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

November 9, 2010

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 05000266/2010004;
05000301/2010004

Dear Mr. Meyer:

On September 30, 2010, 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 results of this inspection, which were discussed on October 5, 2010, with you and members of your staff.

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

Based on the results of this inspection, three NRC-identified findings of very low safety significance were identified. Of these findings, two involved violations 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 the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of the findings, 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 cross-cutting aspect assigned to 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 at the Point Beach Nuclear Plant.

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

Sincerely,

/RA/ By John Jandovitz Acting For/

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

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

Enclosure: Inspection Report 05000266/2010004; 05000301/2010004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-266; 50-301
License Nos:	DPR-24; DPR-27
Report No:	05000266/2010004; 05000301/2010004
Licensee:	NextEra Energy Point Beach, LLC
Facility:	Point Beach Nuclear Plant, Units 1 and 2
Location:	Two Rivers, WI
Dates:	July 1, 2010, through September 30, 2010
Inspectors:	S. Burton, Senior Resident Inspector R. Ruiz, Resident Inspector J. Jandovitz, Project Engineer M. Phalen, Senior Health Physicist R. Langstaff, Senior Reactor Inspector P. Cardona-Morales, Reactor Engineer N. Féliz Adorno, Reactor Engineer W. Lyons, Senior Reactor Engineer D. Cunanan, Reactor Systems Engineer D. Woodyatt, Reactor Engineer
Approved by:	Michael A. Kunowski, Chief Branch 5 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000266/2010004, 05000301/2010004; 07/01/2010 – 09/30/2010; Point Beach Nuclear Plant, Units 1 & 2; Fire Protection and Surveillance Testing, and Other Activities.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified by the inspectors. Two findings were considered non-cited violations (NCVs) of NRC regulations. 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. A finding of very low safety significance and associated non-cited violations of a license condition was identified by the inspectors for the failure to identify hydrogen fire hazards on a pre-fire plan. Specifically, the licensee failed to identify that a compressed gas cylinder in the Unit 1 sample room contained hydrogen and that the Volume Control Tank valve galleries contained hydrogen piping. The licensee entered this issue into their corrective action program and revised the pre-fire plan to reflect the identified hydrogen fire hazards.

  The finding was determined to be more than minor because failure to identify hydrogen fire hazards in the pre-fire plan could impact the fire brigade’s ability to effectively fight a fire due to the unique hazards associated with hydrogen. The inspectors determined that the finding was of very low safety significance because the fire brigade consisted of plant operators familiar with the 46-foot elevation of the auxiliary building and associated hazards. This finding was associated with the Mitigating Systems Cornerstone attribute of Protection Against External Events (Fire) and affected the cornerstone objective of preventing undesirable consequences (i.e., core damage). No cross-cutting aspects associated with this finding were identified. (Section 1R05)

- Green. A finding of very low safety significance was identified by the inspectors for the failure to provide appropriate acceptance criteria for the fire door surveillance procedure. Specifically, the acceptance criteria for fire door functionality did not specify that doors, when opened, returned to the closed and latched position. The licensee entered this issue into their corrective action program and planned to revise the surveillance procedure.

  The finding was determined to be more than minor because, if left uncorrected, the failure to have appropriate acceptance criteria would become a more significant safety concern. Specifically, the lack of appropriate fire door functionality acceptance criteria could result in a nonfunctional door closing mechanism and a degraded fire barrier not being detected during surveillance activities. This finding was associated with the Mitigating Systems Cornerstone attribute of Protection Against External Events (Fire) and affected the cornerstone objective of preventing undesirable consequences (i.e., core damage). The inspectors determined that the finding was of very low safety significance because the inspectors did not identify any instances where a fire door was left open or unlatched, or an instance where a fire door which would not close on its own and was not monitored for closure. Consequently, the inspectors determined that the finding represented a low degradation and, as such, this finding screened as Green. This finding has a cross-cutting aspect in the area of human performance, work practices, because the licensee's failure to follow procedures, such as the procedure writers' guide, resulted in the failure to provide appropriate acceptance criteria for the fire door surveillance procedure [H.4(b)]. (Section 1R22.1)

- Green. A finding of very low safety significance and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure to ensure that the residual heat removal (RHR) system would be capable of responding to a loss of coolant accident that occurred in Mode 4. Specifically, the RHR system could experience flash evaporation during a loss of coolant accident in this Mode resulting in steam binding of the system pumps and/or an adverse waterhammer. The licensee entered this issue into the corrective action program and will make procedure changes to ensure the operability of at least one RHR train while in Mode 4.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low safety significance because a Phase II evaluation determined that it represented a change in core damage frequency of less than 5 E-9. The inspectors determined that this finding did not have a cross-cutting aspect because it was not obvious that the licensee should have identified the potential problem with RHR. (Section 4OA5.3)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at 100 percent power throughout the entire inspection period with the exception of small power reductions during routine surveillance testing; an unplanned downpower to 88 percent power on July 12, due to a loss of main generator hydrogen pressure; an unplanned outage from July 26 through July 28 as a result of a loss of condenser vacuum and reactor trip while at reduced power to repair a main generator hydrogen leak; and an unplanned downpower to 55 percent power due to a feedwater pump temperature issue.

Unit 2 operated at 100 percent power throughout the entire inspection period with the exception of an unplanned outage from July 9 through July 11 as a result of a feedwater regulating valve failure and subsequent manual reactor trip, and small power reductions during routine surveillance testing.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Unit 1 chemical and volume control system, charging pump train “A” after return-to-service;
- direct current distribution for Units 1 and 2 with battery chargers out-of-service for maintenance;
- instrument air compressor K2A with service air compressor K3A out-of-service; and
- Unit 2 turbine-driven auxiliary feedwater pump after return-to-service from service water valve test.

The inspectors selected this system based on their risk-significance relative to the Reactor Safety Cornerstones at the time they were 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, the Final Safety Analysis Report (FSAR), condition reports, and the impact of ongoing work activities on redundant trains of equipment to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked-down portions of the systems 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 any equipment alignment related issues 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 four partial system walkdown samples as defined in Inspection Procedure (IP) 71111.04-05.

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

During the week of July 12, 2010, the inspectors performed a complete system alignment inspection of the safety injection (SI) 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 line-ups, 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 work orders (WOs) was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

Also, additional activities were performed during this system walkdown that were associated with Temporary Instruction (TI) 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." These activities are described in Section 1R04.3.

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

b. Findings

No findings were identified.

.3 System Walkdown Associated with TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems"

a. Inspection Scope and Documentation

On July 15, 2010, the inspectors conducted a walkdown of the SI system in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). In addition, the inspectors verified that the licensee had isometric drawings that describe the SI system configurations and had acceptably confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). 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 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 inspection report.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Unit 2 containment building;
- Fire Zone 141 primary auxiliary building (PAB) corridor north;
- Fire Zone 237 component cooling water (CCW) heat exchanger and boric acid tank room; and
- Fire Zone 184/185 PAB 26-foot elevation south.

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 as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient. Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that 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

Failure to Identify Hydrogen Fire Hazards on Pre-Fire Plan

Introduction: A finding of very low safety significance and associated violation of license condition 4.F was identified by the inspectors for the failure to identify hydrogen fire hazards in the pre-fire plan for the 26-foot elevation of the PAB.

Description: During a tour of the 26-foot elevation of the PAB, the inspectors noted that there was a compressed gas cylinder labeled as containing hydrogen in the Unit 1 sample room. In addition, placards were placed on the entryways to the Unit 1 and Unit 2 Volume Control Tank (VCT) valve galleries indicating that there was hydrogen in the area. The licensee's fire protection coordinator confirmed with operations personnel that there was hydrogen piping in the areas.

The inspectors reviewed the pre-fire plan for the 26-foot elevation of the PAB, Fire Emergency Plan (FEP) 4.8, "PAB," Revision 7, and noted that the plan only discussed that there were gas sample lines in the Unit 1 sample room and that the rooms had compressed gas. The plan did not mention that there was a compressed hydrogen gas cylinder in a sample room. The inspectors noted that although compressed gas cylinders often have relief valves, they can still rupture under fire conditions due to weakening of steel at high temperatures. In addition, the pre-fire plan did not discuss the presence of hydrogen piping in the VCT valve galleries. The inspectors reviewed the National Fire Protection Association (NFPA) Fire Protection Handbook, 20th Edition, and determined that hydrogen presents unique hazards. Specifically, hydrogen burns with a non-luminous flame which is often invisible in daylight. Consequently, people can walk into its flames unaware. Additionally, hydrogen has an extremely wide flammability range, the highest burning velocity of any gas, and very low ignition energy.

Section 3.1.4.5.1 of the Fire Protection Evaluation Report (FPER) discussed the requirements for FEPs. Section 3.1.4.5.1 stated that fire, radiological, electrical, and physical hazards were listed for each plan area.

The licensee initiated Action Request (AR) 01179722, "Flammable Gas Question from the NRC," dated August 24, 2010, and revised FEP 4.8, on September 16, 2010. The inspectors verified that the revised FEP 4.8 included appropriate language discussing hydrogen hazards identified by the inspectors.

<u>Analysis</u>: The inspectors determined that failure to identify hydrogen fire hazards in the pre-fire plan for the 26-foot elevation of the PAB was contrary to the licensee's FPER and was a performance deficiency. Specifically, the licensee failed to identify that a compressed gas cylinder in the Unit 1 sample room contained hydrogen and that the VCT valve galleries contained hydrogen piping. In accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated December 24, 2009, the finding was determined to be more than minor because the failure to identify hydrogen fire hazards in the pre-fire plan was associated with the Mitigating Systems Cornerstone attribute of Protection Against External Events (Fire) and affected the cornerstone objective of preventing undesirable consequences (i.e., core damage). Specifically, the failure to identify hydrogen fire hazards in the pre-fire plan could impact the fire brigade's ability to effectively fight a fire due to the unique hazards associated with hydrogen. The inspectors reviewed IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 3b, "SDP Phase 1 Screening Worksheet for Initiating Events, Mitigation Systems, and Barriers Cornerstones," dated January 10, 2008. Based on this review, the inspectors determined that significance determination required evaluation using IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," dated December 22, 2006, because IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated February 28, 2005, does not address fire brigade issues. The inspectors determined that the finding was of very low safety significance because the fire brigade consisted of plant operators familiar with the 26-foot elevation of the PAB and associated hazards.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not reflective of current performance. Specifically, Revision 7 of FEP 4.8 had been prepared in 2004, which is not reflective of current performance. Although several pre-fire plans were reviewed as part of a focused self-assessment performed in February 2010, FEP 4.8 was not one of the pre-fire plans reviewed.

<u>Enforcement</u>: License condition 4.F for both Unit 1 and Unit 2 required the licensee to implement and maintain in effect all provisions of the approved fire protection program as described in the FSAR and Safety Evaluation Report dated August 2, 1979, (and Supplements dated October 21, 1980, January 22, 1981, and July 27, 1988) and the Safety Evaluation Report issued January 8, 1997, for Technical Specification (TS) Amendment No. 170. Section 9.10 of the FSAR stated that the FPER was incorporated into the FSAR by reference. Section 3.1.4.5.1 of the FPER stated that FEPs listed fire, radiological, electrical, and physical hazards for each plan area. Fire Emergency Plan (FEP) 4.8, "PAB," Revision 7, was the FEP for the 26-foot elevation of the PAB plan area.

Contrary to the above, from October 29, 2004, through August 24, 2010, FEP 4.8, Revision 7, did not list the fire hazards for the 26-foot elevation of the PAB plan area. Specifically, FEP 4.8 did not list fire hazards in the 26-foot elevation of the PAB plan area which included a compressed gas cylinder containing a flammable concentration of hydrogen gas in the Unit 1 sample room and hydrogen piping in the Unit 1 and Unit 2 VCT galleries. The licensee is in transition to NFPA 805 and, therefore, the NRC-identified violation was evaluated in accordance with the criteria established by Section A of the NRC's Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR Part 50.48) for a licensee in NFPA 805 transition. The inspectors determined that for this violation the licensee would not have identified the violation during the scheduled transition to 10 CFR Part 50, Section 48(c), because the licensee did not plan on performing reviews of pre-fire plans as part of their NFPA 805 transition. Consequently, the inspectors determined that enforcement discretion was not appropriate for this violation. Because this violation was of very low safety significance and it was entered into the licensee's CAP as (AR01179722), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000266/2010004-01; 05000301/2010004-01, Failure to Identify Hydrogen Fire Hazards on Pre-Fire Plan).

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

- fire drill in the circulating water pump house (Fire Area A38).

On August 25, 2010, the inspectors observed fire brigade activation for a drill simulating a cable tray fire in the circulating water pumphouse. Based on this observation, the inspectors evaluated 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:

- proper wearing of turnout gear and self-contained breathing apparatus;
- proper use and layout of fire hoses;
- employment of appropriate fire fighting techniques;
- sufficient firefighting equipment brought to the scene;
- effectiveness of fire brigade leader communications, command, and control;
- search for victims and propagation of the fire into other plant areas;
- smoke removal operations;
- utilization of pre-planned strategies;
- adherence to the pre-planned drill scenario; and
- 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 were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On September 27, 2010, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations 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 were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

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

- issue-oriented review of emergency diesel generator G01 due to observation of inaccuracies identified in electrical inspection procedure; and
- issue-oriented review of CCW after pump failure due to high bearing temperature.

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 (SSCs)/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 two quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings 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:

- risk management with gas turbine generator out-of-service due to failed turning gear;
- risk management during Unit 1 unplanned outage to repair generator hydrogen leaks;
- routine work week risk management during failure of Unit 1 turbine generator protection relay; and
- risk management while constructing scaffolding in the control room.

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 four samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- motor-operated valve 1SI-826C, safety injection suction from boric acid storage tank operability when using manual handwheel;
- Unit 2 control rod shroud temperature high alarm;
- Unit 2 containment fan coil unit “C” degraded flow condition;
- diesel fire pump P-35 auto start occurred out of accepted pressure band;
- SW-00165 packing leak and possible flange leakage;
- unexpected safety injection accumulator 2T-34A pressure low alarm; and
- indication of possible minor damage to cell divider on D-05.

The inspectors selected these potential operability and functionality issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that functionality and TS operability 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 TS and FSAR to the licensee’s evaluations to determine whether the components or systems were operable. 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 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 seven samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- Unit 1 emergency diesel generator G-01 after maintenance outage;
- Unit 1 battery charger 1D-207 after maintenance and repair;
- Unit 2 "A" CCW pump after maintenance; and
- Unit 1 "A" CCW pump after maintenance.

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 were identified.

1R20 Outage Activities (71111.20)

.1 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled outage that began on July 9, 2010, and continued through July 11, 2010. The Unit 2 unplanned outage was caused by a feedwater regulating valve (FRV) positioner failure. The positioner failure caused the "A" FRV to fail open, and steam generator level to increase until the steam generator high level bistable actuated, which drove the FRV full shut per design. The FRV continued to cycle in this manner until operators manually tripped the reactor. The licensee's root cause evaluation (RCE01176850) of this event concluded that a manufacturing defect caused the positioner to fail. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the schedule for the resulting outage.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heatup activities, and identification and resolution of problems associated with the outage.

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

b. Findings

No findings were identified.

.2 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled outage that began on July 26, 2010, and continued through July 28, 2010. The unplanned outage was caused by a loss of condenser vacuum, which occurred when the main generator was tripped for maintenance to repair a leak in the hydrogen gas system used for cooling the generator. Due to a valve mispositioning error, the turbine crossover steam dump system was left in service. This created a path between atmosphere and the condenser after the turbine trip, which caused the loss of condenser vacuum. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heatup activities, and identification and resolution of problems associated with the outage.

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

b. Findings

No findings were identified.

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:

- Units 1 and 2 safety-related calibration and testing of electrical bus protection relays;
- Units 1 and 2 nuclear instrumentation power range channel operational test;
- P32A service water pump monthly test;
- Unit 2 safe shutdown fire door inspections;
- Unit 2 CCW pump and value test (inservice testing) and;
- Unit 2 auxiliary feedwater pump and control valve test (containment isolation valve).

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 FSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component 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 four routine surveillance testing samples, one inservice testing sample, and one containment isolation valve sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

Inadequate Acceptance Criteria for Fire Door Surveillance Procedure

Introduction: A finding of very low safety significance was identified by the inspectors for the failure to provide appropriate acceptance criteria for the fire door surveillance procedure. Specifically, the acceptance criteria for fire door functionality did not specify that doors, when opened, returned to the closed and latched position

Description: The inspectors reviewed the surveillance for fire doors completed on May 19, 2010. Facilities maintenance personnel performed the surveillance in accordance with routine maintenance procedure (RMP) 9011-1, “Safe Shutdown Fire Door Inspections,” Revision 11. The inspectors noted that RMP 9011-1 listed the acceptance criteria for door functionality as “door closes and latches properly.” The inspectors noted that the criteria did not specify whether the door should close on its own due to the door closer or whether assistance from a person was acceptable. The NFPA standard for fire doors, NFPA 80, requires that installed door closers be able to close and latch the doors and that, when opened, doors return to the closed position on their own.

During plant walkdowns conducted on August 24, 2010, the inspectors observed that Doors 483 and 486 had a high differential pressure across them due to normally operating ventilation equipment and would not close on their own with just the installed door closers. Substantive effort was required to ensure that the doors were closed and latched. In addition, the inspectors observed that Door 156, which was not in an area having a high differential pressure, would not close on its own with the door closer. The inspectors noted that the surveillance completed on May 19, 2010, listed these doors as having satisfactorily passed the acceptance criteria. The inspectors concluded that personnel performing the surveillance were not requiring the door closers to be able to close and latch the doors consistent with regulatory requirements.

The inspectors noted that Doors 156, 483, and 486 were alarmed doors which, if held open for excessive time, would result in an alarm being generated. For these particular doors, personnel would be dispatched to investigate the doors if they were not fully closed. However, not all of the safe shutdown fire doors were monitored or would result in an alarm being generated if they were left open.

The “Point Beach Nuclear Plant Procedure Writers’ Guide,” Revision 13, was in effect at the time RMP 9011-1, Revision 11, was implemented. The writers’ guide established format and content requirements for the Point Beach Nuclear Plant procedures except for abnormal operating procedures, emergency operating procedures, associated supporting documents, and installation work plans. Section 3.8 of the writers’ guide specified that procedure acceptance criteria sections identify specific information required for determining a procedure had been successfully completed to satisfy regulatory requirements. The inspectors determined that RMP 9011-1, Revision 11, did not satisfy the requirements of Section 3.8.1 of the writers’ guide because it did not provide acceptance criteria that met regulatory requirements. Specifically, NFPA 80, “Standard for Fire Doors and Windows,” specifies that a closing mechanism will ensure that a door is in the closed and latched position. National Fire Protection Association (NFPA) 80 also specifies that doors, when opened, return to the closed and latched position.

Based on questions from the inspectors, the licensee initiated AR01179736, “NRC inspector question regarding maintenance checks of doors,” dated August 24, 2010. In addition, the licensee revised RMP 9011-1 and RMP 9011-2, “Industrial Fire Door, HELB Door and Seismic 2/1 Door Inspections, on September 16, 2010. However, in revising RMP 9011-1 and RMP 9011-2, the licensee made the acceptance criteria more ambiguous. Specifically, the licensee added notes to the procedures stating that it was acceptable, during surveillance inspections, to close a door by pushing on the door surface until the door latches to overcome air pressure. The inspectors were concerned that these notes effectively negated the NFPA 80 requirement for doors to return to the closed position when opened. The licensee did not identify which specific doors had ventilation issues (preventing automatic door closure) and were monitored for closure. Consequently, the notes could be applied to doors that should have been able to close automatically and were not monitored for closure. As such, the criteria could result in a nonfunctional door-closing mechanism and a degraded fire barrier not being detected during surveillance activities using the procedures. To address this issue, the licensee initiated AR01181260, "Revision to RMP 9011 requires additional guidance," dated September 22, 2010, and planned to revise the procedure again.

Analysis: The inspectors determined that the failure to provide appropriate acceptance criteria for the fire door surveillance procedure was contrary to the "Point Beach Nuclear Plant Procedure Writers' Guide" and was a performance deficiency. Specifically, the acceptance criteria for fire door functionality did not specify that doors, when opened, returned to the closed and latched position. In accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated December 24, 2009, the finding was determined to be more than minor because, if left uncorrected, the failure to have appropriate acceptance criteria would become a more significant safety concern. Specifically, the lack of appropriate fire door functionality acceptance criteria could result in a nonfunctional door-closing mechanism and a degraded fire barrier not being detected during surveillance activities. The inspectors determined that this finding was associated with the Mitigating Systems Cornerstone attribute of Protection Against External Events (Fire) and affected the cornerstone objective of preventing undesirable consequences (i.e., core damage).

In accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 3b, dated January 10, 2008, the inspectors determined the finding degraded the fire protection defense-in-depth strategies. Therefore, screening under IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated February 28, 2005, was required. The inspectors did not identify a function degradation. Specifically, the inspectors did not identify any instance where a fire door was left open or unlatched, or an instance where a fire door that would not close on its own was not monitored for closure. Consequently, the inspectors determined that the finding represented a low degradation and, as such, this finding screened as Green, having very low safety significance.

The inspectors determined that the finding had a cross-cutting aspect in the area of human performance, work practices, because the licensee personnel did not follow procedures. Specifically, the failure to follow procedures, such as the procedure writers' guide, resulted in the licensee's failure to provide appropriate acceptance criteria for the fire door surveillance procedure. This cross-cutting aspect is reflective of current performance because the licensee had reviewed and inappropriately revised the procedures during this inspection period [H.4(b)].

Enforcement: Section 3.8 of the "Point Beach Nuclear Plant Procedure Writers' Guide," Revision 13, specified that procedure acceptance criteria sections identify specific information required for determining a procedure had been successfully completed to satisfy regulatory requirements. Contrary to this, RMP 9011-1, Revision 11, did not identify specific information required for determining a procedure had been successfully completed to satisfy regulatory requirements. Specifically, the acceptance criteria for fire door functionality did not specify that doors, when opened, automatically returned to the closed and latched position. Although there were writers' guide procedural requirements to ensure procedure acceptance criteria sections identified specific information required for determining a procedure had been successfully completed to satisfy regulatory requirements, there were no specific regulatory requirements concerning procedural acceptance criteria specific to fire protection door surveillance procedures. Consequently, no violation of NRC requirements occurred. This issue was considered a finding of very low safety significance. The licensee entered this issue into their CAP as AR01179736 and AR01181260. The licensee also planned to revise the surveillance procedure. (FIN 05000266/2010004-02; 05000301/2010004-02, Inadequate Acceptance Criteria for Fire Door Surveillance Procedure)

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

These inspection activities supplement those documented in IR 05000266/2010002; 05000301/2010002, and constitute one complete sample as defined in IP 71124.01-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of radiation protection (RP) program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there have been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors evaluated whether the licensee assessed the potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed two radiological surveys from selected plant areas. The inspectors evaluated whether the thoroughness and frequency of the surveys are appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation.

- Units 1 and 2 auxiliary feedwater pump modification;
- Unit 1 charging pump modifications; and
- steam generator mausoleum.

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if hazards were properly identified, including the following:

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials;
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee’s program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors reviewed various containers holding non-exempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, “Labeling Containers,” or met the requirements of 10 CFR 20.1905(g), “Exemption To Labeling Requirements.”

The inspectors reviewed the following radiation work permits (RWPs) used to access high radiation areas (HRAs) and evaluated the specified work control instructions or control barriers.

- RWP 00000618-08, Locked High Radiation Area Airborne Neutron Containment Entries at Power;
- RWP 00000654-05, Filter Change Outs;
- RWP 00000670-11, Locked High Radiation Area Containment Entries for Start-up/Shutdown – Airborne; and
- RWP 00000687-07, Tri-Nuke Filter Removal.

For these RWPs, the inspectors assessed whether allowable stay time or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter (EPD) alarm setpoints were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker's EPD noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the corrective action program and dose evaluations were conducted as appropriate.

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee's means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

No findings were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiological controlled area and inspected the methods used for control, survey, and release from these areas. To evaluate whether the work was performed in accordance with plant procedures and whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site, the inspectors observed the performance of personnel surveying and releasing material for unrestricted use. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the type(s) of radiation present.

The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee has established a de facto "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact (i.e., they were not leaking their radioactive content).

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of EPDs in high noise areas as HRA monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the EPD was placed in the location of highest expected dose or whether the licensee properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- RWP 00000463-11, Cavity Activities Airborne, Unit 2 Reactor Vessel Head Set and Flange Cleaning;
- RWP 00000873, HRA, High Contamination Area, Airborne, Open and Inspect 1RH-718A; and
- RWP 00000873, HRA, High Contamination Area, Airborne, U-1 RHR HX [Heat Exchanger] Tube Plugging.

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including potentials for significant airborne levels (e.g., grinding, grit blasting, system breaches, entry into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air ventilation system operation.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors examined the posting and physical controls for selected HRAs and very high radiation areas to verify conformance with the Occupational PI.

b. Findings

No findings were identified.

.6 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors assessed whether workers were aware of the radiological conditions in their workplace and the radiation worker performance controls/limits in place, and whether their performance reflected the level of radiological hazards present.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions planned or taken.

b. Findings

No findings were identified.

.7 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technicians with respect to all radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits, and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings were identified.

.8 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's corrective action program. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involved radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings were identified.

2RS2 Occupational As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71124.02)

The inspection activities supplement those documented in IR 05000266/2010002; 05000301/2010002.

.1 Radiation Worker Performance (02.05)

a. Inspection Scope

The inspectors observed radiation worker and radiation protection technician performance during work activities being performed in radiation areas, airborne radioactivity areas, or high radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice (e.g., workers are familiar with the work activity scope and tools to be used, workers used ALARA low-dose waiting areas) and whether there were any procedure compliance issues (e.g., workers are not complying with work activity controls). The inspectors observed radiation worker performance to assess whether the training and skill level was sufficient with respect to the radiological hazards and the work involved.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index - Emergency AC Power Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency Alternating Current (AC) Power Systems performance indicator (PI) for Units 1 and 2 for the third quarter 2009 through the second quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports, and NRC inspection reports (IRs) for July 2009 through June 2010 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 emergency AC power system samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - High Pressure Injection Systems PI for Units 1 and 2 for the third quarter 2009 through the second quarter 2010. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC IRs for July 2009 through June 2010 to validate the accuracy of the submittals. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI 99-02, Revision 6, were used. 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 high pressure injection system samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - Heat Removal Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Heat Removal Systems PI for Units 1 and 2 for the third quarter 2009 through the second quarter 2010. To determine the accuracy of this PI data, definitions and guidance contained in NEI 99-02, Revision 6, were used. The inspectors reviewed the licensee's operator narrative logs, corrective action reports, event reports, MSPI derivation reports, and NRC IRs for October 2009 through June 2010 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 were identified.

.4 Mitigating Systems Performance Index - Residual Heat Removal Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Residual Heat Removal System PI for Units 1 and 2 for the third quarter 2009 through the second quarter 2010. To determine the accuracy of the PI data, definitions and guidance contained in NEI 99-02, Revision 6, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports, and NRC IRs for October 2009 through June 2010 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 residual heat removal system samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.5 Mitigating Systems Performance Index - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Cooling Water Systems PI for Units 1 and 2 for the third quarter 2009 through the second quarter 2010. To determine the accuracy of the PI data, PI definitions and guidance contained in NEI 99-02, Revision 6, were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports, and NRC IRs for July 2008 through June 2010 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 and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI cooling water system samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.6 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent Technical Specifications (RETS)/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences PI for the third quarter 2009 through July 2010. 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 October 2009 and July 2010 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 RETS/ODCM radiological effluent occurrences sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.7 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the Reactor Coolant System (RCS) Specific Activity PI for Units 1 and 2 for the second quarter 2009 through the second quarter 2010. To determine the accuracy of the PI data, PI definitions and guidance contained in NEI 99-02, Revision 6, was used. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, issue reports, event reports, and NRC IRs for the second quarter 2009 through the second quarter 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a RCS sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two RCS specific activity samples (one sample per unit) as defined in IP 71151-05.

b. Findings

No findings were identified.

.8 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences PI for the fourth quarter 2009 through the second quarter 2010. To determine the accuracy of the PI data, PI definitions and guidance contained in NEI 99-02, Revision 6, 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 numerous 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 were identified.

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

.1 Response to Unplanned or Non-Routine Events

a. Inspection Scope

The inspectors reviewed the plant's response to the following non-routine events.

- post-trip review of June 19, 2010 reactor trip;
- post-trip review of July 9, 2010 reactor trip;
- post-trip review of July 26, 2010 reactor trip; and
- unplanned reactor downpower to 15 percent with turbine offline for hydrogen leak repair.

Documents reviewed are listed in the Attachment to this report.

This event follow-up review constituted four samples as defined in IP 71153-05.

b. Findings

No findings were identified.

4OA5 Other Activities

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

As documented in Section 1R04, the inspectors confirmed the acceptability of the described licensee's actions. This inspection effort counts towards the completion of TI 2515/177, which will be closed in a later inspection report.

.2 (Closed) NRC TI 2515/173, "Review of the Industry Ground Water Protection Voluntary Initiative"

a. Inspection Scope

An NRC assessment was performed of the licensee's implementation of the NEI - Ground Water Protection Initiative (dated August 2007 (ML072610036)). The inspectors assessed whether the licensee evaluated work practices that could lead to leaks and spills and performed an evaluation of structures, systems, and components that contain licensed radioactive material to determine potential leak or spill mechanisms.

The inspectors assessed whether the licensee completed a site characterization of geology and hydrology to determine the predominant ground water gradients and potential pathways for ground water migration from onsite locations to offsite locations. The inspectors also determined if an onsite ground water monitoring program had been implemented to monitor for potential licensed radioactive leakage into ground water and that the licensee had provisions for the reporting of its ground water monitoring results. (See http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html)

The inspectors reviewed the licensee's procedures for the decision making process for potential remediation of leaks and spills, including consideration of the long-term decommissioning impacts. The inspectors also assessed whether records of leaks and spills were being recorded in the licensee's decommissioning files in accordance with 10 CFR 50.75(g).

The inspectors reviewed the licensee's notification protocols to determine whether they were consistent with the Ground Water Protection Initiative. The inspectors assessed whether the licensee identified the appropriate local and state officials and conducted briefings on the licensee's ground water protection initiative. The inspectors also determined if protocols were established for notification of the applicable local and state officials regarding detection of leaks and spills.

b. Findings

No findings were identified.

.3 (Closed) 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

The inspectors verified that the onsite documentation, system hardware, and licensee actions were consistent with the information provided in the licensee's response to NRC Generic Letter (GL) 2008-01, "Managing Gas Accumulation in Emergency Core Cooling [ECCS], Decay Heat Removal [DHR], and Containment Spray Systems." Specifically, the inspectors verified that the licensee has implemented or was in the process of implementing the commitments, modifications, and programmatically controlled actions described in the licensee's response to GL 2008-01. The inspection was conducted in accordance with TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (NRC Generic Letter 2008-01)," and considered the site-specific supplemental information provided by the Office of Nuclear Reactor Regulations (NRR) to the inspectors. In addition, members of the NRR staff participated in this inspection.

b. Inspection Documentation

The selected TI areas of inspection were licensing basis, design, testing, and corrective actions. The documentation of the inspection effort and any resulting observations are below.

Licensing Basis: The inspectors reviewed selected portions of licensing basis documents to verify that they were consistent with the NRR assessment report and that they were processed by the licensee. The licensing basis verification included the verification of selected portions of TSs, TS bases, FSAR, and technical requirements manual. The inspectors also verified that applicable documents that described the plant and plant operation, such as calculations, P&IDs, procedures, and CAP documents, addressed the areas of concern and were changed if needed following plant changes. The inspectors also confirmed that the frequency of selected surveillance procedures were at least as frequent as required by TSs. Finally, the inspectors confirmed that: (1) the licensee will review and evaluate the resolution of TS issues with respect to the changes contained in the technical specification task force (TSTF) traveler following

NRC approval; and (2) that a license amendment request will be submitted to the NRC within 180 days following the evaluation, if necessary. The completion date for this regulatory commitment is contingent upon the approval of the TSTF.

Design: The inspectors reviewed selected design documents, performed system walkdowns, and interviewed plant personnel to verify that the design and operating characteristics were addressed by the licensee. Specifically:

- The inspectors assessed the licensee's efforts for identifying the gas intrusion mechanisms that apply to the plant. The inspectors noted that the licensee failed to identify that steam voids would occur at the residual heat removal (RHR) system during a loss-of-coolant-accident (LOCA) in Mode 4. The details of this issue are described in Section 4OA5.3.c of this report.

  The inspectors also verified the licensee had identified the gas intrusion mechanisms associated with two operability evaluations performed for the discovery of voids at the RHR system in an earlier inspection period. This additional activity counted towards the completion of this TI and was documented in Inspection Report 05000266/2010002; 05000301/2010002.

- The inspectors assessed the licensee's void acceptance criteria and noted that the licensee established void volume acceptance criteria for piping locations located at system high points to be used during field verifications. The void volumes were derived based on pipe internal diameter and as-built slope, and internal height of the void. In addition, the licensee relied on the use of the computer software GOTHIC to further evaluate voids that did not meet field acceptance criteria by factoring in void transport behavior into the analysis. GOTHIC performs two-phase and two-component analysis of gas movement to predict such behavior as how a void volume in piping is translated into a transient void fraction at the entrance of a pump following pump start.

  The inspectors noted instances where the basis of the void assessment analysis was questionable. Specifically, the licensee used Westinghouse document WCAP-16631-NP, "Testing and evaluation of gas transport to the suction of ECCS pumps," to justify the acceptability of the field acceptance criteria of 0.300 ft^3 for the RHR full flow test line and 1.421 ft^3 for the RHR hot leg suction piping, where the 0.300 ft^3 criterion was also used as a basis to establish the 1.421 ft^3 criterion. WCAP-16631-NP documented tests that were conducted by Westinghouse to study the transport of a gas void through a piping system. It stated that the attenuation of the void due to the vertical-to-horizontal elbow at the bottom of the vertical pipe can be estimated as a reduction in the void fraction by 20 percent based on the void measurement near the bottom of the vertical pipe and in the lower horizontal pipe. However, the inspectors questioned if a significant contributor to the void fraction change was the elevation pressure change between the two locations as opposed to the elbow. In addition, the inspectors noted that the difference between test and plant pressures was not considered in assessing void decrease in the vertical test section.

  Similarly, the licensee used WCAP-16631-NP to show that GOTHIC can acceptably predict quantitative void transport behavior. However, the inspectors noted that test configuration and conditions differed from actual plant configuration and conditions, and questioned if the application of some of the test results was acceptable. For example:

1. The difference between test and plant pressures was not considered in assessing void decrease in the vertical test section. As already discussed, the pressure range used during the test was significantly lower than the typical range in nuclear power plants. This effect would be insignificant in a nuclear power plant due to the higher pressures. Therefore, the inspectors questioned if the void fraction change observed during testing would be analogous in a nuclear power plant.

2. Two-phase fluid flow test data typically exhibited significant scatter. This was addressed by running many duplicate tests and carefully examining the test results. However, NRR stated in ML090150637, "Forthcoming Meeting With the Nuclear Energy Institute To Discuss NRC Generic Letter 2008-01," that this effort was not fully successful and some of the conclusions were not adequately supported by the test data due to data scatter. For example, this effort did not address allowance for uncertainty and the effect of actual plant pressures in contrast to test pressures.

3. The inspectors questioned if the test report adequately considered a "water fall" effect (also known as "hydraulic jump") when the upper part of the vertical pipe was voided. Specifically, the inspectors questioned if the pipe length used for the test was representative of the limiting conditions of a plant. The inspectors were concerned if such an effect could propel air further down in the pipe than would be predicted using a single dimensional Froude number and would be of concern if the vertical pipe length was significantly less than the pipe used for the test.

4. The use of an average of pipe slopes to determine an equivalent pipe length associated with an elbow with a void reduction of 20 percent was debatable. For example, the average slope of -0.055 was obtained from slopes of -0.333, -0.15, and -0.0883. In addition, as discussed above, the 20 percent factor does not consider the pressures that will be encountered in nuclear power plants.

The inspectors discussed these observations with NRR. It was determined that these observations required further evaluation by NRR to: (1) better understand the acceptability of the application of the test results contained in WCAP-16631-NP to void assessment analysis; and (2) assess potential generic implications. The licensee captured these observations in their CAP as AR01177572.

The inspectors also reviewed the void acceptance criteria used by the licensee when evaluating two operability evaluations performed for the discovery of voids at the RHR system in an earlier inspection period. This additional activity counted towards the completion of this TI and was documented in Inspection Report 05000266/2010002; 05000301/2010002.

- The inspectors selectively reviewed applicable documents, including calculations, engineering evaluations, and vendor technical manuals, with respect to gas accumulation in RHR. Specifically, the inspectors verified that these documents addressed venting requirements, keep-full systems, and void control during system realignments.

- The inspectors conducted a walkdown of selected accessible portions of RHR in sufficient detail to assess the licensee's walkdowns. The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspector's independent walkdown. In addition, the inspectors verified that the licensee had P&IDs and isometric drawings that describe the RHR system configurations and had confirmed the accuracy of the drawings. The inspectors' review of the selected portions of isometric drawings considered the following:

  1. High point vents were identified.

  2. High points that do not have vents were recognizable.

  3. 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 slopped piping, and under closed valves, were described in the drawings or in referenced documentation.

  4. Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified.

  5. All pipes and fittings were clearly shown.

  6. 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 also conducted similar walkdowns of selected inaccessible portions of RHR and accessible portions of the SI in other inspection periods. These additional activities counted toward the completion of this TI and were documented in Inspection Reports 05000266/2009005; 05000301/2009005, and 05000266/2010004; 05000301/2010004.

- The inspectors verified that licensee's walkdowns have been completed. In addition, the inspectors selectively verified that information obtained during the licensee's walkdowns were addressed in procedures, the CAP, and training documents.

Testing: The inspectors reviewed selected surveillance, post-modification test, and post-maintenance test procedures and results to verify that the licensee has approved and was using procedures that were adequate to address the issue of gas accumulation and/or intrusion in the subject systems. This review included the verification of procedures used for conducting surveillances and determination of void volumes to ensure that the void criteria was satisfied and will be reasonably ensured to be satisfied until the next scheduled void surveillance. Also, the inspectors reviewed procedures used for filling and venting following conditions which may have introduced voids into the subject systems to verify that the procedures addressed testing for such voids and provided processes for their reduction or elimination.

The inspectors also review selected portions of procedures used during the surveillance testing of subject systems in a separate inspection activity. This additional activity counted towards the completion of this TI and was documented in Inspection Report 05000266/2010003; 05000301/2010003.

Corrective Actions: The inspectors reviewed selected licensee's assessment reports and CAP documents to assess the effectiveness of the licensee's CAP when addressing the issues associated with GL 2008-01. In addition, the inspectors verified that selected corrective actions identified in the licensee's nine-month and supplemental reports were documented. The inspectors also conducted a similar review of CAP documents in a separate inspection activity. This additional activity counted towards the completion of this TI and was documented in Inspection Report 05000266/2010003; 05000301/2010003.

Documents reviewed are listed in the Attachment to this report.

Based on this review, the inspectors concluded that there is reasonable assurance that the licensee will complete all outstanding items and incorporate this information into the design basis and operational practices. Therefore, this TI is considered closed.

c. Findings

(1) Failure to Ensure That RHR Would Be Capable to Respond to a LOCA in Mode 4

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure to ensure that the ECCS mode of operation of RHR would be capable of performing its mitigating function in Mode 4 following RHR realignment from its decay heat removal mode of operation.

Description: On June 22, 2010, the inspectors identified that the RHR system could experience flash evaporation during a shutdown LOCA in Mode 4. The inspectors were concerned that this condition could lead to steam binding the RHR pumps and/or an adverse waterhammer following system realignment to the refueling water storage tank (RWST).

The inspectors noted that on November 17, 2009, the licensee completed an evaluation of Nuclear Safety Advisory Letter (NSAL) 09-8, "Presence of Vapor in ECCS/RHR in Modes 3/4 LOCA Conditions." This operating experience discussed the potential for water flashing to steam in the RHR suction piping upon switching from DHR to ECCS mode of operation. High temperature water in the RHR system had the potential to flash to steam during a LOCA scenario while heating-up in Mode 3 or while heating-up or cooling-down in Mode 4. Specifically, the RHR system operating in its DHR mode of operation would be at RCS temperature and pressure. Following a LOCA, the trapped fluid in the RHR lines might flash because it would suddenly be exposed to lower pressures.

The licensee's evaluation of NSAL-09-8 considered the lower pressure resulting from swapping the suction of RHR over to the RWST following the system alignment to its ECCS mode of operation. The RWST is open to the atmosphere. The licensee was concerned that the swap over to the RWST would lead to flash evaporation of the RHR system because RCS conditions exceed the saturation conditions provided by the RWST. The licensee performed a calculation to determine the maximum allowable RHR temperature to preclude void formation in the RHR pump suction header considering the static head of and ambient pressure on the RWST and the maximum temperature allowable at the RHR hot leg suction. The calculated maximum allowable RHR temperature was 272^{o}F. However, the licensee concluded that RHR would not experience steam formation because: (1) procedures would require venting RHR after it has cooled down to at least 250°F prior to aligning the system to its ECCS mode of operation in preparation for entry to Mode 3; and (2) during a LOCA in Mode 4, it was likely that by the time RHR would be realigned to take suction from the RWST it would be cooled to near or below 272°F.

The inspectors noted, however, that both RHR systems would experience steam void formation if a LOCA occurred that was of sufficient size to depressurize the RCS. Specifically, the RHR system is subjected to RCS temperature and pressure when operated in the DHR mode, which exceeds saturation conditions of water at atmospheric pressure. During a shutdown-large-LOCA, the reactor could depressurize to atmospheric conditions. This would result in the flash evaporation of water inside the RHR system because its temperature would be above the saturation temperature of water at atmospheric pressure. In addition, the static head provided by the difference in height between the hot leg and the pump inlet could not be credited because the water volume at this location will experience flash evaporation if the temperature at the hot leg were above its saturation temperature. Consequently, the volume of water below it will not benefit from the resulting static head (i.e., there will be a reduction of the height of the column of water above it). As a result, this volume of water would flash if its temperature were above saturation temperatures. This effect would repeat itself all the way down to the lowest elevations. Therefore, the system would be voided before its suction was swapped over to the RWST.

Although the energy of the water volume is not enough to evaporate the entire volume, flashing will occur at all locations where saturation conditions are not met. Also, the resulting volume of the steam would be significantly greater than the water volume that existed before the flash evaporation. Specifically, a simplified thermodynamic analysis that assumes initial and final saturation conditions at 350°F and 212°F, respectively, determined that only approximately 14 percent of the mass of water would evaporate. However, the resulting steam volume would be approximately 208 times the initial volume of water (i.e., at 350°F) and 257 times the final volume of water (i.e., the fraction of water that did not evaporated). Therefore, the resulting steam volume would move to other locations, including the hot leg, possibly by displacing water volumes and further decreasing the available head. Based on this analysis, the inspectors concluded that the RHR system would be significantly voided.

The affected section of Point Beach's TSs was Section 3.5.3, "ECCS – Shutdown." It required one train of ECCS mode of operation of RHR to be operable in Mode 4, "Hot Shutdown," to ensure that sufficient ECCS flow is available to the core following a shutdown LOCA. However, TS 3.5.3 was modified by a note that allowed an RHR train to be considered operable during alignment and operation for DHR mode if capable of being manually realigned to the ECCS mode of operation and not otherwise inoperable. This allowed operation in the DHR mode during Mode 4 to provide force circulation for decay heat removal and transport. Both trains of RHR were typically placed into service by the licensee in the DHR mode to shorten the cooldown time. The DHR mode of RHR operation was governed by TS 3.4.6, "RCS Loops – Mode 4."

The licensee captured the inspectors' concerns in the CAP as AR01175986 and AR01175866. The corrective actions include procedure changes to ensure the operability of at least one RHR train while in Mode 4 and a Licensee Event Report.

Analysis: The inspectors determined that the failure to ensure that the ECCS mode of operation of RHR would be capable of performing its mitigating function in Mode 4 was contrary to 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the current procedures and design of RHR did not ensure that its ECCS mode of operation would be capable of performing its mitigating function in Mode 4. Steam voids could form during a LOCA that initiates at this Mode resulting in loss of safety function.

Since this concern only exists while the plant is in Mode 4, the Region III Senior Reactor Analyst (SRA) evaluated this finding in accordance with NRC IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process (SDP),” dated February 28, 2005. Appendix G, Attachment 1, Checklist 1, Pressurize Water Reactor Hot Shutdown Operation), stated that the finding required an SDP Phase II analysis since the finding increased the likelihood that a loss of DHR would occur. Therefore, the risk evaluation continued with Appendix G, Attachment 2 (SDP Template for Pressurized Water Reactors During Shutdown).

The exposure period was when either unit was operating in Mode 4 with RHR in service at a temperature high enough that flashing could occur if the system was suddenly depressurized. The licensee provided historical time periods during the past 3 years in its Technical Assessment for Reportability for AR01175866-04. The longest time of unavailability for a single unit during a 12-month period was about 23 hours.

The applicable initiating event for this condition finding was the loss of inventory event in Plant Operating State 1 (RCS Closed). Worksheet 5 of Phase II of the SDP notebook represented these conditions and was used for this risk evaluation. The SRA used a bounding exposure period of less than three days with no credit for RHR, credit for one train of safety injection, and nominal credit for the remaining modeled functions resulting in one sequence of “8.”

The licensee had an estimate in its Technical Assessment for Reportability showing that the core damage frequency (CDF) due to a large-break LOCA in Mode 4 was a factor of 6.5 lower than the full power CDF. This resulted in a contribution to plant CDF due to a large-break LOCA while in Mode 4 to be about 5 E-9 per year.

Based on the above, the SRA concluded that the risk due to the performance deficiency is very low (Green).

The inspectors determined that this finding did not have a cross-cutting aspect because it was not obvious that the licensee should have identified the potential problem with RHR.

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, as of June 22, 2010, the licensee did not correctly translate applicable regulatory requirements and the design basis into specifications and procedures. Specifically, the operability requirements of RHR in Mode 4 defined by TS 3.5.3 were not translated into applicable procedures or specifications of the system. Neither the procedures nor design prevented the conditions that would lead to steam void formation during a LOCA that initiates at this Mode resulting in loss of safety function. Because this violation was of very low safety significance and it was entered into the licensee's CAP (as AR01175986 and AR01175866), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000266/2010004-03; 05000301/2010004-03, Failure to Ensure that RHR Would Be Capable to Respond to a LOCA in Mode 4).

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 5, 2010, the inspectors presented the inspection results to Mr. L. Meyer 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 results of the radiation safety inspection conducted August 16 through August 20, 2010, with the Acting Plant Manager, Mr. R. Farrell, on August 20, 2010;
- The results of the radiation safety inspection conducted August 30 through September 3, 2010, with Engineering Director, Mr. C. Trezise, on September 3, 2010; and
- The results of the managing gas accumulation inspection, conducted during this quarter, on October 5, 2010, with Mr. J. Costedio and other members of the licensee staff.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Amundson, Operations Training General Supervisor
B. Beltz, Assistant Operations Manager
J. Costedio, Regulatory Affairs Manager
D. Craine, Radiation Protection Manager
F. Flentje, Regulatory Affairs Supervisor
R. Harrsch, Operations Manager
L. Meyer, Site Vice-President
A. Mitchell, Systems Engineering Manager
J. Schleif, Emergency Preparedness Manager
C. Trezise, Engineering Director
T. Vehec, Plant Manager

Nuclear Regulatory Commission

M. Kunowski, Chief, Division of Reactor Projects, Branch 5

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000266/2010004-01; 05000301/2010004-01	NCV	Failure to Identify Hydrogen Fire Hazards on Pre-Fire Plan (1R05)
05000266/2010004-02; 05000301/2010004-02	FIN	Inadequate Acceptance Criteria for Fire Door Surveillance Procedure (1R22)
05000266/2010004-03; 05000301/2010004-03	NCV	Failure to Ensure That RHR Would Be Capable to Respond to a LOCA in Mode 4 (4OA5)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

<u>1R04 Equipment Alignment</u>

- AR01180655; IA-300 HX-99A IA Compressor Aftercooler HX Drain Is Leaking By
- AR01180656; IA Compressor Has Various Oil Leaks
- CL 5A; Chemical and Volume Control System Unit 1; Revision 24
- CL 7A; Safety Injection System Checklist Unit 2; Revision 32
- CL 7B; Safety Injection System Checklist Unit 2; Revision 30
- CL 9R; Instrument Air; Revision 33
- Drawing 018984; P&ID Safety Injection System; Revision 57
- Drawing 018985; P&ID Safety Injection System; Revision 61
- Drawing 018986; P&ID Safety Injection System; Revision 48
- Drawing 20070A; P Chemical and Volume Control; Revision 71
- Drawing M-217; P&ID Auxiliary Feedwater System; Revision 87
- Drawing PB 01 MWSK000010; P&ID Service Water System; Revision 32
- Drawing PB 31 MIAK000002; P&ID Instrument Air System; Revision 13
- FSAR; Chapter 6; Engineered Safety Features
- FSAR; Section 9.3; Chemical And Volume Control System
- FSAR; Section 9.7; Instrument Air / Service Air
- IT 09B; TDAFP [Turbine-Driven Auxiliary Feedwater Pump] Suction From SW MOV Exercise Test (Quarterly) Unit 2; Revision 10
- Safety Monitor 3.5a.02; Unit 1; September 22, 2010
- TS 3.5.1; Accumulators
- TS 3.5.2; ECCS – Operating

<u>1R05 Fire Protection</u>

- AR01177362; AT-0175 Action Request Record Report; July 16, 2010
- AR01179722; Flammable Gas Question From The NRC
- Duke Engineering And Services Fire Area Analysis Summary Report; August 8, 2010
- FAP 3.0; Fire Attack Plans; Revision 9
- FEP 4.0; Fire Emergency Plan; Revision 5
- FEP 4.19; Circulating Water Pumphouse; Revision 8
- FEP 4.8; PAB; Revision 7
- FEP 4.8; PAB; Revision 8
- Fire Hazards Analysis Report; Revision 5
- Fire Protection Evaluation Report; Revision 10
- FOP 1.1; Brigade Training; Revision 9
- NP 1.9.9; Transient Combustible Control; Revision 18
- PBC 218 SH.1; Fire Protection For Site Plan; Revision 11
- PBC-218 SH.2; Fire Protection For Turbine Building, Auxiliary Building & Containment Elev. 8”-0”; Revision 20
- PBC-218 SH.3; Fire Protection For Turbine Building, Auxiliary Building & Containment Elev. 26’-0”; Revision 10

- PBC-218 SH.4; Fire Protection For Turbine Building, Auxiliary Building & Containment Elev. 44'-0"; Revision 9
- PBC-218 SH.5; Fire Protection For Turbine Building, Auxiliary Building & Containment Elev. 66'-0"; Revision 6
- PBSA-ENG-10-03; Fire Protection Program - Preparation For Triennial Inspection; April 26, 2010
- Safe Shutdown Analysis Report; Revision 7

1R12 Maintenance Rule Implementation

- Documentation Of Maintenance Rule Performance Criteria For Component Cooling Water; March 2, 2009
- Functional Failures For system CC worksheet; September 23, 2010
- IT 13 TRAIN A; 2P 11A, CCW Pump And Valves U2; September 21, 2010
- Maintenance Rule (a)(1) System Action Plan Checklist And Approval For CC System; June 3, 2009
- Maintenance Rule Unavailability Data Sheet For Unit 1, CC System; January 1, 2009 to September 1, 2010
- Performance Criteria Assessments For CC Since 1/1/2009; September 23, 2010
- RMP 9006-2; Component Cooling Water Pump Mechanical Seal (John Crane) Overhaul; Revision 31
- System Matrix/E.R. Dashboard; April 1, 2010 to June 30, 2010
- WO00392007 01; 2P-011A Replace IB And OB BEA Rings And Seals As Needed; September 20, 2010

1R13 Maintenance Risk Assessments and Emergent Work Control

- AR01164283; Action Request Report; March 10, 2010
- AR01173790; Action Request Report; May 20, 2010
- AR01173803; Action Request Report; May 20, 2010
- AR01177324; Action Request Report; July 15, 2010
- CR00032759; TG-01-G Electrical Generator; July 28, 2010
- CR00032791; Z-008 TG-01 Turbine Generator Condition Monitor; July 28, 2010
- CR00032837; TG-01-G Electrical Generator; July 28, 2010
- CR00032841; HG-00004 W061 H_2 Gas Purity Blwr Supply From TG-01 TG Top; July 28, 2010
- Checklist 00031204; Isolate H_2 To ITG-01-G For Manway Cover And Misc. Repairs; July 28, 2010
- Daily Status Report Unit 1; August 24, 2010
- Daily Status Report Unit 1; August 4, 2010
- Daily Status Report Unit 1; July 27, 2010
- Daily Status Report Unit 2; July 27, 2010
- Daily Status Report Unit 2; July 29, 2010
- EN 46129; Point Beach Nuclear Plant Event Notification Worksheet; July 26, 2010
- Narrative Log Report; July 20, 2010
- Narrative Log Report; July 26, 2010 Through July 28, 2010
- Narrative Log Report; July 30, 2010
- OI 32A; Pressurizing And Testing Main Generator With Air; Revision 4
- OP 1B; Reactor Startup; Revision 61
- OP3A; Power Operation To Hot Standby Unit 1; Revision 3
- Open Prompt Operability Determinations List; July 26, 2010

- Operational Decision Making Beckwith Relay Failure Unit 1; July 30, 2010
- Operational Decision Making Generator Hydrogen Leakage Unit 1; July 15, 2010
- Operational Decision Making Generator Hydrogen Leakage Unit 1; July 14, 2010
- Priority Work List; July 26, 2010
- Safety Monitor Calculation Unit 1; July 15, 2010
- Safety Monitor Calculation Unit 2; July 15, 2010
- Safety Monitor Calculation Unit 2; July 17, 2010
- Safety Monitor Calculation Unit 2; July 17, 2010
- Safety Monitor Calculation Unit1; July 17, 2010
- Safety Monitor Calculation Unit1; July 17, 2010
- Safety Monitor Calculation Unit 1; July 18, 2010
- Safety Monitor Calculation Unit 2; July 18, 2010
- Safety Monitor Calculation Unit 1; July 19, 2010
- Safety Monitor Calculation Unit 2; July 20, 2010
- Safety Monitor Calculation Unit 2; July 20, 2010
- Safety Monitor Calculation Unit 1; July 26, 2010
- Safety Monitor Calculation Unit 1; July 26, 2010
- Safety Monitor Calculation Unit 2; July 26, 2010
- Safety Monitor Calculation Unit 2; July 28, 2010
- Safety Monitor Change Notice; 0061
- Unit 1 Forced Outage Work List Test Requirements; July 27, 2010
- Unit 1 Forced Outage Work List; July 27, 2010
- Unit 1 Forced Outage Work List; July 26, 2010
- WO00391806; Z-500-M Motor Breaker Will Not Reset; July 19, 2010
- WR00038333; Work Request; June 27, 2010
- WR00044296; Work Request; July 15, 2010

1R15 Operability Evaluations

- 01178472; AT-0075 AR Screening; August 5, 2010
- AR01175119; AT-0075 AR Screening; June 10, 2010
- AR01177363; AT-0175 Action Request Record Report; July 16, 2010
- AR01178558; AT-0175 Action Request Record Report; August 5, 2010
- AR01178558; Prompt Operability Determination (POD); Revision 1
- AR01178558; Prompt Operability Determination (POD); Revision 2
- AR01178558; Prompt Operability Determination for UW “C” CFC SW Flow Trending Lower; Prepared August 8, 2010
- AR01180573; AT-0075 AR Screening; September 8, 2010
- AR01180889; AT-0075 AR Screening; September 15, 2010
- AR01180934; AT-0075 AR Screening; September 16, 2010
- AR01181443; AT-0075 AR Screening; September 25, 2010
- AR1178558-01; Engineering Task Pre-Job Briefing Form; August 6, 2010
- Drawing 110E017; P&ID Safety Injection System; Revision 55
- Evaluation of the Decreasing SW Flow Trend For Containment Fan Cooler 2HX-015C (EC16068-Rev. 1); Revision 1
- Evaluation Of The Decreasing SW Flow Trend For Containment Fan Cooler 2HX-015C (EC16068-Rev. 1); Performed August 5, 2010
- TS 34; Containment Accident Recirculation Fan-Cooler Units (Monthly) Unit 2 Attachment B Data Sheet; Performed August 17, 2010

1R19 Post-Maintenance Testing

- AR01176078; AT-0175 Action Request Record Report; June 27, 2010
- AR01177622; Action Request Report; July 21, 2010
- Documentation Of Maintenance Rule Performance Criteria For Component Cooling Water; March 2, 2009
- Functional Failures For System CC worksheet; September 23, 2010
- IT 13 TRAIN A; 2P 11A, CCW Pump And Valves U2; September 21, 2010
- LM 2.1; PBNP Lubrication Manual For PBNP Equipment Lube List; Revision 47
- Maintenance Rule (a)(1) System Action Plan Checklist And Approval For CC System; June 3, 2009
- Maintenance Rule Unavailability Data Sheet For Unit 1, CC System; January 1, 2009 to September 1, 2010
- MI 9644; Maintenance Instruction; Revision A
- Performance Criteria Assessments For CC since 1/1/2009; September 23, 2010
- RMP 9006-2; Component Cooling Water Pump Mechanical Seal (John Crane) Overhaul; Revision 31
- System Matrix/E.R. Dashboard; April 1, 2010 to June 30, 2010
- WO00382839/00382840; IP-011A Job Walkdown Checklist; August 3, 2010
- WO00382839; 1P-011A Change Oil, Flush Bearings And Clean Intake Grills; August 31, 2010
- WO00382840; 1P-011A Grease Coupling WM-0156 Completed WO Task Report; August 31, 2010
- WO00382840; 1P-011A Grease Coupling Work Order Package; August 31, 2010
- WO00382840; 1P-011A Operations PMT/RTS WM-0156 Completed WO Task Report; August 31, 2010
- WO00382840; Change Oil Job Safety Analysis; August 31, 2010
- WO00382840; Job Grease Coupling Job Safety Analysis; August 31, 2010
- WO00382840; WM-0156 Completed WO Task Report for 1P-011A Shift CCW Pump In Necessary; August 31, 2010
- WO00382840; Work Package Closeout Checklist; August 31, 2010
- WO00392007 01; 2P-011A Replace IB And OB BEA Rings And Seals As Needed; September 20, 2010
- Work Order Package 00350422 01; February 18, 2010
- Work Order Package 00351202 01; June 2, 2010
- Work Order Package 00351669 01; April 22, 2010
- Work Order Package 00351757 01; January 21, 2010
- Work Order Package 00351964 01; May 19, 2010
- Work Order Package 00351965 01; June 8, 2010
- Work Order Package 00352646 01; January 21, 2010
- Work Order Package 00354818 01; May 6, 2010
- Work Order Package 00355123 01; April 27, 2010
- Work Order Package 00355123 02; April 27, 2010
- Work Order Package 00355123 03; April 27, 2010
- Work Order Package 00355123 04; April 27, 2010
- Work Order Package 00355123 05; April 27, 2010
- Work Order Package 00355123 07; June 2, 2010
- Work Order Package 00361499 01; May 4, 2010
- Work Order Package 003644938 01; April 19, 2010
- Work Order Package 00368225 01; May 4, 2010
- Work Order Package 00368226 01; May 4, 2010
- Work Order Package 00372621 01; February 18, 2010

- Work Order Package 00372623 01; February 5, 2010
- Work Order Package 00372653 01; April 28, 2010
- Work Order Package 00373792 01; June 23, 2010
- Work Order Package 00378593 01; April 26, 2010
- Work Order Package 00388229 02; June 9,, 2010

1R20 Refueling and Other Outage Activities

- 00386437-02; WO Work Plan; April 5, 2010
- 1R22 Surveillance Testing
- 1RP 9056-1; Calibration And Testing Of Safety Related Protective Relays A-05; Revision 24
- AR01175644; Action Request Report; June 19, 2010
- AR01175650; Action Request Report; June 19, 2010
- AR01175719; Action Request Report; June 21, 2010
- AR01176098; Action Request Report; June 28, 2010
- AR01176472; Action Request Report; July 2, 2010
- AR01176850; Unit 2 SC 'A' Feed Reg. Valve Malfunction; July 9, 2010
- AR01177018; Action Request Report; July 14, 2010
- AR01178563; Action Request Report; August 5, 2010
- AR01178684; Action Request Report; August 7, 2010
- CR00032759; TG-01-G Electrical Generator; July 28, 2010
- CR00032791; Z-008 TG-01 Turbine Generator Condition Monitor; July 28, 2010
- CR00032837; TG-01-G Electrical Generator; July 28, 2010
- CR00032841; HG-00004 W061 H_2 Gas Purity Blwr Supply From TG-01 TG Top; July 28, 2010
- Checklist 00031204; Isolate H_2 to ITG-01-G For Manway Cover And Misc. Repairs; July 28, 2010
- Daily Status Report Unit 1; July 27, 2010
- Daily Status Report Unit 2; July 27, 2010
- Daily Status Report Unit 2; July 29, 2010
- EN 46129; Point Beach Nuclear Plant Event Notification Worksheet; July 26, 2010
- Fleet Outage Report PBN Unit 2 Station Forced Outage Status Report; July10, 2010
- Foreign Material Exclusion Checklist (FMEC) For WO386437; August 14, 2010
- IT 09A; Cold Start Of Turbine-Driven Auxiliary Feed Pump And Valve Test (Quarterly) Unit 2; Revision 50
- Job Safety Analysis For WO386437; August 12, 2010
- Narrative Log Report; July 26, 2010 Through July 28, 2010
- Narrative Log Report; July 30, 2010
- NP 5.33; Incident Investigation And Post-Trip Review; Revision 9
- OI 32A; Pressurizing And Testing Main Generator With Air; Revision 4
- OP 1B Appendix A; Estimated Critical Position Calculation; Revision 14
- OP 1B; Reactor Startup; Revision 61
- OP 1B; Reactor Startup; Revision 61
- OP 1C; Procedure Record And Field Copy Tracking; Revision 18
- OP 3A; Power Operation To Hot Standby Unit 1; Revision 3
- Open Prompt Operability Determinations List; July 26, 2010
- Operational Decision Making Beckwith Relay Failure Unit 1; July 30, 2010
- Operational Decision Making Generator Hydrogen Leakage Unit 1; July 15, 2010
- Operational Decision Making Generator Hydrogen Leakage Unit 1; July 14, 2010
- PBNP ICRR Plot; July 10, 2010

- Pre-Job, High Risk, Or Infrequently Performed Task or Evolution (IPTE) Brief Checklist For WO386437; Revision 14
- Priority Work List; July 26, 2010
- Rapid Operating Experience Report; July 10, 2010
- Safety Monitor Calculation Unit 1; July 26, 2010
- Safety Monitor Calculation Unit 1; July 26, 2010
- Safety Monitor Calculation Unit 2; July 26, 2010
- Safety Monitor Calculation Unit 2; July 28, 2010
- Unit 1 Forced Outage Work List Test Requirements; July 27, 2010
- Unit 1 Forced Outage Work List; July 26, 2010
- Unit 1 Forced Outage Work List; July 27, 2010
- Unit 2 Forced Outage Schedule (FRV Transient), From July 9, 2010 to July 11, 2010
- Unit 2 Forced Outage; From July 9, 2010 to July 11, 2010
- Work Activity Risk Evaluation Form For WO386437; August 4, 2010
- Work Order Package 00386437 01 Through 13; July 19 And July 22, 2010

<u>1R22 Surveillance Testing</u>

- 1ICP 02.007; Nuclear Instrumentation Power Range Channels 92 Day Channel Operational Test; Performed July 9, 2010 to September 2, 2010
- AR01176850; Unit 2 SC ‘A’ Feed Reg. Valve Malfunction; July 9, 2010
- AR01179736; NRC Inspector Question Regarding Maintenance Checks Of Doors
- AR01180537; SW-7 & SW-8 Closed – SW-2912 Failure
- AR01181260; Revision To RMP 9011 Requires Additional Guidance
- DBD-12; Service Water System; Revision 16
- Documentation Of Maintenance Rule Performance Criteria For Component Cooling Water; March 2, 2009
- Drawing PB 01 MWSK000008; ISI Classification Diagram Service Water; Revision 34
- Fleet Outage Report PBN Unit 2 Station Forced Outage Status Report; July 10, 2010
- Functional Failures For system CC worksheet; September 23, 2010
- IT 13 TRAIN A; 2P 11A, CCW Pump And Valves U2; September 21, 2010
- IT-07A; P-32 A Service Water Pump Test (Quarterly); Revision 25
- Maintenance Rule (a)(1) System Action Plan Checklist And Approval For CC System; June 3, 2009
- Maintenance Rule Unavailability Data Sheet For Unit 1, CC System; January 1, 2009 to September 1, 2010
- OI 70; Service Water System Operation; Revision 61
- Operator Logs For the Applicable Work Days
- Performance Criteria assessments For CC Since January 1, 2009; September 23, 2010
- Priority Work Schedule Printed March 9, 2010
- Rapid Operating Experience Report; July 10, 2010
- RMP 9006-2; Component Cooling Water Pump Mechanical Seal (John Crane) Overhaul; Revision 31
- RMP 9011-1; Safe Shutdown Fire Door Inspections; Performed May 19, 2010
- RMP 9011-1; Safe Shutdown Fire Door Inspections; Revision 11
- RMP 9011-1; Safe Shutdown Fire Door Inspections; Revision 12
- RMP 9011-2; Industrial Fire Door, HELP Door And Seismic 2/1 Door Inspections; Revision 8
- System Matrix/E.R. Dashboard; April 1, 2010 to June 30, 2010
- U2 Forced Outage; From July 9, 2010 to July 11, 2010
- Unit 2 Forced Outage Schedule (HSD(FRV Transient), From July 9, 2010 to July 11, 2010
- WO00379624; 1ICP 2.7 – Power Range NIS Quarterly Surveillance Unit 1; September 2, 2010

- WO00382895 01; P-32 A Service Water Pump Test
- WO00392007 01; 2P-011A Replace IB And OB BEA Rings And Seals As Needed; September 20, 2010

2RS1 Radiological Hazard Assessment and Exposure Controls

- AR01157837; Quick Hit Assessment; Occupational Dose Assessment And Radiological Hazard Assessment And Exposure Controls; August 11, 2010
- AR01159344; Quick Hit Assessment; Alpha Monitoring Program; July 26, 2010
- AR01169425; Worker Locked In Secured High Radiation Area; March 12, 2010
- AR01175266; Loose / Empty Radioactive Material Bag Found In RCA Yard; July 15, 2010
- AR01175691; Low Activity Particle Found In RCA Yard; July 21, 2010
- AR01179348; SGSF Lighting Panel 71L Has Water Coming Out The Bottom (Steam Generator Storage Facility); August 18, 2010
- HP 2.15.1; High Level Contamination And Discrete Radioactive Particle Control; Revision 05
- HP 3.2; Radiological Labeling, Posting And Barricading Requirements; Revision 50
- HP 3.6; Alpha Monitoring Program; Revision 1
- HPCAL 2.15; Small Articles Monitor Type SAM 9/11 Calibration And Efficiency; Various dates
- HPIP 1.66; Dosimetry Placement For Extremity And Multiple Whole Body Locations And Extremity Dose Determinations; Revision 14
- HPIP 3.50; Radiation Surveys; Revision 13
- HPIP 3.51; Contamination Surveys; Revision 19
- HPIP 3.52; Airborne Radioactivity Surveys; Revision 33
- HPIP 3.52.1; Radiological Sampling For Release Accountability; Revision 29
- HPIP 3.53; Counting Of Air Samples For Low Level, Long-Lived Radioactive Alpha Particulate Contamination; Revision 12
- HPIP 8.0; Source Control Program; Revision 11
- HPIP 8.1; Radioactive Source Inventory; Revision 6
- HPIP 8.2; Sealed Source Leak Testing; Revision 7
- MA-AA-112-1000; Conduct Of Radiological Diving Operations; Revision 00
- NP 4.2.12; Requirements For Radiologically Controlled Area Entry; Revision 23
- NP 4.2.16; Visitor Access To A Radiologically Controlled Area; Revision 16
- NP 4.2.17; Response to Exposure Events; Revision 6
- NP 4.2.19; Entry Requirements Into Various Radiologically Controlled Areas; Revision 17
- NP 4.2.27; Personnel Exposure Monitoring Device Minimum Requirements And General Use; Revision 18
- NSTS (National Source Tracking System) Annual Inventory Reconciliation; January 19, 2010
- PBF-4021; Radiological Surveys; Various Dates
- PBF-4022; Airborne Radioactivity Survey; Various Dates
- PBF-4085u; AMS-4 Calibration Data Sheet; August 30, 2010
- Point Beach Nuclear (PBN) Oversight Report 10-008; June 30, 2010
- RP-AA-101-2001; Sentinel Software Transactions Associated With Issuance And Control Of Personnel Monitoring Devices; Revision 02
- RWP 00000463-11; Cavity Activities Airborne; Unit 2 RVCH Set And Flange Cleaning
- RWP 00000618-08; LHRA Airborne Neutron Containment Entries At Power
- RWP 00000654-05; Filter Change Outs
- RWP 00000670-11; LHRA Containment Entries For Start-up/Shutdown - Airborne
- RWP 00000687-07; Tri-Nuke Filter Removal
- RWP 00000820-05; Containment Keyway And VHRA Entries
- RWP 00000873; HRA, HCA, Airborne; Open And Inspect 1RH-718A
- RWP 00000873; HRA, HCA, Airborne; U-1 RHR HX Tube Plugging

- RWP 00000877-00; Seal Table Eddy Current - Airborne
- RWP 10-0001; Radiation Protection Routine; Revision 00
- RWP 10-0012; USNRC Surveillance; Revision 00
- RWP 10-0014; Primary Filter Change-outs; Revision 00
- RWP 10-0017; EPU Activities; Revision 1
- RWP 10-2000; U2 Containment At Power Entries; Revision 00
- RWP 11-2011; Shutdown/Start-Up Activities; Draft
- RWP 11-2015; Seal Table Activities; Draft
- RWP 11-2020; Keyway Entries; Draft
- Source Leak Test Records; August 02, 2010

2RS2 Occupational ALARA Planning and Controls

- AR01167257; Quick Hit Assessment; RP Related Lessons Learned U1R32; June 21, 2010
- CY-AA-101-1000; Guidance For Outage Chemistry Control; Revision 1
- PBF-4246; Radiological Pre-Job Briefing Form; Revision 2
- RP-AA-104-1000; ALARA Implementing Procedure; Revision 1
- RP-AA-104-1001; Sentinel RWP Writer's Guide; Revision 02

4OA1 Performance Indicator Verification

- CAMP 310; Operation Of The Canberra GENIE 2000 Pro-Count And Inspector Portable Gamma Spectroscopy Counting Systems; Revision 07
- CAMP 401; Radioactive Standards And Sample Placement For Multichannel Analyzer Calibration And Quantification; Revision 07
- CAMP 410; Determination Of Radioactive Iodine And Iodine 131 Equivalents In Reactor Coolant; Revision 06
- CAMP 600.3; Primary Side Sampling Procedures: Hot Leg Liquid Sampling Depressurized Liquid; Revision 05
- Control Room Log Entries; July 2009 through June 2010
- EPG 1.1; Performance Indicator Guideline; Revision 6
- FG-E-MSPI-01; Mitigating System Performance Index; Revision 3
- Mitigating Systems Performance Index (MSPI) Basis Document Data For Point Beach Nuclear Plant; 14
- Mitigating Systems Performance Index Derivation Reports For Units 1 And 2; Residual Heat Removal System; June 2010.
- Mitigating Systems Performance Index Derivation Reports For Units 1 And 2; High Pressure Injection System; June 2010.
- Mitigating Systems Performance Index Derivation Reports For Units 1 And 2; Heat Removal System; June 2010.
- Mitigating Systems Performance Index Derivation Reports For Units 1 And 2; Emergency AC Power System; June 2010.
- Mitigating Systems Performance Index Derivation Reports for Units 1 And 2; Cooling Water Systems; June 2010.
- NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 6
- NP 3.2.2; Primary Water Chemistry Monitoring Program; Revision 20
- NP 5.2.16; NRC Performance Indicators; Revision 14
- NP 5.2.16; NRC Performance Indicators; Revision 15
- NRC Occupational Exposure Performance Indicator Data; March 2009 Through June 2010
- Point Beach Units 1 And 2; Second Quarter Of 2010 Performance Indicators

- Unit 1 And Unit 2 Mitigating System Performance Index Monthly Unavailability And Verification Data (Monthly); High Pressure Injection System; July 2009 Through June 2010
- Unit 1 And Unit 2 Mitigating System Performance Index Monthly Unavailability And Verification Data (Monthly); Heat Removal System; July 2009 Through June 2010
- Unit 1 And Unit 2 Mitigating System Performance Index Monthly Unavailability And Verification Data (Monthly); Emergency AC Power System; July 2009 Through June 2010.
- Unit 1 And Unit 2 Mitigating System Performance Index Monthly Unavailability And Verification Data (Monthly); Cooling Water Systems; July 2009 Through June 2010.
- Unit 1 And Unit 2 Mitigating System Performance Index Monthly Unavailability And Verification Data (Monthly); Residual Heat Removal System; July 2009 Through June 2010.

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

- AR01175644; Action Request Report; June 19, 2010
- AR01175650; Action Request Report; June 19, 2010
- AR01175719; Action Request Report; June 21, 2010
- AR01176098; Action Request Report; June 28, 2010
- AR01176472; Action Request Report; July 2, 2010
- AR01177018; Action Request Report; July 14, 2010
- AR01178563; Action Request Report; August 5, 2010
- AR01178684; Action Request Report; August 7, 2010
- CR00032759; TG-01-G Electrical Generator; July 28, 2010
- CR00032791; Z-008 TG-01 Turbine Generator Condition Monitor; July 28, 2010
- CR00032837; TG-01-G Electrical Generator; July 28, 2010
- CR00032841; HG-00004 W061 H_2 Gas Purity Blwr Supply from TG-01 TG Top; July 28, 2010
- Checklist 00031204; Isolate H_2 To ITG-01-G For Manway Cover And Misc. Repairs; July 28, 2010
- Daily Status Report Unit 1; July 27, 2010
- Daily Status Report Unit 2; July 27, 2010
- Daily Status Report Unit 2; July 29, 2010
- EN 46129; Point Beach Nuclear Plant Event Notification Worksheet; July 26, 2010
- Narrative Log Report; July 26, 2010 Through July 28, 2010
- Narrative Log Report; July 30, 2010
- NP 5.33; Incident Investigation And Post-Trip Review; Revision 9
- OI 32A; Pressurizing And Testing Main Generator With Air; Revision 4
- OP 1B Appendix A; Estimated Critical Position Calculation; Revision 14
- OP 1B; Reactor Startup; Revision 61
- OP 1B; Reactor Startup; Revision 61
- OP 1C; Procedure Record And Field Copy Tracking; Revision 18
- OP3A; Power Operation To Hot Standby Unit 1; Revision 3
- Open Prompt Operability Determinations List; July 26, 2010
- Operational Decision Making Beckwith Relay Failure Unit 1; July 30, 2010
- Operational Decision Making Generator Hydrogen Leakage Unit 1; July 15, 2010
- Operational Decision Making Generator Hydrogen Leakage Unit 1; July 14, 2010
- PBNP ICRR Plot; July 10, 2010
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- Safety Monitor Calculation Unit 1; July 26, 2010
- Safety Monitor Calculation Unit 1; July 26, 2010
- Safety Monitor Calculation Unit 2; July 26, 2010
- Safety Monitor Calculation Unit 2; July 28, 2010

- Unit 1 Forced Outage Work List Test Requirements; July 27, 2010
- Unit 1 Forced Outage Work List; July 26, 2010
- Unit 1 Forced Outage Work List; July 27, 2010

4OA5 Other Activities

- 10 CFR 50.72(g) File Index And Selected Files; August 17, 2010
- AR01173991; Quick Hit Self-Assessment; June 2010
- ATI Environmental, Inc. Midwest Laboratory; Monthly Progress Report; August 06, 2010
- CAMP 920; Groundwater Protection Sampling Procedure; Revision 04
- Communications / Notification Plan; Revision 03
- EV-AA-100; FPL Nuclear Fleet Ground Water Protection Program; Revision 01
- EV-AA-100-1000; FPL Nuclear Fleet Guideline; Ground Water Protection Program
- HPIP 3.58; Groundwater Sampling And Subsoil Tritium Sampling; Revision 16
- NEI 07-07; Industry Groundwater Protection Initiative; Final August 2007
- NP 3.4.7; Groundwater Protection Program (GWPP); Revision 03
- NP 3.4.9; Groundwater Protection Plan Technical basis; Revision 01
- Point Beach Nuclear Assurance Report; Environmental Monitoring; July 2009
- Point Beach Nuclear Plant Environmental Manual; Revision 22
- Point Beach Nuclear Plant Off-Site Dose Calculation Manual; Revision 18
- Point Beach Nuclear Plant Radiological Effluent Control Manual; Revision 05
- STS/AECOM; Preliminary Site Conceptual Model; FPL Energy, Point Beach, LLC; Project No. 200802277
- PBSA-ENG-10-16; Quick Hit Assessment Report; 06/10/10
- RCE 99-054; Steam Generator Blowdown System Pressure Transients; December 11, 2000
- AR1170322; Revise FSAR Figures 5.2-8 And 5.2-22; March 24, 2010
- AR1160370; NSAL-09-8 – Presence Of Vapor In ECCS; November 4, 2009
- AR1136576; Minor Gas Void In Unvented Section Of HHSI Discharge Line; October 1, 2008
- AR1136737; UT Inspection Of Void At 1-RH-D03; October 5, 2008
- AR1137890; UT Inspection Results For GL 2008-01; October 17, 2008
- AR1151927; SGBD Water Hammer--RCE99-054 Action Item Not Completed; June 24, 2009
- AR1159862; Acceptance Criteria For Gas Voids May Be Incomplete; October 28, 2009
- AR1160512; Post Modification UT Exams On RHR Piping Detected Gas Voids; November 5, 2009
- AR1167534; Non-Conservative Initial Condition In ECCS Venting Procedure; February 26, 2010
- AR1169919; Gas Void Detected During Post Modification UT Exams – RHR; March 19, 2010
- AR1175042; GL 2008-1 Self-Assessment GAP Venting Surveillance Scope; June 9, 2010
- AR1175403; PB2 Train A LHSI Gas Void UT Results; June 15, 2010
- AR1170237; Gas Voids Detected During Post Mod UT Exams – SI Train A; March 23, 2010
- Gas Accumulation Program Manual; June 26, 2009
- 1-TS-ECCS-002 Train A; Safeguards System Venting (Monthly) Unit 1-Train A; March 10, 2010
- 1-TS-ECCS-002 Train B; Safeguards System Venting (Monthly) Unit 1-Train B; March 10, 2010
- 2-TS-ECCS-002 Train A; Safeguards System Venting (Monthly) Unit 1-Train A; May 3, 2010
- 2-TS-ECCS-002 Train B; Safeguards System Venting (Monthly) Unit 1-Train B; May 3, 2010
- OI-136E; LHSI Core Deluge Venting Train A Inside Containment Unit 2; May 3, 2010
- OI-136; Fill And Vent The RHR System Unit 2; November 2, 2009
- NDE-115; Manual UT Examination Of Piping And Components For The Determination Of Fluid Levels; June 5, 2009

- WO376946; Train A Safeguards System Venting (Monthly) Unit 1; May 4, 2010
- WO378341; Train A Safeguards System Venting (Monthly) Unit 1; May 30, 2010
- WO376946; Train B Safeguards System Venting (Monthly) Unit 1; May 5, 2010
- WO378341; Train B Safeguards System Venting (Monthly) Unit 1; May 30, 2010
- WO377272; Train A Safeguards System Venting (Monthly) Unit 2; May 12, 2010
- WO378652; Train A Safeguards System Venting (Monthly) Unit 2; June 8, 2010
- WO377272; Train B Safeguards System Venting (Monthly) Unit 2; May 12, 2010
- WO378652; Train B Safeguards System Venting (Monthly) Unit 2; June 9, 2010
- WO387127; LHSI Core Deluge Venting Train A Inside Containment Unit 2; May 13, 2010
- WO388242; LHSI Core Deluge Venting Train A Inside Containment Unit 2; June 15, 2010
- WO356497; 1SI/Scoping Walkdown GL-08-01; October 31, 2008
- WO356577; 2SI/Scoping Walkdown/NDE GL-08-01; October 20, 2009
- WO375151; 2SI/Scoping Walkdown GL-08-01; October 4, 2009
- PB20081010-2; Point Beach Walkdown Closure Report; November 7, 2008
- PB20091116; Point Beach Walkdown Closure Report; November 16, 2009
- QF-0515A; Design Input Checklist; November 25, 2009
- PBNP-994-40-M02; ECCS Suction Piping Gas Void Calculation And Operability Determination; December 7, 2009
- NAI-1459-001; Comparison Of GOTHIC Gas Transport Calculations With Test Data; June 26, 2009
- CALC 2010-0006; Vent From ECCS Common Suction Piping Vents; February 25, 2010
- EC12669; Vent Valve Modification For ECCS Piping; February 11, 2008
- ECN12810; ECN; 12669-03 Install Vent On 12”-SI-151R-2 At High Point 2SI-S01; October 8, 2008
- ECN12823; ECN; 12669-05 Install Vent On 12”-SI-151R-2 At High Point 2SI-S02; October 8, 2008
- ECN12860; ECN 12669-05 Install Two Vent Lines On Line 6”-601R-2 At High Point 1RH-D02 And One At 1RH-D03; October 13, 2008
- ECN14157; Installation Of A Vent Line On Line 6”-ST-601R-2 At High Point IC-2-SI-D01; December 15, 2009
- ECN14163; Installation Of A Vent Line On Line 6”-SI-1501R-1 At High Point OC-2-SI-D01; December 14, 2009
- ECN14721; Installation Of A Vent Line On High Point OC-1-SI-S12 On Valve Body Of Valve 1SI-857B; December 11, 2009

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
AR	Action Request
CAP	Corrective Action Program
CCW	Component Cooling Water
CDF	Core Damage Frequency
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EPD	Electronic Personal Dosimeter
FEP	Fire Emergency Plan
FIN	Finding
FPER	Fire Protection Evaluation Report
FRV	Feedwater Regulating Valve
FSAR	Final Safety Analysis Report
GL	Generic Letter
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LOCA	Loss-of-Coolant-Accident
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulations
NSAL	Nuclear Safety Advisory Letter
P&ID	Piping and Instrumentation Diagram
PAB	Primary Auxiliary Building
PI	Performance Indicator
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMP	Routine Maintenance Procedure
RP	Radiation Protection
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
SI	Safety Injection
SRA	Senior Reactor Analyst
SSC	Structure, System, and Component
TI	Temporary Instruction
TS	Technical Specification
TSTF	Technical Specification Task Force
VCT	Volume Control Tank
WO	Work Order

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Sincerely,

/RA/ By John Jandovitz Acting For/

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

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

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Letter to L. Meyer from M. Kunowski dated November 9, 2010

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000266/2010004;
05000301/2010004

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