I Seismic Documentation Incomplete (NRC-Identified)
- AR 01163219; Lack Of Documentation To Support A Decision Of 2/1 Acceptability (NRC-Identified)
- AR 01160941; No Requirement To Document Seismic II/I Evaluations; (NRC-Identified)
- AR 01158870; Found Badly Burned Contacts On 2B52-429K For Compressor K-4B
- AR 01159029; G-02 Foreign Material
- AR 01159056; Found G-02 Emergency Diesel Generator Start Lockout Relay 2 Out Of Specification
- AR 01159161; 40 T Relay In G-02 Found Out Of Specification
- AR 01159187; Mis-Communication During Work Activity
- AR 01159410; Z-013 Main Hoist Has A Pinched Cable
- AR 01159721; Oil Addition To 2P-10B RHR Pump
- AR 01159843; Thermal Overloads Found Tripped On 2B52-329K
- AR 01159845; Minor Procedural Issues Encountered During G-02 PMT
- AR 01159960; 2P-010B Oiler Adjustment Mechanism Setup Improperly
- AR 01160179; 2P-10A RHR Pump Oiler May Be Incorrectly Installed
- AR 01160366; Low Flow Indication In OI 136A RHR "A" Train F & V
- AR 01160551; Inconsistent RHR Flow Limitations In Various Procedures
- AR 01160557; Discrepancies Found During NRC Observed IT-04A RHR Test
- AR 01160749; SLP-1 And -2 Conflict With OSHA Required Crane Checks
- AR 01161191; No Corrective Action Report Has Been Written To Document Trend Of Failed Welds
- AR 01161192; Contrary to Requirement A 3-Inch Elbow Between Welds 44Q And 44M On The Auxiliary Feed Water Project Was Cut Out Due To Being Deficient

- AR 01161222; Site Evaluation Of NRC Information Notice 2009-20
- AR 01161691; Main generator Rotor(s) Weight Exceeds TB Crane (Z-14) Capacity
- AR 01161694; New Generator Rotor Weight Exceeds TB Crane (Z-14) Capacity
- AR 01161706; ASME B30.2 Code Year For Wire Rope Inspections
- AR 01161946; ACE 1160527 Not Accepted In A timely Manner
- AR 01162048; Load Block Leveler And White Substance On Wire Rope On Z-015
- AR 01162940; Work Orders Not Yet Completed From RCE
- IT 04A; RHR Pump And Valve Tests In DHR Mode (Cold Shutdown); Unit 2; Revision 26
- PI-AA-100-1002; Guideline For Failure Investigation Process; Revision 0
- 2-SOP-RH-002; Residual Heat Removal System Operation; Revision 3
- TS 3.7.5; Auxiliary Feedwater
- TS 82; Emergency Diesel Generator G-02 Monthly; Revision 77
- WO 376979; Replace Wire Rope On the Polar Crane; Unit 2
- Drawing 25494-200-M0K-0000-06061; Weld Map For FE-4036 Assembly; Revision 4
- Drawing 25494-200-M0K-0000-06062; Weld Map For 2FE-04036 Spool; Revision 1
- Drawing 25494-200-M0K-0000-06063; Weld Map For 2FE-4037 Assembly; Revision 6
- Drawing 25494-200-M0K-0000-06064; Weld Map For 2FE-4037 Spool; Revision 1
- Master Weld Log – Job No. 25494; Weld Map For 2FE-4037 Spool
- Point Beach Daily Quality Summary; November 12, 2009
- Point Beach U2R30 Outage Schedule; Polar Crane Cable Repair Data; October 25-26, 2009
- Polar Crane 2Z-013 Estimated Wire Rope Stretch Data
- Trico Manufacturing Corp; Technical Information Sheet; Effects Of Aeration On Constant Level Oilers
- Trico Manufacturing Corp; Technical Information Sheet; Affects Of Air Movement On Opto-Matic Oilers
- Trico Manufacturing Corp; Technical Information Sheet; Glass, LS, Or SS Opto-Matic Oilers Instructions Before Installing
- Trico Manufacturing Corp; Technical Information Sheet; Opto-Matic Installation
- Trico Manufacturing Corp; Technical Information Sheet; Preventing Excessive Lubrication In Oil Sump Applications
- Weld Failure Casual Evaluation; Aux Feed/Containment Spray Weld Failures; November 14, 2009

<u>1R20 Refueling And Other Outage Activities</u>

- AOP-2B; Unit 2; Feedwater System Malfunction; Revision 15
- AR 01158914; Reactor Vessel Level Indication Wide Range Calculations On Hold
- AR 01160451; Add Transmitter Valving To I&C Pre-Outage Training
- AR 01161576; Unit 2 Reactor Heat Removal Components Will Exceed 125 Percent
- AR 01161998; Revise 535A To Better Document Full Stroke Manual Exercise Of 2RH-715C
- AR 01162196; Inservice Testing Program Acceptance Criteria
- AR 01162379; Unit 2, 2CC-738A Valve Did Not Go Full Shut
- ASTM Designation; A 193/A 193M-93a; Standard Specification For Alloy-Steel And Stainless Steel Bolting Materials For High-Temperature Service
- ASTM Designation; B 16/B 16M-00; Standard Specification For Free-Cutting Brass Rod, Bar And Shapes For Use In Screw Machines-EC 14895; 2RH-716A – Yoke Bushing Nut Bolt Installation
- AR 01159071; Unable To Complete 21CP 04.024 Due To Mode Change
- AR 01159076; Unexpected Unit 2 Reactor Vessel High Alarm
- AR 01161058; PMT for RC-537 Not Performed According To Work Order Task
- AR 01161630; Cut Reinforcing Bar In AFW Pump Room Wall

- AR 01161966; P-31B Discharge Elbow Support Degraded
- AR 01161994; Testing Of SG Atmospherics Prior To Mode 4
- AR 01162014; Issue With SG Atmospheric Testing In OP-1A
- AR 01162073; Duct Tape On 2MS-02020 Yoke And Gland Follower
- AR 01162088; 2MS-2015 Atmospheric Dump Stroke Time Exceeded IST Limit
- AR 91162106; 2FD-2608 HX-22B MSR BTV Stuck In Mid Position
- AR 01162110; 2AF-4006 Closed Light Continuity Not As Required
- AR 01162119; Lone Wire Laying On Floor Below Apron Section of 2C03
- AR 01162139; MOB-276 Tripping
- AR 01162146; Valve Contractor Missing Step Sign Offs
- AR 01162166; 2C-03 Control Board Indication Deficiencies
- AR 01162202; Mode Change Hold Process Improvement Suggestions
- AR 01162223; U2 Purge Spool Pieces Restrict Access To Valves
- AR 01162253; BALCM – Dried Boric Acid Found On Packing Gland – 2SI-V-09
- AR 01162316; Additive Valve Position Out-Of-Tolerance For GV 4
- AR 01162353; Feed Pump Seal Inlet Valve Frozen/Doesn't Move
- AR 01162379; Unit 2 2CC-738A Valve Did Not Go Full Shut
- AR 01163155; Ground Water Drain Line Dripping On U1F 6.5" Floor
- AR 01163605; Wrong Valves For Tubing And Valve Replacement For K-2b
- AR 01153633; 2Z-104B Needs Replacement
- CL 1B; Containment Barrier Checklist; Unit 2; Revision 58
- CL 2B; Mode 6 To Mode 5 Checklist; Revision 11
- CL 2C; Mode 5 to Mode 4 Checklist; Revision 15
- CL 2E; Mode 3 To Mode 2 Checklist; Revision 16
- CL 20; Post Outage Containment Closeout Inspection; Revision 19
- CR 99-2241; Need To Evaluate Implementation Of The Service Water Model To Ensure Assumptions Are Valid
- EC 0014645; D-08 Battery Charger Temp Power From Alternate Source
- FP-E-MOD-02; Engineering Change Control; Revision 6
- FP-E-RTC-02; Equipment Classification – Q List; Revision 4
- IT 06; Containment Spray Pumps And Valves (Quarterly) Unit 2; Revision 61
- IT 45; Safety Injection Valves (Quarterly) Unit 2; Revision 51
- IT 45B; SI Valves (Shutdown) Unit 2; Revision 4
- IT 395; Safety Injection Valves (Annual) Unit 2; Revision 12
- NP 4.2.19; Entry requirements Into Various Radiologically Controlled Areas; Revision 16
- IWA-4000; Repair/Replacement Activities
- IWA-5000; System Pressure Tests
- IWB-5000; System Pressure Tests
- MR 97-102; RC Piping Overpressurization Relief – Unit 1; Final Design Description; October 22, 1997
- OI 53; Positioning Of The Fuel Transfer Cart; Revision 12
- OP 1A; Cold Shutdown To Hot Standby; Revision 99
- OP 1B; Reactor Startup; Revision 61
- OP 1C; Startup To Power Operation; Unit 2; Revision 16
- OP5A; Reactor Coolant Volume Control; Revision 42
- 10 CFR 50.99/72.48 Screening For MR 97-102; RC Piping Overpressurization Relief – Unit 1
- RESP 4.1; BOL Physics Tests; Revision 24
- TRHB 10.2; Primary Systems Descriptions: Reactor Coolant System; Revision 9
- WO 00378956; 2RH-716A Yoke Bushing Nut Bolt Installation
- 10 CFR 50.59/72.48 Screening of WO 00378956; 2RH-716A Yoke Bushing Nut Bolt Installation

- 2-PT-RCS-1; Reactor Coolant System Pressure Test – Inside/Outside Containment; Unit 2; Revision 3
- 21CP 04.023-1; Reactor Vessel Level Outage Calibration; Revision 7
- Calculation 2003-0057; Evaluation Of Service Water System Debris Transport To Auxiliary Feedwater
- Control Room Log Entries Data; October 19-20, 2009
- Drawing 018941; Fuel Transfer Arrangement System 2224; Revision 6
- Drawing 018977; Auxiliary Coolant System; Unit 2
- Drawing 152353; Auxiliary Cooling System; Residual Heat Exchanger; Discharge To Valve 720 To Loop B To Valve 742 To MOV 871 AC 601R-G; Unit 2
- Equipment Specification 677020; Fuel Transfer System; Revision 0
- NRC Generic Letter 88-17; Loss Of Decay Heat Removal 10 CFR 50.54(f); October 17, 1988
- Operations PCRA Backlog Scrub Data; December 23, 2009
- Point Beach AT-0246 Outage Action Request Mode Change Restraints Data; December 3, 2009
- Pro-Line Water Screen Services, Inc.; Installation Of Lower Boot Flapper Seal And Main Frame To Non-Metallic Basket Seals; September 12, 2001
- Rex Chainbelt Inc.; Conveyor And Process Equipment Division Service Manual; June 1965
- Unified Screw Threads Data; Table 3a - Coarse-Thread Series, UNC And UNRC – Basic Dimensions; Table 3b – Fine-Thread Series, UNF And UNRF – Basic Dimensions

1R22 Surveillance Testing

- AR 00151138; OSHA Required Crane Inspection Not Performed
- AR 01158712; Possible Discrepancies Noted During 2Z-13 Visual Inspection
- AR 01158730; 2Z-013 Visually Indeterminable Lateral Support Connections
- AR 01158949; 2Z-013 Polar Crane Inspection Weaknesses
- AR 01159254; 2Z-013 Polar Crane Inspection Weaknesses
- AR 01159410; Z-013 Main Hoist Has A Pinched Cable
- ANSI B30.2.0 – 1976; Overhead And Gantry Cranes (Top Running Bridge, Multiple Girder)
- ASME B30.2-2001; Overhead And Gantry Cranes (Top Running Bridge, Single Or Multiple Girder, Top Running Trolley Hoist)
- ASME B30.2-2005; Overhead And Gantry Cranes (Top Running Bridge, Single Or Multiple Girder, Top Running Trolley Hoist)
- ASME OM CODE-1995; Code For Operation And Maintenance Of Nuclear Power Plants
- AR 01158563; Unit 2 Containment Polar Crane Trolley Failure To Move
- AR 01158730; 2Z-013 - Visually Indeterminable Lateral Support Connection
- AR 01158746; Unit 2 Z-13 Crane #1 Controller Bridge Control Broken
- AR 01158788; 2RMP 9118-1 Emergent Issuance
- AR 01159790; Polar Crane Stopped Working
- AR 01159794; Potential Improvement To PBV-9240
- AR 01160749; SLP-1 And -2 Conflict With OSHA Required Crane Checks
- AR 01160844; Outdated Daily Crane Inspection Form Used
- AR 01162152; 12L-25 Lighting Panel Breaker Found Tripped
- AR 01162165; AR Not Initiated For Adverse Condition
- AR 01162167; DC Ground Found During ORT 3A
- AR 01162172; D-09 AC Input Breaker Tripped
- AR 01162173; Sliders Found Open During RF-445
- AR 01162177; G-01 Alarms Received During ORT 3A
- AR 01162205; Use Of CAPs Not Reinforced In ORT 3A
- AR 01162206; SA-51 Interstage Bleed On K-3B SA Compressor Does Not Work

- AR 01162212; Unexpected Alarm, 2C20A 2-2, D-01/D-03 DC Bus Under Voltage
- AR 01162222; Full Shut 2MS-5958 Indicates 12% Open Locally During ORT-54
- AR 01162638; 2DT-2081 Gasket Failure
- AR 01162668; 2P029T Oil Sample Contained Water
- AR 01162712; 2MS-2082 Trip Valve Leakage Observed During IT 09A
- AR 01162728; TS-81 G-01 EDG Testing While 2P-29 TDAFW Pump OOS
- AR 01162762; OBD Completion Did Not Reverse Changes To Procedure
- CMP 11.1; Component Maintenance Program; Revision 0
- FSAR Appendix A.3; Control Of Heavy Loads
- IT 09A; Cold Start Of Turbine-Driven Auxiliary Feed Pump And Valve Test (Quarterly) Unit 2; Revision 49
- ORT 3A; Safety Injection Actuation With Loss Of Engineered Safeguards AC (Train A)
- NRC Correspondence To Wisconsin Electric Power Company; February 1, 1982
- NUREG-0612; Control Of Heavy Loads At Nuclear Power Plants
- 2RMP 9118-1; Containment Building Crane OSHA Operability Inspections; Revision 5
- SLP 10; Load Weight Listings And Rigging Figures; Revision 22
- WO 359117; Wire Rope Inspection
- ALPS Wire Rope Corporation; Certificate Of Conformance; October 25, 2009
- Control Room Log Entries Data; TDAFW Test; December 4 – 11, 2009
- Drawing 275460; Auxiliary Feedwater System Units 1 and 2
- Point Beach Nuclear Plant Wire Rope Inspection Criteria Instructions
- Priority Work Schedule Data; September 10, 2009

<u>1EP2 Alert and Notification Evaluation</u>

- ENS Notification 45553; Notification Due To A Single Emergency Siren Actuation; December 9, 2009
- EPMP 6.0; Alert And Notification System; Revision 9
- FEMA Prompt Alert And Notification System Approval Letter And Design Report; December 7, 1987
- PBNP ANS Maintenance Records; October 2007 - November 2009
- AR 01162916; Power Outages Caused Sever Sirens Out-of-Service Due To Weather
- AR 01160553; Replaced Siren P-013 Antenna
- AR 01130759; Siren Test Postponed Due To Severe Weather

<u>1EP3 Emergency Response Organization Augmentation Testing</u>

- EP 5.0; Organizational Control Of Emergencies; Revision 52
- EPIP 1.1; ERO Notification; Revision 56
- EPG 1.0; Point Beach Nuclear Plant Shift Augmentation Drill Guideline; Revision 13
- EPMP 7.0; Emergency Response Organization Notification System; Revision 6
- PBN EP TP; Emergency Preparedness Training Program Description; Revision 8
- Emergency Response Organization Training Drill Team Roster; December 3, 2009
- LMS ERO Qualification Status Verification; December 10, 2009
- NPM 2008-0130; March 27, Quarterly ERO Augmentation Drills; May 2, 2008 – September 17, 2009
- AR 01162982; Augmentation Drills Taking Credit For 30-Minute Chemistry Technician With On-shift Chemistry Technician
- AR 01162977; Augmentation Drill Start Time Questioned During NRC Inspection
- AR 01162972; Loss Of Dialogics ERO Notification System Capabilities
- AR 01155763; EP ERO Expectations For Wearing A Pager

- AR 01153790; July 28, 2009 Drill Dose Assessment Challenge
- AR 01156706; September 17, 2009 Augmentation Drill Two Responders Greater Than 30 Minutes And One Responder Greater Than 60 Minutes
- AR 01151489; June 16, 2009 ERO Augmentation Drill Two Responders Greater Than 30 Minutes

1EP4 Emergency Action Level And Emergency Plan Changes

- EP 2.0; Emergency Plan Acronyms And Definitions; 41 and 42
- EP 6.0; Emergency Measures; 50, 51, and 52
- EPIP 1.2.1; Emergency Action Level Technical Basis; 3
- 10 CFR 50.54(q) Reviews For Emergency Plan And EAL Revisions

1EP5 Correction Of Emergency Preparedness Weaknesses And Deficiencies

- Focused Self-Assessment Report PBSA-EP-09-03; Point Beach Emergency Preparedness Pre-NRC Inspection; November 3, 2009
- Point Beach Toxic Gas Unusual Event July 3, 2008 Report; July 14, 2008
- Point Beach Security Unusual Event April 8, 2008 Report; May 7, 2008
- Point Beach Loss Of Off-Site Power Unusual Event January 15, 2008 Report; February 26, 2008
- PBNP 09-026; Emergency Preparedness Audit; August 12, 2009
- PBNP 08-026; Emergency Preparedness Assessment; August 12, 2008
- PBNP 08-011; Emergency Preparedness Assessment; May 3, 2008
- AR 01151074; EPlan Organization Chart Different Than Site Organization Chart
- AR 01149526; Radiation Protection Leader Position Drops Below Three Deep
- AR 01136999; Self-Assessment DEP Data Discrepancy
- AR 01131429; July 3, 2008 Evaluate Toxic Gas EAL
- AR 01131394; July 3, 2008 Unusual Event
- AR 01121253; Transfer Of Command And Control Confusion During January 15, 2008 Unusual Event
- AR 01120314; Unusual Event January 15, 2008 ENS Notification Made At 59 Minutes

2OS1 Access Control to Radiologically Significant Areas

- RWP 00000861, Fuel Motion And Sent Fuel Pool Activities; Revision 1
- HP 2.14; Containment Keyway Personnel Access; Revision 15
- HP 2.15.1; High Level Contamination And Discrete Radioactive Particle Control; Revision 5
- HP 2.17; Very High Radiation Area Personnel Access; Revision 7
- HP 2.6; Locked And Very High Radiation Area Key Control; Revision 32
- HP 3.2; Radiological Labeling, Posting And Barricading Requirements; Revision 50
- HP 3.2.10; Secure High Radiation Area Controls; Revision 1
- HP 3.6; Alpha Monitoring Program; Revision 0
- HPIP 1.64; Control of Underwater Diving In Radiologically Hazardous Areas; Revision 7
- HPIP 2.1.1; Response Checks Of Portable Survey Instruments; Revision 11
- HPIP 3.50; Radiation Surveys; Revision 13
- FP-RP-JPP-01; Radiation Protection Job Planning; Revision 6
- 0-SOP-FH-001; Fuel/Insert/Component Movement In the Spent Fuel Pool Or New Fuel Vault; Revision 15
- RP 1C, Refueling; Revision 65
- RP 2A; Receipt Of New Fuel Assemblies; Revision 47

- RP-18 Part 3; Place Loaded DSC/TC Back Into The Spent Fuel Pool; Revision 3
- RESP- 2.3; Defective Removable Top Nozzle Replacement; Revision 7
- HPCAL 1.1; Radiation Protection Instrument Calibration, Repair And Response Checks; Revision 22
- NP 4.2.19; Entry Requirements Into Various Radiologically Controlled Areas; Revision 16
- NP 4.2.32; Respiratory Protection Program; Revision 7
- AR SAR 01142742; Access Control To Radiologically Significant Areas And ALARA Planning And Controls
- AR SAR 0115197; Access Control To Radiologically Significant Areas And ALARA Planning And Controls

2OS2 As-Low-As-Is-Reasonably-Achievable Planning And Controls

- FP-WM-PLA-01; Work Order Planning Process; 5
- NP 4.2.1; ALARA Program; Revision 20
- FP-RP-JPP-01; RP Job Planning; Revision 6
- FP-RP-RWP-01; Radiation Work Permit; Revision 8
- Radiological Controls And Associated ALARA Files For Insulation; Work Orders 00371055, 00371056, And 00371057
- Radiological Controls And Associated ALARA Files For RCP Work; Work Orders 00356469, 00358775, And 00366298
- Radiological Controls And Associated ALARA Files For Core Barrel Move; Work Order 00365421

4OA1 Performance Indicator Verification

- 2-PT-AF-2; Turbine Driven Auxiliary Feedwater System And MS Supply Pressure Test Outside Containment – Unit 2
- AR 01135651; AF Mod Deferral Requires MSPI Basis Document Update
- AR 01138122; PRA Change For MSPI Not Explained In Submittal File
- AR 01138400; PRA Change For MSPI Not Explained In Submittal File
- AR 01142718; MSPI Margin Reduced Due To PRA Change
- EPG 1.1; Performance Indicator Guideline; Revision 6
- EPMP 6.0; Alert And Notification System Siren Function Data; October 2008 – September 2009
- FG-E-MSPI-01; Mitigating System Performance Index; Revision 3
- LI-AA-200-1000-10000; FPL Fleet Licensing Performance Indicators; Revision 00
- Mitigating Systems Performance Index (MSPI) Basis Document Data For Point Beach Nuclear Plant; Revisions 12 And 14
- MSPI Monthly Unavailability And Verification Data; July, August, And September, 2008
- MSPI Monthly Unavailability And Verification Data; October, November, And December, 2008
- MSPI Monthly Unavailability And Verification Data; January, February, And March, 2009
- MSPI Monthly Unavailability And Verification Data; April, May, And June, 2009
- NP 5.2.16; NRC Performance Indicators; Revision 14
- NRC Occupational Exposure Performance Indicator Data; October 2008 Through September 2009
- Alert and Notification System Performance Indicator Records; October 2008 – September 2009
- Atmospheric Effluent Radioisotopic Quantification Report; March 2009
- Atmospheric Effluent Radioisotopic Quantification Report; June 2009
- Atmospheric Effluent Radioisotopic Quantification Report; September 2009

- Drill And Exercise Performance PI Results; October 2008 - September 2009
- Drill And Exercise Performance Records; October 2008 - September 2009
- ERO Drill Participation Summaries; December 2008 - September 2009
- ERO Participation Monthly Reports; December 2008 - September 2009
- Emergency Preparedness Attendance Reports; December 2008 - September 2009
- Liquid Effluent Radioisotopic Quantification Report; March 2009
- Liquid Effluent Radioisotopic Quantification Report; June 2009
- Liquid Effluent Radioisotopic Quantification Report; September 2009
- Mitigating Systems Performance Index Derivation Report Units 1 And 2; Heat Removal System; Third Quarter of 2008 Through Second Quarter of 2009
- NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 5
- NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 6; October 2009
- NP 1.10.1; Record Keeping For NRC Licensed Operators; Revision 8
- NP 5.2.16; NRC Performance Indicators; Revision 14
- NP 5.2.17; Equipment Performance And Information Exchange (EPIX); Revision 2
- OI 62A; Motor-Driven Auxiliary Feedwater System (P-38A And P-38B)
- TRHB 11.4; Secondary Systems Descriptions: Auxiliary Feedwater System; Revision 10
- Control Room Log Entries; July 2008 through June 2009

4OA2 Identification and Resolution of Problems

- AR 01114734; Lack Of Progress On Cable Submergence Issue
- AR 01163603; Trend Coding Of CAPS
- AR 01138519; FM Found During Lower Core Plate Inspection
- AR 01157789; FME Barrier Found Inside FW Heater 4A During Inspection
- AR 01158516; Component Cooling Water Heat Exchanger FME Issues
- AR 01158573; Wrench Dropped Into Cavity
- AR 01159958; Foreign Material Found In Discharge Of 2CV-257
- AR 01160348; FM Debris Scan Challenged RV Lower Internal Install (2R30)
- AR 01160355; LUVS Screen Dropped In Refuel Cavity
- AR 01160443; Found Washer Between Gasket And Flange Face On 2HX-15A3
- AR 01160489; Foreign Material On Lower Core Plate
- AR 01160494; Trend – Submerged Electrical Cables
- AR 01160572; Resource Needs Were Not Identified To Support FM Inspection In RMP
- AR 01160820; U2R30 Cavity Foreign Material Controls
- AR 01160980; SFMEA Concerns At The Spent Fuel Pool
- AR 01161181; Untimely Reporting Of foreign Material
- AR 01161214; Z-756 Hoist Pendant Damage Causes Hoist Inoperability
- AR 01161216; FME Found While Inspecting Portion Of 2A02 Bus
- AR 01161285; Sump Bravo Needs Fabricated FME Covers When Elbows Are Removed
- AR 01161310; During 2ICP 02.019 Testing, We Found A Hair In PC-949B-XA
- AR 01161672; Bechtel Contamination Control For Valves And Pipes
- AR 01162133; Foreign Material Found In The New Output Breakers
- AR 01162169; FME Issue Of Bottle Dropped In Stabrex Tanker
- AR 01162213; No Housing Covers Installed On FD Valve Operators
- AR 01162509; Four Absorbent Bags Found In the Unit 2 Turbine Hall Sump
- CMP 12.0; Equipment Failure Trending; Revision 5
- FG-PA-CTC-01; CAP Trend Code Manual; Revision 11
- FG-PA-DRUM-01; Department Roll Up Meeting Manual – Department Performance Trending; Revision 8

- PBN-09-010; Point Beach Nuclear Assurance Report; System Engineering; May 26, 2009
- REI 48.0; Reactor Engineering Trending Program; Revision 2
- Point Beach Nuclear Plant AT-0384 Activity Trending Data; December 21, 2009
- Point Beach Nuclear Plant Drum Summary Report; First Quarter 2009
- Point Beach Nuclear Plant Drum Summary Report; Second Quarter 2009

4OA5 Other Activities

- AR 01165164; NP-413 Policy Requirement Not Implemented
- Policy HR-AA-01; Involuntary Termination Or Other Significant Employment Actions Affecting Nuclear Division Employees; Revision 0
- Policy SY-AA-02; Denial of Unescorted Access to FPL/FPLE Nuclear Facility; Revision 0
- FP&L NUC GET Plant Access Training 003; Revision Dated July 26, 2006
- HPIP 1.60; Calculating Shallow And Deep Dose Rates Due To Skin Contamination; Revision 11
- NP 1.7.3; Site Specific Requirements For Access To And Termination From Point Beach Nuclear Plant; Revision 18
- NP 4.2.25; Release Of Material, Equipment And Personal Items From The Radiologically Controlled Areas; Revision 14
- Apparent Cause Evaluation – AR 01150045; Loss Of Radioactive Material Control Inside Protected Area; Revision 1 and 2
- Chesapeake Nuclear Services Final Report, Dose Assessment For May 21, 2009 Contamination Event At The Point Beach Nuclear Plant; September 10, 2009
- Dispersed Contamination Dose Assessment Summary; July 2, 2009
- Personnel Contamination Event Report; May 21, 2009
- DRW 110E029, Sheet 1; Auxiliary Coolant System; September 10, 2008.
- DRW 110E035, Sheet 1; Safety Injection System; August 1, 2007
- DRW P-248; Residual Heat Removal System; December 25, 1999
- DRW P-237; SIS to Primary Coolant Cold Leg; January 22, 2004
- PO No. 00024065; Point Beach Walkdown Closure Report; November 16, 2009
- AR 01129366; PBNP Confirmatory Order Requirements Sustainability For Adverse Employment Actions
- AR 01129462; Schedule For Incumbent Mgrs/Supv For NLA Course
- AR 01129565; 4 Individuals Not Meeting SCWE Confirmatory Order
- AR 01129659; EA 06-178 Confirmatory Order Inspection Finding
- AR 01152228; Independent Assessment Of The Effectiveness Of Corrective Actions From Safety Culture Survey
- AR 01157190; Schedule PBN Personnel For SDA/LF Slots
- AR 01157534; Quick Hit Assessment PBSA-SRC-09-04
- AR 01162560; Security Supervisor Not Tracked For Required SCWE Training
- AR 01162564; 7 People Required To Attend SCWE Training And Not Being Tracked
- AR 01163410; Follow-up Issue SCWE Confirmatory Order Inspection
- FPL Nuclear Policy NP-413; Involuntary Termination of Division Employees; Revision 5
- NMC Policy CP 0087; Material Employment Action Review; Revision 0
- Corrective Action Effectiveness Review –AR01070153-12, April 29, 2009
- Memo from F. Flentje to J. Costedio; Verification of 2007 SCWE Confirmatory Order Actions Committed During September, 24, 2008 Public Meeting with NRC; February 14, 2009
- PARB Presentation for Non-Performance of EFR 1070334, Adverse Employment Action Policy, November 30, 2007
- Memo from B. Deuel to Nuclear Safety Culture Improvement Team; September 30, 2009 Nuclear Safety Culture Improvement Team Meeting Minutes; September 30, 2009

- Memo from B. Deuel to Nuclear Safety Culture Improvement Team; December 2, 2009 Nuclear Safety Culture Improvement Team Meeting Minutes; December 2, 2009
- Memo from L Meyer to File; August 2009 PBNP PTAB Meeting Minutes; September 12, 2009
- Memo from L Meyer to File; February 2009 PBNP PTAB Meeting Minutes; February 23, 2009
- Point Beach Supervisor Leadership Development Program; Training Program Description; Revision 6
- Point Beach Succession Plan; January 2010
- Point Beach Knowledge Retention Program; December 2009
- NRC 2007-0015, NMC Letter to NRC; NMC Plan to Address the Safety Culture Issues an at Point Beach Nuclear Plant; March 29, 2007 (ML070890434)
- NRC 2008-0078, FPL Energy Letter to NRC; Status of Action Plans Taken in Response to Confirmatory Order EA-06-178; November 11, 2008 (ML083170356)
- NRC 2008-0090, FPL Energy Letter to NRC; Confirmatory Order EA-06-178 Section IV.6 Nuclear Safety Culture Survey Results; December 22, 2008 (ML083660387)
- Point Beach Independent Assessment of Safety Culture Survey Corrective Action Effectiveness; June 28, 2009

LIST OF ACRONYMS USED

AC	Alternating Current
ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access Management System
ADR	Alternative Dispute Resolution
AFW	Auxiliary Feedwater
ALARA	As-Low-As-Is-Reasonably-Achievable
ANS	Alert and Notification System
AOV	Air Operated Valve
AR	Action Request
ASME	American Society of Mechanical Engineers
AV	Apparent Violation
BACC	Boric Acid Corrosion Control
CAP	Corrective Action Program
CCWHX	Component Cooling Water Hear Exchanger
CFC	Containment Fan Cooler
CFR	Code of Federal Regulations
EA	Enforcement Action
EC	Engineering Change
EDE	Effective Dose Equivalent
ELHX	Excess Letdown Heat Exchanger
EP	Emergency Preparedness
EPRI	Electric Power Research Institute
EPU	Extended Power Up-Rate
ERO	Emergency Response Organization
FPL	Florida Power and Light
FSAR	Final Safety Analysis Report
FW	Feedwater
GL	Generic Letter
GSI	Generic Safety Issue
I&C	Instrumentation and Control
IEL	Initiating Event Likelihood
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
LER	Licensee Event Report
LI	Level Indicator
LOCA	Loss of Coolant Accident
LOLC	Loss of Level Control
LOOP	Loss of Off-site Power
LT	Level Transmitter
mrem	Millirem
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NMC	Nuclear Management Company
NRC	U.S. Nuclear Regulatory Commission
NSCIT	Nuclear Safety Culture Improvement Team
OSHA	Occupational Health and Safety Administration

P&ID	Piping and Instrumentation Diagram
PARS	Publicly Available Records System
PBNP	Point Beach Nuclear Plant
PI	Performance Indicator
POS	Plant Operating State
PMT	Post-Maintenance Testing
PT	Pressure Test
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RFO	Refueling Outage
RHR	Residual Heat Removal
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
SCWE	Safety-Conscious Work Environment
SDP	Significance Determination Process
SG	Steam Generator
SI	Safety Injection
SLP	Safe Load Path
SQUG	Seismic Qualification Users Group
SRA	Senior Reactor Analyst
SW	Service Water
TI	Temporary Instruction
TS	Technical Specification
TSAC	Technical Specification Action Statement
TTB	Time-to-Boil
URI	Unresolved Item
VT	Visual Examination
WO	Work Order

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Sincerely,

/RA/

Michael Kunowski, Chief
Branch 5
Division of Reactor Projects

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Letter to L. Meyer from M. Kunowski dated February 10, 2010

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, NRC INTEGRATED INSPECTION REPORT 05000266/2009005; 05000301/2009005 AND STATUS OF CONFIRMATORY ORDER EA-06-178

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