August 10, 2005

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

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000266/2005004;
05000301/2005004

Dear Mr. Koehl:

On June 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Point Beach Nuclear Plant, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on July 22, 2005, 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, five findings of very low safety significance were identified, four of which involved violations of NRC requirements. However, because these violations were of very low safety significance, not willful, and not repetitive, and because the issues were entered into your corrective action program, the NRC is treating these findings as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

In addition to the routine NRC inspection and assessment activities, Point Beach performance is being evaluated quarterly as described in the Annual Assessment Letter - Point Beach Nuclear Plant, dated March 4, 2004. Consistent with Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," plants in the multiple/repetitive degraded cornerstone column of the Action Matrix are given consideration at each quarterly performance assessment review for (1) declaring plant performance to be unacceptable in accordance with the guidance in IMC 0305; (2) transferring to the IMC 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems," process; and (3) taking additional regulatory actions, as appropriate. On July 6, 2005, the NRC reviewed Point Beach operational performance, inspection findings, and performance indicators. Based on this review, we concluded that Point Beach is operating safely.

We determined that no additional regulatory actions, beyond the already increased inspection activities and management oversight, are currently warranted.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Point Beach Nuclear Plant.

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

Sincerely,

/RA/

Mark A. Satorius, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000266/2005004; 05000301/2005004
w/Attachment: Supplemental Information

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REGION III

Docket Nos:	50-266; 50-301
License Nos:	DPR-24; DPR-27
Report No:	05000266/2005004; 05000301/2005004.
Licensee:	Nuclear Management Company, LLC.
Facility:	Point Beach Nuclear Plant, Units 1 and 2.
Location:	6610 Nuclear Road Two Rivers, WI 54241
Dates:	April 1, 2005, through June 30, 2005
Inspectors:	R. Krsek, Senior Resident Inspector M. Morris, Resident Inspector M. Kunowski, Project Engineer J. Giessner, Reactor Engineer C. Zoia, Project Engineer T. Tongue, Senior Project Engineer M. Salter-Williams, Resident Inspector, Davis Besse B. Dickson, Senior Resident Inspector, Clinton R. Winter, Reactor Engineer J. Neurauter, Reactor Engineer S. Orth, Health Physics Program Manager (TL)
Approved by:	P. Louden, Chief Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000266/2005004, 05000301/2005004; 04/01/2005 - 06/30/2005; Point Beach Nuclear Plant, Units 1 and 2; Non-Routine Evolutions, Operability Evaluations, Refueling and Outage Activities, Surveillance Testing, and Other Activities.

This report covers a 3-month period of baseline resident inspection for the Point Beach Nuclear Plant, Units 1 and 2, conducted by Region III and resident inspectors. The inservice inspection baseline (71111.08), radiation safety baseline (71121.01), reactor vessel head replacement (71007), and Temporary Instruction TI2515/163 inspections were conducted by the resident and Region III based inspectors. Five Green findings, four of which had associated NCVs were identified during this inspection period. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process.” Findings for which the Significance Determination Process (SDP) does not apply, may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Mitigating Systems

- Green. A Green finding associated with a Non-Cited Violation of Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed for the failure to verify the appropriate residual heat removal (RHR) system lineup prior to the issuance of a tagging order. As a result, upon implementation of the tagging order, the licensee also failed to maintain cooling for the Unit 2 reactor coolant system (RCS) in accordance with licensee procedures. Specifically, on April 19, 2005, the licensee performed a tagout on the ‘B’ train of safety injection while the ‘B’ RHR heat exchanger was in service and inadvertently isolated flow through the ‘B’ RHR heat exchanger, causing a loss of RHR for approximately 40 minutes.

  The inspectors determined that a primary cause of this finding was related to the cross-cutting area of Human Performance, because the licensee failed to verify the appropriate conditions were established for implementation of the tagout.

  The issue was more than minor because the finding was associated with the configuration control attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using IMC 0609, Appendix G, Phase 1 Screening, Checklist 4, “Pressurized Water Reactor (PWR) Refueling Operations: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer,” specifically, Section I.C, “Core Heat Removal Guidelines - Equipment,” was applicable to this finding. The finding affected the RHR loop which was operable and in operation; however, the finding did not meet the requirements for a Phase 2 or Phase 3 analysis per Appendix G. Therefore the finding was determined to be of very low significance. The licensee took prompt action to enter the item into the corrective action process, develop and implement interim corrective actions and evaluate the issues to develop additional corrective actions. (Section 1R14.1)

- Green. The inspectors identified a finding of very low significance (Green) for an adverse trend of failures to perform causal evaluations for conditions adverse to quality which only received operability recommendations, to ensure the cause of the conditions were identified and corrected. The licensee further evaluated the issue and corroborated the adverse trend, and in addition identified the issue potentially extended to condition reports documenting conditions adverse to quality with only maintenance rule evaluations performed. No violation of NRC requirements occurred.

  The inspectors also determined that the primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution, because the licensee failed to perform causal evaluations commensurate with the significance of the condition reports to ensure the conditions adverse to quality were identified and corrected.

  The issue was more than minor because the underlying issues associated with the finding were associated with the equipment performance and design control attributes of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening for the Mitigating Systems cornerstone and determined the finding was of very low significance. The licensee took action to enter the item into the corrective action process and develop interim corrective actions. At the end of the inspection period, the licensee had not completed the evaluation of the finding. (Section 1R151)

- Green. **A Green finding associated with a Non-Cited Violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when an inadvertent inventory loss from the Unit 2 refueling water storage tank** occurred. The inventory loss occurred when licensee personnel performed two procedures concurrently, which was not appropriate to the circumstances due to the equipment configuration conflicts created by performing the test procedures in this manner.

  The inspectors determined that the primary cause of this finding was related to the cross-cutting area of Human Performance, because the licensee failed to appropriately validate and verify the procedures could be performed concurrently.

  The issue was more than minor because the finding was associated with the configuration control and procedure quality attributes of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent core damage. The inspectors evaluated the finding using IMC 0609, Appendix G, Phase 1 Screening, Checklist 4, “PWR Refueling Operations: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer,” specifically Section II.C, “Inventory Control Guidelines-Equipment,” was applicable to this finding. The inspectors determined the finding affected equipment necessary for makeup to the refueling cavity; however, the finding did not meet the requirements for a Phase 2 or Phase 3 analysis per Appendix G. Therefore the finding was determined to be of very low significance. The licensee took prompt action to enter the item into the corrective action process, evaluate the issues and develop corrective actions to address the causes of this finding. (Section 1R22.1)

- Green. A Green finding associated with a Non-Cited Violation of Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors. Specifically, the licensee failed to apply design control measures to verify the adequacy of the design for the head assembly upgrade package (HAUP) associated with the replacement reactor vessel closure head. Specifically, design calculations that support the HAUP design basis contained errors, including the failure to specify the American Institute of Steel Construction (AISC) or American Society of Mechanical Engineers Boiler and Pressure Vessel Code minimum fillet weld size requirements, the failure to transform bolt design loads into the analysis bolt pattern coordinate system, and the failure to evaluate the control rod drive mechanism cooling duct as a slender component in accordance with Appendix B5 of the AISC design code.

  The finding was more than minor because if left uncorrected the finding could become a more significant safety concern. Specifically, failure to specify the AISC or American Society of Mechanical Engineers Code required minimum fillet weld size, or failure to transform bolt design loads into the analysis bolt pattern coordinate system, or failure to evaluate slender section components in accordance with AISC Appendix B5 in similar design calculations could result in modifications that exceed licensing basis design acceptance limits. The finding was of very low safety significance because the calculation errors in these instances did not result in an HAUP structure or component to exceed its design basis acceptance limit. The licensee took prompt action to enter the item into the corrective action process, evaluate the issues and develop corrective actions to address the causes of this finding. (Section 4OA5.1)

Cornerstone: Barrier Integrity

- Green. A Green finding associated with a Non-Cited Violation of Title 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to take adequate corrective action to preclude repetition of a significant condition adverse to quality was identified by the inspectors. Specifically, the licensee identified that the root cause of an April 9, 2004, potential loss of a hot leg vent path during nozzle dam installation, a failure to adequately identify, track and maintain licensee commitments to Generic Letter 88-17 in plant procedures, a significant condition adverse to quality. Prior to the start of the Unit 2 Refueling

Outage, the inspectors identified that the approved outage shutdown safety analysis contained an orange risk path, during which the licensee would have been unable to close the containment equipment hatch within the time to boil the water around the fuel. The licensee's root cause evaluation for this issue identified the root cause was the same as the April 2004 event; therefore, the licensee's corrective actions from the April 2004 event failed to preclude repetition of the identified cause. The licensee took prompt corrective action to remove these planned activities from the outage schedule to ensure the equipment hatch was closed when the RCS was breached; however, the licensee also identified in the root cause evaluation that this configuration actually occurred in the 1999 Unit 1 Refueling Outage.

The inspectors also determined that a primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution, because the licensee failed to take adequate corrective actions to preclude repetition of a significant condition adverse to quality.

The issue was more than minor because the finding was associated with preserving the containment boundary attribute of the Barrier Integrity cornerstone and affected the cornerstone objective of providing reasonable assurance that the physical design barriers (Containment) protect the public from radionuclide releases cause by accidents or events. The inspectors evaluated the finding using IMC 0609, Appendix G, Phase 1 Screening, Checklist 3, "PWR Cold Shutdown and Refueling Operation RCS Open and Refueling Cavity Level <23'," specifically Section IV, "Containment Control Guidelines." The finding dealt with the procedures and training to close containment prior to core boiling when the RCS was open. The finding did not meet any of the criteria requiring a Phase 2 or 3 Analysis per Appendix G, Checklist 3, specifically findings that degrade the ability of containment to remain intact following a severe accident. This was in part due to the type of RCS system breach which was scheduled. Therefore, the finding was determined to be of very low significance. The licensee took prompt action to enter the item into the corrective action process, evaluate the issues and develop corrective actions to address the causes of this finding to preclude repetition. (Section 1R20.1)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1 was at 100 percent power throughout the inspection period with the exception of brief downpowers during routine auxiliary feedwater (AFW) and secondary system valve testing.

Unit 2 began the inspection period at 100 percent power on April 1, 2005, and commenced a downpower to begin the Cycle 27 Refueling Outage (U2R27). Unit 2 remained in the refueling outage for the remainder of the inspection period.

1. REACTOR SAFETY

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

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

.1 Reactor Vessel Head (RVH) Replacement (71007)

a. Inspection Scope

The inspectors reviewed the licensee's evaluations of applicability determination and screening questions for the design changes associated with the Unit 2 RVH replacement. The inspectors assessed, for each change, whether the requirements of Title 10 CFR Part 50.59 were appropriately applied. Specifically, the inspectors reviewed Modification No. 03-056, which included a review of the function of each changed component, the change description and scope, the Title 10 CFR Part 50.59/72.48 screening, and Title 10 CFR Part 50.59 evaluation for the following items:

- Replacement RVH;
- Penetrations;
- Lifting lugs;
- Shroud support ring;
- Thermal sleeves/guide funnels;
- CRDM changes;
- Removal of four unused part length CRDMs;
- Core exit thermocouple penetration changes;
- Reactor coolant gas vent system (RCGVS) changes; and
- Reactor vessel level indication system (RVLIS) changes.

The inspectors also reviewed the Title 10 CFR Part 50.59/72.48 screening associated with Modification Nos. 03-057, 03-058, 03-059, 03-060, and 03-061 for the following items:

- Replace RVH insulation;
- Head assembly upgrade package (HAUP);
- Unit 2 containment equipment hatch shield wall modification;

- Analog rod position indicator system cable and connector modification; and
- CRDM cable and connector modification.

The inspectors used, in part, Nuclear Energy Institute 96-07, "Guidelines for Title 10 CFR Part 50.59 Implementation," to determine acceptability of the completed pre-screenings and screenings. This document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of Title 10 CFR Part 50.59, Changes, Tests, and Experiments." The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

This review constituted one inspection sample.

b. Findings

Introduction: The inspectors identified an Unresolved Item (URI) for issues raised regarding the RVH drop analysis results submitted by the licensee to the NRC in 1982. The inspectors noted that the licensee's design modifications and regulatory evaluations for the RVH did not address the increased weight of the new RVH nor the RVH drop analysis of 1982.

Description: The combined weight of the new HAUP and replacement RVH was approximately 27,000 pounds heavier than the original weight of the RVH and 17,000 pounds heavier than the current weight of the RVH (lead shielding was added to the original RVH). In March 2005, as part of the RVH replacement inspection, the inspectors requested licensee documentation related to the RVH replacement, including any applicable RVH drop analysis. The licensee initially indicated that a RVH drop analysis was not part of the facility's license basis; however, the inspectors further questioned this interpretation based on docketed licensee correspondence dated February 25, 1982, which committed to provide the NRC with a RVH drop analysis in the fall of 1982.

Further discussion took place with the licensee, inspectors and Office of Nuclear Reactor Regulation Project Manager. The Project Manager and inspectors discussed with the licensee that since the 1982 RVH drop analysis was performed in response to a request from the NRC as detailed in NRC Generic Letter (GL) 80-113 dated December 20, 1980, Title 10 CFR Part 50.71(e) required that the results of the evaluation be incorporated into the Final Safety Analysis Report (FSAR). As a result, the licensee initiated corrective action program document CAP063450, "NUREG-0612 Information Not Fully Incorporated into FSAR."

The licensee later provided the inspectors with a docketed submittal to the NRC dated November 22, 1982, from the Point Beach Nuclear Plant (PBNP) which contained the results of a RVH drop analysis. This letter stated that:

> "The results of this analysis show that upon impact of the head drop the initial reactor vessel nozzle stresses are well within allowables. However, the loads imposed upon the reactor vessel supports caused by the impact of the head are greater than the critical buckling load of the support columns. These supports cannot be relied upon to absorb enough of the

energy of impact to prevent severe damage to the safety injection (SI) lines attached to the reactor vessel or to the primary coolant loop piping."

The licensee also provided the inspectors a November 15, 1982, Westinghouse Electric Company letter to Wisconsin Electric Power Company, which contained a summary of the results for a reactor head drop analysis. The analysis concluded that a head drop would likely result in the buckling of the reactor support columns, and the potential loss of reactor piping connections which would prevent removal of decay heat from the core.

As a result of a review of the technical aspects of Condition Report CAP063450 and the docketed analysis, the licensee initiated Condition Report CAP063536, "Unable to Meet NUREG-0612 Phase II Requirements for Head Drop Analyses." As an immediate corrective action the licensee placed an administrative hold on the movement of any RVH over irradiated fuel.

In a letter dated April 12, 2005, to the NRC, the licensee discussed an understanding of the licensing basis associated with the 1982 RVH drop analysis. In subsequent discussions with the licensee, the Headquarters NRC staff informed the licensee that the April 12, 2005, letter did not properly characterize the PBNP licensing basis related to the 1982 RVH drop analysis.

Subsequently, the licensee completed a Title 10 CFR Part 50.59, "Changes, tests, and experiments," review of the proposed incorporation of the 1982 RVH drop analysis into the FSAR. This review concluded that the proposed change to the FSAR required prior NRC approval in accordance with the requirements of Title 10 CFR Part 50.59(c)(2)(v), because the change created the possibility for an accident of a different type than any previously evaluated in the FSAR. On April 29, 2005, the licensee submitted an application for a proposed amendment, which was supplemented by letters dated May 13, May 19, June 1, June 4, June 9, June 20, and June 23, 2005. On June 24, 2005, the Office of Nuclear Reactor Regulation issued a Safety Evaluation for the proposed license amendment which incorporated a Unit 2 RVH drop accident into the PBNP FSAR.

On July 24, 2005, the licensee submitted and requested for NRC review and approval a proposed amendment for Review of Heavy Load Analysis for Units 1 and 2. Therefore, this issue will be considered an Unresolved Item pending further NRC review of the licensee's evaluations and proposed amendment (URI 05000266/2005004-01; 05000301/2005004-01).

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of risk-significant systems to determine the operability of the systems. The inspectors utilized system valve lineups and electrical breaker checklists (CLs), tank level books, plant drawings, and selected operating procedures to determine if the systems were correctly aligned to perform the intended design functions. The inspectors also examined the material condition of the components and observed operating equipment parameters to determine if there were deficiencies. The inspectors reviewed completed work orders and calibration records associated with the systems, for issues that could affect component or train functions. The inspectors used the information in the appropriate sections of the FSAR to determine the functional requirements of the systems. Partial system walkdowns of the following systems constituted two inspection procedure samples:

- Spent Fuel Pool System, and
- Unit 2 Residual Heat Removal (RHR) System.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Walkdowns of Selected Fire Zones

a. Inspection Scope

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

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

- FZ 611, Fire Area A46, Unit 2 Containment 21' Elevation;
- FZ 615, Fire Area A46, Unit 2 Containment 46' Elevation;
- FZ 583, Fire Area A01-E, Unit 2 Turbine Building General Area - 8 ft;
- FZ 681, Fire Area A01-F, Gas Turbine - G05;
- FZ 552, Fire Area A38, Service Water (SW) Pump Room;
- FZ 156, Fire Area A06, 1B32 Motor Control Center (MCC) Area;
- FZ 166, Fire Area A15, 2B32 MCC Area; and
- Unit 2 Containment Fire Detector Walkdown.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary, risk-significant piping system boundaries, and the containment boundary. Steam generator and reactor vessel upper head penetration inspections were not conducted by the licensee during this outage. The ISI activities constituted two inspection procedure samples:

- Inspection Activities Other Than Steam Generator Tube Inspections, PWR Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control (BACC)

  The inspectors conducted direct observations and/or records review of the following nondestructive examination activities to evaluate compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and to verify that indications and defects were dispositioned in accordance with the ASME Code or a Nuclear Regulatory Commission (NRC) approved alternative:

  1. Visual examination (VT-2) of the reactor vessel bottom mounted instrumentation penetrations;

  2. Ultrasonic examination of the Unit 2 reactor pressure vessel shell to flange weld, Weld RPV-14-683;

  3. Ultrasonic examination of a Unit 2 elbow to Valve 2SI-853D weld, Weld AC-06-SI-2001-18; and

  4. Ultrasonic examination of a Unit 2 regenerative heat exchanger nozzle inside radius weld, Weld RHE-N4-IRS.

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

  Remote visual examination (VT-1) and liquid penetrant examination of Feedwater Weld FW-16-FW-2001-28BC identified one linear and several rounded indications (Indication Disposition Report 2003-0008). Light grinding of the indications followed by another visual and liquid penetrant examination removed the linear indication and resulted in the remaining rounded indications to be acceptable using Table IWB-3414-4 as guidance.

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

1. Replacement of ISI Class 1 SI Pump P-15A to reactor coolant Loop 'A' Cold Leg SI Check Valve SI-00845A; and

2. Remove and replace ISI Class 2 refueling water storage tank (RWST) level indicating equalizing Line SI-0151R to facilitate replacement of RWST immersion heater.

- BACC Inspection Activities

Following shutdown, the inspectors reviewed a sample of BACC walkdown visual examination activities through direct observation. This walkdown was completed with Unit 2 in Mode 3 and included the lower containment building inner volume and annulus. The inspectors determined whether the visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components.

The inspectors performed a review of ISI and boric acid-related issues that were identified by the licensee and entered into the corrective action program. The review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The engineering evaluation for a boric acid leak identified on the manifold of narrow range Level Transmitter 2LT-497 was also reviewed. The inspectors performed these reviews to ensure compliance with the ASME Code and Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Resident Inspector Quarterly Observation of Licensed Operator Requalification

a. Inspection Scope

During Licensed Operator Requalification Cycle 05-03, the inspectors observed operating crew performance during a simulator as-found requalification examination. The inspectors also reviewed some of the changes to the simulator model against modifications made in the plant. Observation of the requalification quarterly evaluation constituted one inspection procedure sample.

The inspectors assessed crew performance in the following areas:

- Clarity and formality of communications;
- Understanding of the interactions and function of the operating crew during an emergency;
- Prioritization, interpretation, and verification of actions required for emergency procedure use and interpretation;
- Oversight and direction from supervisors; and
- Group dynamics.

Crew performance in these areas was also compared to licensee management expectations and guidelines, as presented in Nuclear Plant Procedures Manual Procedure (NP) 2.1.1, “Conduct of Operations.”

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

For each system reviewed, the inspectors reviewed significant work orders and CAPs to determine if failures were appropriately identified, classified, and corrected, and if unavailable time was correctly calculated. The reviews of maintenance effectiveness for the following components and systems constituted four inspection procedure samples:

- Unit 1, 4160-Volt Electrical System;
- Units 1 and 2, Chemical and Volume Control System;
- Emergency Diesel Generators; and
- Units 1 and 2, RHR System.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

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

- Planned and emergent maintenance during the week of April 11, 2005;
- Planned and emergent maintenance during the week of April 25, 2005;
- Planned and emergent maintenance during the week of May 16, 2005;
- Planned and emergent maintenance during the week of May 23, 2005;
- Planned and emergent maintenance during the week of June 6, 2005; and
- Planned and emergent maintenance during the week of June 13, 2005.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Non-Routine Plant Evolutions and Events (71111.14)

.1 Loss of RHR During the Unit 2 Outage

a. Inspection Scope

On April 19, 2005, the inspectors observed operator response to a loss of RHR flow during a tagging evolution for SI Train 'B' during the Unit 2 refueling outage. The inspectors reviewed the procedures used during the event and the documents used during the licensee event investigation. The inspectors also assessed the adequacy of the immediate corrective actions and the results of the initial investigation documented in Root Cause Evaluation (RCE) RCE-278 completed on May 23, 2005. This observation constituted one inspection procedure sample.

b. Findings

Introduction: A Green finding associated with a Non-Cited Violation (NCV) of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and

Drawings," was self-revealed for the failure to verify the appropriate RHR system lineup prior to the issuance of a tagging order. As a result, upon implementation of the tagging order, the licensee also failed to maintain cooling for the RCS in accordance with licensee procedures. Specifically, on April 19, 2005, the licensee performed a tagout on the 'B' train of SI while the 'B' RHR heat exchanger was in service and inadvertently isolated flow through the 'B' RHR heat exchanger, causing a loss of RHR for approximately 40 minutes.

Description: On April 19, 2005, Unit 2 was in a refueling outage with the RVH removed, the refueling cavity full, and the time to boil for the fuel greater than 20 hours. The method of heat removal was via the RHR system with the 'A' RHR pump and 'B' RHR heat exchanger in service. The work control center issued a tagout which removed the 'B' SI train from service for maintenance. The tag series included actions to close and tag the 'B' RHR Heat Exchanger Inlet Valve 2RH-715B. Auxiliary operators initially questioned whether the performance of the tagout would affect the RHR system. However, when the tagout was hung, the 'B' RHR heat exchanger was isolated from the 'A' RHR pump.

A RHR low flow indication was noticed in the control room followed by the RHR low flow alarm. At 5:09 a.m., the operators stopped the 'A' RHR pump and entered Shutdown Emergency Procedure (SEP) SEP-1, "Degraded RHR Flow Conditions." An initial attempt to restart the RHR pump indicated no RHR flow. The work control center personnel informed the control room that the 'B' SI train was being tagged out and that the line-up needed to be verified. At 5:19 a.m., the operators then aligned the 'A' pump with the 'A' heat exchanger and re-started the 'A' RHR pump. Residual heat removal flow was re-established; however, component cooling water (CCW) flow was not established to the 'A' RHR heat exchanger until 5:41 a.m.

The licensee initiated a condition report and performed a RCE along with an extent of condition evaluation. Prior to the licensee's completion of RCE-278 however, another tagging incident occurred which was incorporated into the RCE, due to a similar failure to verify the current plant configuration prior to tagging implementation. On April 25, 2005, the licensee was performing a tagout of the Unit 1 SW system to the Unit 1 CCW heat exchanger and inadvertently caused a disruption of SW flow to the CCW heat exchanger. The CCW temperature increase was quickly identified in the control room and the system alignment was rapidly returned to normal. The RCE for these issues identified two root causes. The first was a weakness in the tagging program associated with scheduling activities and system alignment controls, the second was a lack of supervisory oversight on the part of the Work Control Center Tagging SRO. Both root causes stemmed from a failure to use conduct of operations human performance tools, specifically: validation and verification, communications, awareness of plant conditions, and effective pre-job briefing.

Analysis: The inspectors determined that the failure to verify the appropriate system lineup prior to performing the tagout in accordance with NP 1.9.15, "Tagging Procedure," was a licensee performance deficiency warranting a significance evaluation.

The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on May 19, 2005, in that, the finding was associated with the configuration control attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. In addition, if left uncorrected, the finding would become a more significant safety concern.

The inspectors evaluated the finding using IMC 0609, Appendix G, Phase 1 Screening, CL 4, “PWR Refueling Operations: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer,” specifically Section I.C, “Core Heat Removal Guidelines - Equipment,” was applicable to this finding. The finding affected the RHR loop which was operable and in operation; however, the finding did not meet the requirements for a Phase 2 or Phase 3 analysis per Appendix G because the finding did not: increase the likelihood of a loss of RCS inventory; degrade the licensee’s ability to terminate a leak path or add RCS inventory; or degrade the licensee’s ability to recover decay heat removal once it was lost. The inspectors also determined that the finding was of very low safety significance because the SI system was available as a standby injection source and the recovery time for the RHR system was minimal compared to the time to boil for the fuel. Therefore, the finding was considered to be of very low significance (Green).

The inspectors also determined that a primary cause of this finding was related to the cross-cutting area of human performance, because the licensee failed to ensure that the appropriate conditions were established prior to implementation of a tag out for the ‘B’ SI Train.

Enforcement: Title 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” requires, in part, that the licensee accomplish activities affecting quality in accordance with instructions and procedures. Contrary to the above, the licensee did not maintain RHR cooling flow per procedure RP-1A, “Preparation for Refueling,” when RHR flow to the refueling cavity was isolated. In addition, the licensee implemented a tagout on the RHR valves which isolated RHR flow, without appropriately verifying the RHR pump and heat exchanger lineup in service, in accordance with procedure NP 1.9.15, “Tagging Procedure.” Therefore, the inspectors determined this finding was a violation of Criteria V, “Instructions, Procedures, and Drawings.” Because this violation was of very low safety significance, non-willful, non-repetitive, and documented in the licensee’s corrective action program as CAP063860, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000301/2005004-02).

The licensee took immediate interim corrective actions which included: the Operations Manager issuance of additional instructions for Danger Tagging; issuance of a pre-job briefing CL specifically for tagouts; issuance of a tagout preparation flow chart CL along with an operational impact statement; issuance of a tagout walkdown CL; and operators were required to protect, through postings, all equipment critical to maintain shutdown cooling. In addition, the licensee completed RCE-278 which developed additional corrective actions which included, but were not limited to: revision of procedure NP 1.9.15 to ensure tag series development received adequate evaluation/impact assessment against system operations; revision of work control management process Nuclear Procedure NP 10.2.4 to include the requirement of an impact statement for all work; and improvements to outage shift turnover practices which ensured that all shift, work control center and relief crew personnel received appropriate information on plant status.

1R15 Operability Evaluations (OPRs) (71111.15)

a. Inspection Scope

The inspectors reviewed selected OPRs associated with issues entered into the licensee's corrective action system. The inspectors reviewed design basis information, the FSAR, Technical Specifications (TS) requirements, and licensee procedures to determine the technical adequacy of the OPRs. In addition, the inspectors determined if compensatory measures were implemented, as required. The inspectors assessed whether system operability was properly justified and that the system remained available, such that no unrecognized increase in risk occurred. The reviews of the following OPRs constituted 15 inspection procedure samples:

- OPR 127; Local control using abnormal operating procedure (AOP) AOP-10A could result in pressurizer indicated level dropping off-scale in the event of a control room fire;
- OPR 125; Formal Documentation of Battery Charger TS Requirement change;
- OPR 132; Unanticipated Load on G-04;
- OPR 123; D-07, -08,-09, Westinghouse Battery Chargers;
- OPR 128; Performance of 2P-15A Shows Declining Trend;
- OPR 129; Equipment Hatch Shadow Shield Not 3 Feet Thick;
- OPR 131; 2P-2C, IT-22 Reduction on Flow;
- OPR 135; Containment Equipment Hatch Monorail Turnbuckles are not Loosened;
- OPR 136; Temperature Discrepancy in RR MR 02-018-2;
- OPR 137; As-found Weight of Installed SW Valve Greater Than Analyzed;
- OPR 138; Power Supply Credited for Appendix R Fire Area;
- Prompt Operability for 2P-2C Seal Leak Versus Emergency Core Cooling System;
- Prompt Operability for Unit 1 Fire Protection Area for Charging Pumps;
- Prompt Operability for G-03 and G-04 Kiene Valves; **and**
- **Prompt Operability for Evaluations Associated With Unit 1 CCW Relief Valves.**

b. Findings

Introduction: The inspectors identified a finding of very low significance (Green) for an adverse trend of failures to perform causal evaluations for conditions adverse to quality which received operability recommendations, to ensure the cause of the conditions were identified and corrected. No violation of NRC requirements occurred.

Description: The inspectors reviewed the corrective actions and causal evaluations associated with the initiating CAPs for operability recommendations. The inspectors noted during the review of CAPs which had operability recommendations performed in accordance with Generic Letter 91-18, that a majority of the CAPs had not received any type of causal evaluation, only an initial operability recommendation. Specifically, the inspectors noted that the 15 most recent CAPs which had operability recommendations performed, did not have any type of causal evaluation done to determine why the issue may have occurred and potentially correct the cause. The inspectors noted that for this sampling of CAPs, the conditions adverse to quality were generically treated as broke-fix type issues, without having a causal evaluation performed commensurate with the significance of the issue.

The inspectors acknowledged that treating CAPs with operability recommendations as broke-fix type issues was appropriate in some cases; however, the inspectors noted that five of the 15 CAPs were significance level B (licensee's second highest CAP significance level) and none of these CAPs had causal evaluations performed. The inspectors also noted that licensee's guidelines for implementing the corrective action program recommended performance of an apparent cause evaluation associated with an extent of condition for significance level B type issues. The inspectors shared several of these issues with specific examples of potential corrective actions not taken as a result of not performing causal evaluations, which included, but were not limited to:

- CAP064738 documented incorrect Engineered Safeguards Features Actuation Setpoint Values in TS. An operability recommendation was performed and corrective actions were developed to create an action plan to revise the Technical Specification values and compare other values in Section 3.3 of the TS. However, no causal evaluation was performed to determine why the incorrect setpoints were placed in TS and to potentially evaluate the extent of condition beyond Section 3.3 of the TS;
- CAP063871 documented a potential temperature discrepancy in a Relief Request submitted to the NRC for the Unit 1 RVH. An action plan was developed to correct the inaccurate information and submit the correct information to the NRC. However, no causal evaluation was performed to determine why the issue occurred so that corrective actions could be taken to address the submittal of incorrect information to the NRC;
- CAP063202 documented a reduction in flow of Charging Pump 2P2C during testing. Corrective actions were developed to troubleshoot, correct and track completion of the issue during the Refueling Outage. However, no causal evaluation was performed to determine why the reduction in flow occurred or to assess the potential extent of condition associated with this particular issue; and
- CAP61420 documented questionable resistance readings during Unit 2 Train B 480-Volt Relay Testing. Corrective actions were developed to troubleshoot the relay. However, no causal evaluation was performed to determine why the questionable readings were obtained or to assess the potential extent of condition associated with this particular issue.

The regulatory affairs and performance assessment departments further assessed the issues raised by the inspectors and corroborated the adverse trend through further analysis of additional CAPs associated with operability recommendations. The licensee initially reviewed the past 25 CAPs associated with operability recommendations and noted 12 CAPs were assigned a significance level B, and none had causal evaluations performed. Consequently, the licensee initiated CAP065028 to document the apparent adverse trend of failing to perform causal evaluations for CAPs which had received operability recommendations.

Analysis: The inspectors determined that the failure to perform causal evaluations commensurate with the significance of conditions adverse to quality to ensure the cause of the conditions were identified and corrected was a performance deficiency which warranted a significance determination.

The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” issued on May 19, 2005, in that, the finding was associated with the equipment performance and design control attributes of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. In addition, if left uncorrected, the finding would become a more significant safety concern.

Using IMC 0609, Appendix A, "SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barrier Integrity] Cornerstones," the inspectors determined that only the Mitigating Systems cornerstone was affected for the specific examples of failing to perform causal evaluations to ensure the cause of the conditions were corrected. The inspectors determined the finding was of very low safety significance because the finding was not a design or qualification deficiency, the finding did not represent an actual loss of safety function, and the finding did not screen as potentially risk significant due to a seismic, flooding or severe weather initiating event. Therefore, the finding was considered to be of very low safety significance (Green).

The inspectors also determined that a primary cause of this finding was related to the cross-cutting area of problem identification and resolution, because the licensee failed to perform causal evaluations for conditions adverse to quality which had received operability recommendations and also warranted causal evaluations.

Enforcement: No violation of regulatory requirements occurred. This performance deficiency was considered a finding of very low significance (FIN 05000266/2005004-03; 05000301/2005004-03).

The licensee entered this issue into the corrective action program as CAP065028. The licensee had not completed the apparent cause evaluation at the end of the inspection, therefore finalized corrective actions had not been developed. However, licensee staff indicated at the exit meeting that the issue was also found to extend to CAPs which received only maintenance rule evaluations, and that corrective actions would include briefings of the screening team members, as well as process improvements. In addition, the licensee staff indicated that a number of issues would require the performance of causal evaluations, to ensure the appropriate corrective actions were identified and implemented to address the conditions adverse to quality.

1R16 Operator Workarounds (71111.16)

.1 Semiannual Review of Operator Workarounds

a. Inspection Scope

The inspectors completed a semi-annual review of the cumulative effects of operator workarounds. The inspectors verified that the workarounds did not have a significant effect on the reliability, availability, or the ability to correctly operate mitigating systems. The inspectors also assessed whether the workarounds would significantly increase operator response time to transients and accidents. The inspectors reviewed the licensee's plans and schedules established to correct the conditions within a reasonable period of time. In addition to operator workarounds, the inspectors reviewed OPRs, operator challenges and burdens, and temporary modifications for cumulative effects. This review constituted one inspection procedure sample.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Selected Post-Maintenance Test Reviews

a. Inspection Scope

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

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

During this inspection period, the inspectors reviewed the following post maintenance activities, which constituted 10 quarterly inspection procedure samples:

- Relay 271 Testing Conducted per Procedure ORT-3B;
- Transformer 2X-04 Incoming Breaker;
- Control Room Emergency Filtration System Damper;
- Busses 2B41 and 2B42 Bucket Replacements;
- Bus 2B04, Relay MG-6 Replacement;
- Instructions RHP-1791 and RHP-1792A for RVH Lift Hold for Levelness;
- Relay 2SI-21X Replacement;
- Procedure IT-205 Pressurizer Power Operated Relief Valves and Block Valves;
- Accumulator 2LT-938 Calibration; and
- Valve 2SI-862G Post ORT-59 Re-Test.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Unacceptable Orange Risk Path Approved in Unit 2 Refueling Outage Schedule

a. Inspection Scope

During the week of March 27, 2005, prior to the start of the Unit 2 Refueling Outage, the inspectors reviewed the Plant Operations Review Committee approved Unit 2 Refueling Outage 27 Shutdown Safety Analysis. The inspectors compared the licensee's shutdown safety analysis against the planned equipment out-of-service and reviewed the requirements for containment closure contained in the licensee's procedures. The inspectors also reviewed the licensee's implementation of commitments to Generic Letter 88-17, "Loss of Decay Heat Removal."

b. Findings

Introduction: The inspectors identified a Green Finding with an associated NCV of Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to take adequate corrective action to preclude repetition of a significant condition adverse to quality. Specifically, the licensee identified the root cause of an April 9, 2004, potential loss of a hot leg vent path during nozzle dam installation as a failure to adequately identify, track and maintain licensee commitments to Generic Letter 88-17 in plant procedures, a significant condition adverse to quality. Prior to the start of the Unit 2 Refueling Outage, the inspectors identified that the approved outage shutdown safety analysis contained an orange risk path, during which time the containment equipment hatch would have been unable to be closed within the time to boil of the fuel. The licensee's RCE for this issue identified the root cause was the same as the April 2004 event; therefore, the licensee's corrective actions from the April 2004 event failed to preclude repetition of the identified cause.

Description: On March 25, 2005, during a review of the Unit 2 Refueling Outage, U2R27, Shutdown Safety Analysis, the inspectors noted that the outage schedule contained an orange risk path associated with the reactor coolant pump (RCP) motor removal with the containment equipment hatch removed. However, the approved orange path was in direct conflict with CL-1E, “Containment Closure Checklist,” which implemented the licensee’s Generic Letter 88-17 requirements for containment closure from the Nuclear Management and Resource Council NUMARC 91-06 document, “Guidelines for Industry Actions to Assess Shutdown Management.” Checklist CL-1E specified that, when the RCS was not intact and fuel was in the core, the ability to close the equipment hatch prior to time to core boiling was to be maintained.

The inspectors identified to licensee staff that this requirement could not be met in the orange path configuration specified in the approved outage schedule and shutdown safety analysis. In addition, the inspectors noted that the contingency actions did not appear to be adequate. Licensee personnel highlighted to the inspectors that a procedure change request had been approved, but was not yet implemented, to remove those requirements from CL-1E. In fact, licensee personnel processed the procedure change to remove the requirements, following the inspectors questions.

However, the licensee also initiated a CAP, which licensee management assigned a significance level ‘A’, and required a RCE to be performed. In addition, licensee outage management, immediately revised the outage schedule. The licensee changed the schedule so that the motor removal did not occur while the equipment hatch was removed, thereby removing the orange path in the schedule. The licensee determined in RCE-276, that, “The root cause of this event is the failure to adequately translate the requirements of Generic Letter 88-17 and Nuclear Management and Resource Council NUMARC 91-06 for Containment Closure, as well as, the PBNP commitments to them, into the procedures and contingency plans implementing them.” In addition, the RCE team identified that in 1999, during a Unit 1 Refueling Outage, this actual condition occurred, whereby the RCS was breached without the ability to close the equipment hatch within the time to boil.

The inspectors and licensee noted that the root cause of this event was the same root cause identified by the licensee following an April 9, 2004, incident (CAP055538) in which workers were allowed to proceed with the installation of hot leg nozzle dams prior to establishing a hot leg vent path (NRC Inspection Report 2004003). The licensee determined in RCE-254 for this event, that, “The Root cause of this event was Point Beach’s inadequate response, identification, tracking, and maintenance of the actions taken in response to the expeditious actions in Generic Letter 88-17.”

In reviewing the corrective actions to prevent recurrence for the April 2004 the inspectors determined the licensee’s corrective actions failed to prevent recurrence of the root cause of the event, as evidenced by the root cause for RCE-276. The inspectors also noted that during the inspection period, the licensee identified an additional failure to comply with Generic Letter 88-17 requirements in CAP063583, “Change of Dedicated Thermocouple Without Updating Plant Process Computer System Configuration.” This CAP documented that for the first half of the refueling outage, the thermocouples utilized to monitor RCS temperatures, were not on separate channels, as required, due a procedure change made prior to the refueling outage.

Finally, the inspectors noted that the licensee had corrective actions in place to ensure the rail system in containment for the replacement RVH project did not interfere with the licensee's ability to close the containment equipment hatch, which addressed recent industry operating experience.

Analysis: The inspectors determined that the failure to take adequate corrective actions to preclude repetition of a significant condition adverse to quality, was a performance deficiency warranting a significance evaluation.

The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on May 19, 2005, in that, the finding was associated with preserving the containment boundary attribute of the Barrier Integrity cornerstone and affected the cornerstone objective of providing reasonable assurance that the physical design barriers (Containment) protect the public from radio-nuclide releases cause by accidents or events. In addition, if left uncorrected, the finding would have become a more significant safety concern.

The inspectors evaluated the finding using IMC 0609, Appendix G, Phase 1 Screening, CL 3, "PWR Cold Shutdown and Refueling Operation RCS Open and Refueling Cavity Level <23'," specifically Section IV, "Containment Control Guidelines." The finding dealt with the procedures and training to close containment prior to core boiling when the RCS was open. The finding did not meet any of the criteria requiring a Phase 2 or 3 Analysis per Appendix G, CL 3, specifically findings that degrade the ability of containment to remain intact following a severe accident. This was in part due to the type of RCS breach scheduled, the lowering of a pump seal for RCP motor removal. In this scenario, upon a postulated loss of decay heat removal and the start of boiling in the RCS, the system configuration was such that the RCP impeller would lift due to the system pressure and seal the RCS leak. The licensee evaluated that this would occur prior to a loss of inventory that would challenge the ability to remove decay heat via natural circulation and impede the ability to close the equipment hatch. Therefore, the finding was considered to be of very low significance (Green).

The inspectors also determined that a primary cause of this finding was related to the cross-cutting the area of problem identification and resolution, because the licensee failed to take adequate corrective actions to preclude repetition of a significant condition adverse to quality.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that conditions adverse to quality are promptly identified and corrected, and in the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to this, the licensee failed to assure that corrective actions were taken to preclude repetition of the failure to identify, track and maintain the actions taken in response to Generic Letter 88-17, a significant condition adverse to quality identified in April 2004. Specifically, the inspectors identified that the approved shutdown safety assessment for the Unit 2 Spring 2005 Refueling Outage contained an orange path which was in conflict with the licensee's actions taken in response to Generic Letter 88-17. Therefore, the corrective actions taken in June 2004 that resulted from RCE-254 failed to preclude repetition of the cause of the condition, and the inspectors determined this finding was a violation of Criterion XVI, “Corrective Action.” Because this violation was of very low safety significance, non-willful, non-repetitive, and documented in the licensee’s corrective action program as CAP63088, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000266/2005004-04; 05000301/2005004-04).

As stated previously, the licensee took prompt corrective action to change the outage schedule prior to the occurrence of the previously approved orange path. In addition, the licensee performed a RCE and developed several additional corrective actions, which included, but were not limited to: revision to CL1E to specify the appropriate requirements; revision to site procedures for developing safe shutdown analysis to incorporate all pertinent commitments from Generic Letter 88-17; development of a site commitment list that was cross referenced to existing procedures and maintained; perform a review of the Fall 2005 Unit 1 Outage to ensure Generic Letter 88-17 commitments were met for scheduled activities; and perform effectiveness reviews of the planned corrective actions.

.2 Routine Refueling Outage Inspection Activities

a. Inspection Scope

The inspectors observed activities during the Unit 2 Cycle 27, U2R27, Refueling Outage conducted between April 2 and July 11, 2005. These inspection activities constituted one refueling outage inspection sample.

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

The inspectors conducted in-plant observations of the following daily outage activities:

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

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

- Performed Mode 3 walkdowns at the start and end of the refueling outage to check for active boric acid leak indications;
- Observed the control room staff perform the Unit 2 shutdown and initial cooldown;
- Verified that RCS cooldown rates were within Technical Specification limits;
- Observed control room staff operations during reduced inventory conditions;
- Observed core unloading activities in the containment, SFP, and control room;
- Observed core reload from the control room;
- Observed operators align the RHR system for shutdown cooling;
- Observed placement of the over-pressure protection system into operation;
- Monitored a pre-job briefing for fuel handling evolutions;
- Performed walkdowns of the auxiliary building to verify the placement of clearance orders on Unit 2 electrical buses, RHR systems, and SW systems;
- Observed lifting and transport of the RVH in preparation for core offload;
- RVH head replacement;
- Performed a walkdown of the control room and turbine building to verify safety-related electrical alignments following battery charger and 4160-Volt electrical bus routine maintenance;
- Performed a closeout inspection of the Unit 2 containment including a review of the results of the emergency core cooling sump inspection that had been performed earlier by the licensee. As part of this inspection, the inspectors also assessed whether all discrepancies noted during the walkdown were recorded and corrected;
- Reviewed shutdown margin calculations;
- Reviewed SFP cooling and SW pump configurations during partial core offload;
- Reviewed reduced inventory level RCS transmitter configurations;
- Reviewed the proper alignment and operation of the potential-dilution-in-progress alarm;
- Reviewed the evaluation of the fuel handling bridges in containment and the SFP;
- Reviewed Mode change CLs to verify that selected requirements were met while transitioning from the refueling Mode to full power operations;
- Observed portions of low power physics testing and approach to criticality; and
- Observed portions of the plant ascension to full power operations.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Unit 2 Containment Spray Sequence Test

a. Inspection Scope

The inspectors observed the performance of Procedure ORT-6, “Containment Spray Sequence Test Unit 2.” The inspectors attended the pre-job brief and observed the test.

In addition, the inspectors reviewed the test procedures and completed test records to determine if:

- Effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- Acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- Plant equipment calibration was correct, accurate, properly documented, as-left setpoints were within required ranges, and the calibration frequency was in accordance with TS, the FSAR, procedures, and applicable commitments;
- Measuring and test equipment calibration was current;
- Applicable prerequisites described in the test procedures were satisfied;
- Test frequencies met Technical Specification requirements to demonstrate operability and reliability;
- Tests were performed in accordance with the test procedures and other applicable procedures;
- Test data and results were accurate, complete, within limits, and valid; and
- Test equipment was removed after testing.

This review constituted one inspection procedure sample.

b. Findings

Introduction: **A Green finding associated with an NCV of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when an inadvertent inventory loss from the Unit 2 RWST** occurred. The inventory loss occurred when licensee personnel performed two procedures concurrently, which was not appropriate to the circumstances due to the equipment configuration conflicts created by performing the test procedures in this manner.

Description: The original outage schedule sequence prescribed on May 10, 2005, that Test Procedure IT-515B, "Leakage Reduction and Preventive Maintenance Program Test of Safety Injection Test Line (Refueling) Unit 2," was performed prior to the start of Test Procedure ORT-6, "Containment Spray Sequence Test Unit 2." However, a potential conflict was identified with respect to Procedure ORT-6 and potential commitments made for the movement of the RVH. As a result, the licensee personnel in the outage control center changed the outage schedule to have both tests performed concurrently. The licensee performed a limited evaluation for conflicts between the two test procedures, which included a schedule review, and a tagout comparison using software to perform a tagout conflict check on the licensee's electronic tagging system.

Procedure IT-515B commenced in the morning of May 10, 2005, while Procedure ORT-6 was briefed in the morning and scheduled to start mid-day of the same day. During the sign-off of initial conditions for Procedure ORT-6, licensee personnel discovered that a procedure conflict existed between Procedures ORT-6 and IT-515B with respect to the position of the control switches for Containment Spray Pumps 2P-14A and 2P-14B. Procedure IT-515B required the control switches in pullout, while Procedure ORT-6 required the control switches in 'auto' with the associated electrical breakers in test. Licensee personnel discussed the situation and determined that Procedure IT-515B required the control switches in pullout for equipment protection only. Therefore, in part due to a perceived higher priority for completion of Procedure ORT-6 due to coordination of personnel, licensee personnel elected to stop Procedure IT-515B in the middle of the test, and begin Procedure ORT-6.

Following completion of testing, the operators implemented the restoration section of Procedure ORT-6 which required the containment spray pump suction valves to be opened. However, these valves were previously closed in Procedure IT-515B to establish a vent path on the suction side of the containment spray pumps. When the suction valves were opened per Procedure ORT-6, a leak path from the Unit 2 RWST to the primary auxiliary building floor drain was established. Approximately 300 gallons of water drained from the RWST to the sump, prior to isolation of the leak path.

The licensee determined the cause of the event was inadequate conflict identification between the two test procedures, and the licensee determined the performance of the two tests concurrently was incorrect and should not have been scheduled or performed in this manner. In addition, the licensee identified that upon realization of the containment spray control switch alignment conflict , the decision to place Procedure IT-515B on hold and commence Procedure ORT-6 was made without adequate evaluation through verification and validation of any additional potential procedure conflicts.

The licensee also identified that the tagout conflict check comparison performed by licensee personnel using the licensee's electronic tagging system was not performed in the correct manner. Furthermore, the licensee determined that a general knowledge weakness existed amongst licensee personnel on how to correctly perform procedure conflict checks using the electronic tagging system.

Analysis: The inspectors determined that the failure to maintain configuration control of safety systems while performing procedures used to test safety systems was a performance deficiency which warranted a significance determination.

The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Disposition Screening," since the finding was associated with the configuration control and procedure quality attributes of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent core damage.

Using IMC 0609, Appendix G, Phase 1 Screening, CL 4, "PWR Refueling Operations: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer," specifically Section II.C, "Inventory Control Guidelines-Equipment," was applicable to this finding. The inspectors determined the finding affected equipment necessary for makeup to the refueling cavity; however, the finding did not meet the requirements for a Phase 2 or Phase 3 analysis per Appendix G because the finding did not: result in an increase in the likelihood of a loss of RCS inventory; degrade the licensee's ability to terminate a leak path; or degrade the licensee's ability to recover decay heat removal. Therefore, the finding was considered to be of very low significance (Green).

The inspectors also determined that a primary cause of this finding was related to a cross-cutting issue in the area of human performance, because the licensee personnel failed to verify and validate that the performance of two procedures concurrently was appropriate to the circumstances.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” states, in part, that activities affecting quality shall be satisfactorily accomplished in accordance with instructions, procedures or drawings of a type appropriate to the circumstances. Contrary to this, performance of Procedure IT-515B in conjunction with Procedure ORT-6 was not appropriate to the circumstances, as evidenced by the inadvertent inventory reduction in the Unit 2 RWST. Therefore, the inspectors determined this finding was a violation of Criteria V, “Instructions, Procedures and Drawings.” Because this violation was of very low safety significance, non-willful, non-repetitive, and documented in the licensee’s corrective action program as CAP064491, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000301/2005004-05).

The licensee took immediate corrective actions to correct the condition by isolating the leak path. In addition, the licensee planned additional corrective actions to further ensure the outage schedule is adhered to in terms of planned test sequences in the outage schedule, address knowledge issues associated with performing procedure conflicts with the licensee’s tagging software and to enhance plant procedures regarding resolution of conflicts between procedures in progress concurrently.

.2 Selected Surveillance Test Reviews

a. Inspection Scope

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

- Effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- Acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- Plant equipment calibration was correct, accurate, properly documented, as-left setpoints were within required ranges, and the calibration frequency was in accordance with TS, 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 Technical Specification 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 in-service testing activities, testing was performed in accordance with the applicable version of Section XI, ASME Code, and reference values were consistent with the system design basis;
- Where applicable, test results not meeting acceptance criteria were addressed with an adequate OPR or the system or component declared inoperable;
- Where applicable for safety-related instrument control surveillance tests, reference setting data was 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.

During this inspection period, the inspectors observed activities associated with the following surveillance procedures, which constituted 10 quarterly inspection procedure samples:

- ORT-3A, SI Actuation With Loss Of Engineered Safeguards AC (Train A) Unit 2;
- ORT-3B, SI Actuation With Loss Of Engineered Safeguards AC (Train B) Unit 2;
- IT-215, SI Valve (Cold Shutdown) Unit 2;
- IT-305, Main Feedwater Line Check Valve Unit 2;
- IT-535C/D, Leakage reduction and PMT Program Train A SI and RHR Piggyback Test;
- ORT-59, Unit 2 Train A Spray System Containment Isolation Valve Leakage Test;
- IT-02A, SI Flow Test;
- IT-06B, Unit 2 Containment Spray Pump Test;
- ORT-3C, AMSAC and AFW Testing; and
- Routine Maintenance Procedure (RMP) 2RMP-9071-2, 4160/480 Degraded/Loss of Voltage Test.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

.1 Temporary Modifications

a. Inspection Scope

The inspectors conducted in-plant observations of physical changes to the plant and equipment associated with Temporary Modification 2005-008, “Reactor Vessel Head Temporary Modification for Core Cooling.” The inspectors reviewed design basis documents (DBDs) and safety evaluation screenings to determine if the modifications were consistent with applicable documents, drawings, and procedures. The inspectors also reviewed the post-installation results to confirm that any impacts of the temporary modifications on permanent and interfacing systems were adequately verified. The review of the temporary modifications constituted one inspection procedure sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

2OS1 Access Control to Radiologically Significant Areas (71121.01)

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

a. Inspection Scope

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

- RCP Motor and Seal Replacement;
- Reactor Pressure Vessel Head Replacement;
- Seal Table Activities (Thimble Tube Jacking); and
- Regenerative Heat Exchanger Maintenance.

This review represented one sample.

The inspectors reviewed the RWPs and work packages used to access these four areas and other high radiation work areas to identify the work control instructions and control barriers that had been specified. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. Workers were interviewed to verify that they were aware of the actions required when their electronic dosimeters noticeably malfunctioned or alarmed. This review represented one sample.

The inspectors walked down and surveyed (using a survey meter) these four areas to verify that the RWP, procedure, and engineering controls were in place; that licensee surveys and postings were complete and accurate; and that air samplers were properly located. This review represented one sample.

The adequacy of the licensee's internal dose assessment process for internal exposures greater than 50 millirem committed effective dose equivalent was assessed. (There were no internal exposures greater than 50 millirem.) This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, Licensee Event Reports, and Special Reports related to the access control program to verify that identified problems were entered into the corrective action program for resolution. This review represented one sample.

The inspectors reviewed nine corrective action reports related to access controls and HRA radiological incidents when available (non-performance indicators identified by the licensee in HRAs less than 1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

1. Initial problem identification, characterization, and tracking;
2. Disposition of operability/reportability issues;
3. Evaluation of safety significance/risk and priority for resolution;
4. Identification of repetitive problems;
5. Identification of contributing causes;
6. Identification and implementation of effective corrective actions;
7. Resolution of NCVs tracked in the corrective action system; and
8. Implementation/consideration of risk significant operational experience feedback.

This review represented one sample.

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

The inspectors reviewed licensee documentation packages for all performance indicator (PI) events occurring since the last inspection to determine if any of these PI events involved dose rates greater than 25 R/hr at 30 centimeters or greater than 500 R/hr at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. (There were no PI events occurring since the last inspection.) This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Job-In-Progress Reviews

a. Inspection Scope

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

- RCP Motor and Seal Replacement;
- Reactor Pressure Vessel Head Replacement;
- Seal Table Activities (Thimble Tube Jacking); and
- Regenerative Heat Exchanger Maintenance.

The inspectors reviewed radiological job requirements for these four activities, including RWP requirements and work procedure requirements, and attended As Low As Reasonably Achievable (ALARA) job briefings. This review represented one sample.

The above review is combined with IP71121.02, "ALARA Planning and Controls," and documented in Section 2OS2.2.

Job performance was observed with respect to these requirements to verify that radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings. The inspectors also verified the adequacy of radiological controls including required radiation, contamination, and airborne surveys for system breaches; radiation protection job coverage which included audio and visual surveillance for remote job coverage; and contamination controls. This review represented one sample.

Radiological work in high radiation work areas having significant dose rate gradients was reviewed to evaluate the application of dosimetry to effectively monitor exposure to personnel and to verify that licensee controls were adequate. The inspectors focused on work areas where the dose rate gradients were potentially severe, which increased the necessity of providing multiple dosimeters and/or enhanced job controls. (No jobs observed required multiple dosimeters.) This review represented one sample.

b. Findings

No findings of significance were identified.

.4 High Risk Significant, High Dose Rate High Radiation Areas and Very High Radiation Area (VHRA) Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager concerning high dose rate/HRA and VHRA controls and procedures, including procedural changes that had occurred since the last inspection, in order to verify that any procedure modifications did not substantially reduce the effectiveness and level of worker protection. This review represented one sample.

The inspectors conducted plant walkdowns to verify the posting and locking of entrances to high dose rate HRAs and VHRAs. This review represented one sample.

b. Findings

No findings of significance were identified.

.5 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation protection work requirements and evaluated whether workers were aware of the significant radiological conditions in their workplace, of the RWP controls and limits in place, and that their performance had accounted for the level of radiological hazards present. This review represented one sample.

The inspectors reviewed radiological problem reports which found that the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. These problems, along with planned and taken corrective actions were discussed with the Radiation Protection Manager. This review represented one sample.

b. Findings

No findings of significance were identified.

.6 Radiation Protection Technician (RPT) Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated RPT performance with respect to radiation protection work requirements and evaluated whether they were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities. This review represented one sample.

The inspectors reviewed radiological problem reports which found that the cause of the event was RPT error to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. This review represented one sample.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning And Controls (71121.02)

.1 Radiological Work Planning

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities ranked by estimated exposure that were in progress and reviewed the following five work activities of highest exposure significance:

- Reactor Pressure Vessel Head Replacement;
- RCP Motor and Seal Replacement;
- Split Pin Modification;
- Regenerative Heat Exchanger Maintenance; and
- Seal Table Activities (Thimble Tube Jacking).

This review represented one sample.

For these five activities, the inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements in order to verify that the licensee had established procedures, and engineering and work controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances. This review represented one sample.

The inspectors compared the results achieved including dose rate reductions and person-rem used with the intended dose established in the licensee's ALARA planning for these five work activities. Reasons for inconsistencies between intended and actual work activity doses were reviewed. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Job Site Inspections and ALARA Control

a. Inspection Scope

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

- Reactor Pressure Vessel Head Replacement;
- RCP Motor and Seal Replacement;
- Regenerative Heat Exchanger Maintenance; and
- Seal Table Activities (Thimble Tube Jacking).

The licensee's use of ALARA controls for these work activities was evaluated. Specifically, the licensee's use of engineering controls to achieve dose reductions was evaluated to verify that procedures and controls were consistent with the licensee's ALARA reviews, that sufficient shielding of radiation sources was provided for, and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding. This review represented one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

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

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

.2 SI Accumulator Level Transmitters

a. Inspection Scope

As a result of accumulator level transmitter issues experienced during the Unit 2 outage, the inspectors reviewed RCE-251, "Safety Injection Accumulator Level Instruments Returned to Service Without Proper Post Maintenance Testing," completed June 2004. The inspectors also observed the licensee's activities for the return to service of the Unit 2 Accumulator 2T-34A level transmitters (2LT-939 and 2LT-938) and evaluated the corrective actions taken from RCE-251.

The inspectors reviewed the CAP, the RCE, and the licensee's implementation of corrective actions discussed in the RCE for the root, significant contributing, and contributing causes identified by the RCE team. The inspection activity included a review of documents and interviews of plant staff involved with the RCE and corrective actions.

This review was an annual sample review of the licensee's problem identification and resolution program and constituted one inspection procedure sample.

b. Issues

During the Spring 2005 Refueling Outage, following the initial calibration of the Unit 2 Accumulator 2T-34A, the operators noted that the two accumulator level indicators were indicating five percent different. The differences between the indicators was within the indicating range of the instrumentation and was corrected prior to the accumulators being placed back in service. In addition, the accumulators were not required to be operable while the plant was in Mode 6 (refueling outage). However, the inspectors reviewed RCE-251 from the Spring of 2004 as compared to the issues associated with the level indicating difference in the Spring 2005 Unit 2 Refueling Outage. While no violation of NRC requirements and no findings were identified, the inspectors had some observations concerning the RCE and corrective action implementation.

In order to obtain an accurate level reading for the two instruments on both the 2T-34A and 2T-34B accumulators, a sight glass was required to be installed as part of the final test to assure accurate level indication. This activity was not called for previously as a corrective action for an accurate post maintenance test of the level indicators in RCE-251.

Corrective Action 7 of RCE-251 stated, "Determine whether 2LT-934, 2LT-935, and 2LT-938 should be modified to the same design configuration as 2LT-939. Initiate action to drive plant modification as appropriate." This item was dispositioned in CA056651, "Vendor questions the application and configuration of his equipment installed at Point Beach," in which the licensee concluded that, "Although the present transmitter installation of 2LT-934, 935, and 938 is not ideal, it performed the intended design function. The present installation is within the manufacturers specifications and the transmitters are still in production. Cost and Benefit analysis do not support a modification to make all the transmitters identical. Ops, I&C and Engineering consider the present installation acceptable and no replacements are required." As a final corrective action from the Unit 2 Spring 2005 accumulator issues, the licensee took as a corrective action, replacement of an additional level transmitter, such that each accumulator had two different types of level transmitters.

.3 Control Room Ventilation Flow Low Out of Specification

a. Inspection Scope

In December 2004, the licensee completed Revision 1 of RCE-270, for an event beginning on August 27, 2004, where 6½ days of a 7-day TS Action Statement (TS 3.7.9.A) were needed to bring control room ventilation flow above an acceptance criterion. The RCE was associated with CAP058833, F-16 CR [Control Room] Filter Flow Low Out of Specification per HPIP [Health Physics Implementing Procedure] 11.54, August 27, 2004.

The inspectors reviewed the CAP, the RCE, and the licensee's implementation of corrective actions discussed in the RCE for the root, significant contributing, and contributing causes identified by the RCE team. The inspection activity included a review of documents and interviews of plant staff involved with the RCE and corrective actions. This review was an annual sample review of the licensee's problem identification and resolution program and constituted one inspection procedure sample.

b. Issues

The control room ventilation system at Point Beach is nonsafety-related and, for the problem that was evaluated by the RCE, the TS Action Condition Completion Time was not exceeded; consequently, no violation of NRC requirements and no findings were identified. However, the inspectors had several observations concerning the RCE and corrective action implementation.

One of the four root causes of the event was that HPIP 11.54 did not identify the activity as satisfying TS surveillance requirements and the extent of condition assessment in the RCE determined that there was no unique numbering system for procedures, in general, used to satisfy TS surveillance requirements. A corrective action (CA) to address the general problem with procedures was initiated, but subsequently closed with no action.

In addition to the lack of a specific TS designator, the RCE identified other problems with HPIP 11.54 that needed to be corrected to help prevent recurrence of two of the four root causes; however, the revision of the procedure, originally scheduled for February 16, 2005, was subsequently deferred to June 30, 2005. In addition, the deferral contained the statement that if the procedure needed to be used before the revision date, it could be used without revision.

The extent of condition assessment identified that, in addition to the surveillance requirement for the control room ventilation system, TS surveillance requirements for two other systems contained low flow limits. However, there was no discussion in the RCE that the systems associated with these surveillances requirements were checked to ensure that a problem similar to that with the control room ventilation system was not present.

Corrective action (CA060506) was initiated to address one of the two contributing causes identified in the RCE for the event. This action, which involved developing guidance for designating TS testing in the work schedule, was closed without implementation. The inspectors noted that the RCE team that developed this corrective action included a member of the work scheduling group, and the corrective action was closed without implementation after an evaluation by another member of the work scheduling group who had not been part of the RCE team.

4OA3 Event Followup (71153)

.1 Event Notification 41754; Unanalyzed Condition Due to Appendix R Safe Shutdown Strategy Deficiency: On June 7, 2005, the licensee reported a condition under Title 10 CFR Part 50.72(b)(3)(ii)(B) as a result of an ongoing evaluation of a previously identified deficiency with the Appendix R Safe Shutdown Strategy with respect to the use of charging pumps for a fire in Fire Area A06, Bus 1B-32 480-Volt MCC Area. The licensee discovered a postulated fire could damage both the power and the control cables for 2 charging pumps, while the control cables for the remaining charging pump would also be damaged. The resultant condition of the remaining charging pump prevented operation of the pump as directed in the Safe Shutdown Analysis and Fire Organizational Plan.

The licensee took immediate corrective actions to mitigate the consequences of this issue. The licensee had entered this issue into the corrective action program and at the end of the inspection period, continued to analyze this condition. Therefore, this issue will be considered an Unresolved Item pending NRC review of the licensee evaluations (URI 05000266/2005004-06; 05000301/2005004-06).

.2 Event Notification 41758; Unanalyzed Condition Fire Organizational Plan No Longer Aligned with Safe Shutdown Analysis: On June 8, 2005, the licensee reported a condition under Title 10 CFR Part 50.72(b)(3)(ii)(B) when the licensee discovered a revision change to the Fire Organizational Plan which inadvertently omitted actions to be taken to preserve safe shutdown equipment in the event of a fire. Therefore the Fire Organizational Plan was no longer aligned with the Safe Shutdown Analysis as a result of the omission of manual actions necessary to accomplish safe shutdown in the current revision.

The licensee took immediate corrective actions to mitigate the consequences of this issue. The licensee had entered this issue into the corrective action program and at the end of the inspection period, continued to analyze this condition. Therefore, this issue will be considered an Unresolved Item pending NRC review of the licensee evaluations (URI 05000266/2005004-07; 05000301/2005004-07).

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 1R14.1 of this report had, as the primary cause, a human performance deficiency, in that, the licensee failed to ensure that the appropriate conditions were established prior to implementation of a tag out for the ‘B’ SI Train. As a result, when the tagout was implemented in the field, the inservice RHR heat exchanger was inadvertently taken out of service.

.2 A finding described in Section 1R151 of this report had, as the primary cause, a problem identification and resolution deficiency, in that, the licensee failed to perform causal evaluations for conditions adverse to quality which had received operability recommendations and also warranted causal evaluations.

.3 A finding described in Section 1R20.1 of this report had, as the primary cause, a problem identification and resolution deficiency, in that, the licensee failed to take adequate corrective actions to preclude repetition of a significant condition adverse to quality.

.4 A finding described in Section 1R22.1 of this report had, as the primary cause, a human performance deficiency, in that, the licensee failed to appropriately verify and validate that the performance of two procedures in conjunction was appropriate to the circumstances. As a result, when one procedure was stopped during implementation and a second procedure started, an inadvertent transfer of RWST inventory occurred.

4OA5 Other Activities

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

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

- New CRDM cooling system;
- New integral reactor vessel missile shield;
- RVH lift rig;
- Control rod drive mechanism (CRDM) seismic spacer plates;
- Cable drawbridge with connector panels;
- New RVLIS pipe supports;
- New RCGVS pipe supports;
- Handrail modifications and new ladders;
- New cable supports on head platform and lift legs; and
- RVH insulation.

a. Inspection Scope

From June 1, 2005, through June 24, 2005, the inspectors reviewed the licensee's design documentation associated with the installation of the HAUP. Specifically, the inspectors reviewed the design specification and a representative sample of design calculations to confirm that HAUP structures and components were designed in accordance with the requirements of the HAUP design specification and the AISC and ASME design codes.

b. Findings

Introduction: On June 24, 2005, the inspectors identified an NCV of Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," with multiple examples of very low safety significance (Green). Specifically, design calculations that support the HAUP design basis contained errors, including the failure to specify the AISC or ASME minimum fillet weld size requirements, the failure to transform bolt design loads into the analysis bolt pattern coordinate system, and the failure to evaluate the duct as a slender component in accordance with Appendix B5 of the AISC design code.

Description: The inspectors identified in calculation CN-RVHP-04-16, Revision 2, that the 3/16-inch fillet weld size for RCGVS supports attaching the rectangular column member to the baseplate was less than the minimum fillet weld size specified in ASME Table XVII-2452.1-1 (1977 Edition) and AISC Table J2.4 (9^{th} Edition). The installed column member is 1/4-inch thick, and the installed baseplate is 1-inch thick. To meet ASME and AISC minimum weld size requirements, the installed weld should have been 1/4-inch. The Code concern regarding minimum weld size was whether sufficient preheat was provided to ensure soundness of the weld.

The licensee subsequently provided information to justify the soundness of the installed 3/16-inch fillet weld. Because AISC adopted the provisions of AWS D1.1 for structural welding requirements, and the weld was installed using low-hydrogen electrodes, the licensee concluded that the minimum weld size in accordance with AWS D1.1-2000, Table 5.8 ensured weld soundness, i.e., a 3/16-inch minimum weld based on the thinner 1/4-inch member thickness was acceptable.

The inspectors reviewed later revisions of the ASME code (1998 Edition through 2000 Addenda) for guidance regarding minimum fillet weld requirements. ASME NF-3324.5(d)(1) permitted a fillet weld size less than 1/4-inch (6 mm) to join heavy section members, where the designer considered preheat requirements to ensure adequate weld deposition. The inspectors verified that AWS D1.1-2000 permitted a weld size based on the thickness of the thinner part for fillet welds installed using low-hydrogen electrodes and the weld procedure that installed the 3/16-inch fillet welds to verify use of low-hydrogen electrodes. Also, the inspectors verified that the design calculation demonstrated significant design margin for the 3/16-inch fillet weld size. The inspectors concluded there was a reasonable basis to justify the soundness of the installed 3/16-inch fillet weld.

The licensee entered the incorrect fillet weld size concern into their corrective action program as CAP065156. Based on the licensee's evaluation of the concern as documented in CAP065156, the inspectors considered this concern to be a design control issue. The licensee documented that during the design process the baseplate thickness was changed from 1/2-inch to 1-inch, but the weld size change in accordance with AISC Table J2.4 had not been made. Also, the licensee documented that its design contractor initially informed PBNP that rewelding would be required for the two RCGVS supports to meet Code. Therefore, the inspectors concluded that failure to increase the weld size to be in accordance with ASME Table XVII-2452.1-1 or AISC Table J2.4 was a design control error.

As a second example of a design control problem, the inspectors identified that non-conservative loads were used in bolt stress evaluations (calculations CN-RVHP-04-3, Revision 6 and CN-RVHP-04-44, Revision 3). Specifically, the bolt stresses were evaluated using a text book methodology (Structural Steel Design by McCormick) that determined structural section properties based on interaction between the bolts and the supporting steel baseplate. The neutral axes for the combined section were offset from the coordinate system that defined the reaction forces and moments that needed to be resolved into the joint. The reaction forces, when transformed into the combined section coordinate system, result in additional moment, and hence that additional moment was not evaluated.

The licensee planned to revise the calculations to transform the reaction forces into the combined section coordinate system. Preliminary evaluations by the licensee's design organization indicated that installed bolts still met design acceptance limits. The licensee entered this issue into their corrective action program as CAP065202 (CN-RVHP-04-3) and CAP065204 (CN-RVHP-04-44) to track revisions to the calculations.

As a third example of a licensee design control problem, the inspectors identified that the structural evaluation for the HAUP duct used for CRDM cooling, (calculation CN-RVHP-04-44, Revision 3) did not determine that the thin wall duct met AISC Table B5.1 limiting width-thickness ratios. The calculation used gross cross sectional properties to evaluate stress in the CRDM cooling duct. However, based on the width to thickness ratio of the installed CRDM cooling duct, the CRDM cooling duct should have been defined as a "slender compression element" and designed to rules specified in Appendix B5 of the AISC design code.

The licensee planned to revise the calculation to evaluate the installed CRDM cooling duct in accordance with the rules of AISC Appendix B5. Preliminary evaluations by the licensee's design organization indicated that the installed CRDM cooling duct still met AISC acceptance limits when Appendix B5 rules that consider the potential of local buckling of slender compression elements were considered. The licensee entered this issue into their corrective action program as CAP065204 to track revision to the calculation.

Analysis: The inspectors determined that a performance deficiency existed because the owner's acceptance review of their design contractor's supplied calculations failed to ensure that the calculations were accurate and complete. Furthermore, the inspectors determined that it was reasonably within the licensee's control to have identified the calculation errors and ensured that the appropriate design requirements for the HAUP were correctly translated into the design and installation documents.

The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because it affected the Mitigating Systems Cornerstone attribute of design control, and if left uncorrected, the finding could become a more significant safety concern. Specifically, the failure to specify the AISC or ASME Code required minimum fillet weld size, or the failure to transform bolt design loads into the analysis bolt pattern coordinate system, or the failure to evaluate slender section components in accordance with AISC Appendix B5 in similar design calculations could result in modifications that exceed licensing basis design acceptance limits.

The inspectors determined the finding was of very low safety significance (Green) because the calculation errors in these instances did not result in an HAUP structure or component exceeding their design basis acceptance limits.

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, for those systems, structures and components for which this appendix applies, are correctly translated into specifications, drawings, procedures and instructions. It further requires that the design control measures provide for verifying or checking the adequacy of the design to be in accordance with the HAUP design specification and the AISC and ASME design codes.

Contrary to the above, the adequacy of the design was not adequately verified or checked in the following instances:

- The inspectors identified on June 15, 2005, that calculation CN-RVHP-04-16, Revision 2, failed to determine that the 3/16-inch fillet weld size for RCGVS supports attaching the rectangular column member to the baseplate was less than the minimum fillet weld size specified in ASME Table XVII-2452.1-1 (1977 Edition) and AISC Table J2.4 (9^{th} Edition). The installed column member is 1/4-inch thick, and the installed baseplate is 1-inch thick. To meet ASME and AISC minimum weld size requirements, the installed weld should have been 1/4-inch. The Code concern regarding minimum weld size was whether sufficient preheat was provided to ensure soundness of the weld.

- The inspectors identified on June 17, 2005, that calculation CN-RVHP-04-3, Revision 6, and calculation CN-RVHP-04-44, Revision 3, approved on May 20, 2005, failed to transform bolt design loads into the analysis bolt pattern coordinate system. Specifically, the bolt stresses were evaluated using a text book methodology (Structural Steel Design by McCormick) that determined structural section properties based on interaction between the bolts and the supporting steel baseplate. The neutral axes for the combined section were offset from the coordinate system that defined the reaction forces and moments that needed to be resolved into the joint. The reaction forces, when transformed into the combined section coordinate system, result in additional moment, and hence that additional moment was not evaluated.

- The inspectors identified on June 22, 2005, that calculation CN-RVHP-04-44, Revision 3, approved on May 20, 2005, failed to determine that the thin wall CRDM cooling duct met AISC Table B5.1 limiting width-thickness ratios. The calculation used gross cross sectional properties to evaluate stress in the CRDM cooling duct. However, based on the width to thickness ratio of the installed CRDM cooling duct, the CRDM cooling duct should have been defined as a “slender compression element” and designed to rules specified in Appendix B5 of the AISC design code.

Because of the very low safety significance of the issues and because they were entered into the licensee’s corrective action program as CAP065156, CAP065202, and CAP065204, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000301/2005004-08).

.2 <u>Reactor Vessel Closure Head and CRDM Housing Replacement</u> (IP 71007)

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

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

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

a. Inspection Scope

From April 4, 2005, through April 8, 2005, and from May 9, 2005, through May 13, 2005, the inspectors reviewed the licensee's design changes associated with the replacement efforts.

The inspectors reviewed certified design specifications, certified design reports, (ASME) Code reconciliation reports, fabrication deviation notices, non-conformance reports, and design calculations to confirm that the replacement RVH and CRDM housings were in compliance with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subsection NB (1998 Edition including addenda through 2000 Addendum). Specifically, the inspectors confirmed that the design specifications and design reports were certified by registered professional engineers competent in ASME Code requirements. The inspectors confirmed that adequate documentation existed to demonstrate the certifying registered professional engineers were qualified in accordance with the requirements of the ASME Code Section III (Appendix XXIII of Section III Appendices). The inspectors also confirmed that the replacement RVH and CRDM housings were provided as Code Normal Pressure and Temperature stamped components.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

As part of the Refueling Outage, the licensee replaced the RVH and CRDM housings. The inspectors conducted inspections of these activities and in some cases performed inspections of specific implementation activities as samples of other inspection procedure modules as documented in Sections 1R02, 1R19, 1R20, and 1R23 of this Report. In addition, as part of this inspection the inspectors reviewed the following activities associate with the Reactor Vessel Closure Head replacement:

- Temporary modifications associated with this modification;

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

Issues associated with the RVH drop analysis identified by the inspectors are documented in Section 1R02 of this Report.

b. <u>Findings</u>

No findings of significance were identified.

.4 <u>Operational Readiness of Offsite Power (OSP)</u> (TI 2515/163)

The objective of TI 2515/163, "Operational Readiness of Offsite Power," was to confirm, through inspections and interviews, the operational readiness of OSP systems in accordance with NRC requirements. In May 2005, the inspectors reviewed licensee procedures and discussed the attributes identified in TI 2515/163 with licensee personnel. In accordance with the requirements of TI 2515/163, the inspectors evaluated licensee procedures against the attributes discussed below.

The operating procedures that the control room operator use to assure the operability of the OSP have the following attributes:

1. Identify the required control room operator actions to take when notified by the transmission system operator (TSO) that post-trip voltage of the OSP at the nuclear power plant will not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply.

2. Identify the compensatory actions the control room operator is required to perform if the TSO is not able to predict the post-trip voltage at the nuclear power plant for the current grid conditions (Plant procedures did not specifically address the situation where the TSO is unable to predict the post-trip voltage, but did include

compensatory actions if the licensee did not have reasonable assurance of the post-trip voltage).

3. Identify the notifications required by Title 10 CFR Part 50.72 for an inoperable OSP system when the nuclear station is either informed by its TSO or when an actual degraded voltage condition is identified.

The procedures to ensure compliance with Title 10 CFR Part 50.65(a)(4) have the following attributes:

1. Direct the plant staff to perform grid reliability evaluations as part of the required maintenance risk assessment before taking a risk-significant piece of equipment out-of-service to do maintenance activities.

2. Direct the plant staff to ensure that the current status of the OSP system has been included in the risk management actions and compensatory actions to reduce the risk when performing risk-significant maintenance activities or when loss of offsite power or station blackout mitigating equipment are taken out-of-service.

3. Direct the control room staff to address degrading grid conditions that may emerge during a maintenance activity.

4. Direct the plant staff to notify the TSO of risk changes that emerge during ongoing maintenance at the nuclear power plant.

The procedures to ensure compliance with Title 10 CFR Part 50.63 have the following attribute:

1. Direct the control room operators on the steps to be taken to try to recover OSP within the station blackout coping time.

The results of the inspectors' review were forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation.

4OA6 <u>Meetings</u>

.1 <u>Exit Meeting</u>

On July 22, 2005, the resident inspectors presented the inspection results to Mr. D. Koehl and members of his staff, who acknowledged the findings. The licensee did not identify any information, provided to or reviewed by the inspectors, as proprietary in nature.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Inservice Inspection Procedure (IP 71111.08) with Ms. F. Flentje and other members of the licensee's staff on April 21, 2005. The licensee confirmed that none of the potential report input discussed was considered proprietary.

- Reactor vessel head replacement procedure (IP 71007) with Mr. M. Lorek and other members of the licensee's staff on June 28, 2005. The licensee confirmed that the design documentation prepared by their contractors was considered proprietary. All copies of proprietary documents would be returned to the licensee.

- Occupational Radiation Safety inspection with Mr. D. Koehl, Site Vice President on April 15, 2005. The licensee confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Koehl, Site Vice-President
J. McCarthy, Director of Site Operations
M. Lorek, Plant Manager
A. Capristo, Regulatory Affairs Manager
N. Stuart, Maintenance Manager
G. Casadonte, Fire Protection Coordinator
G. Corell, Chemistry Manager
J. Schweitzer, Site Engineering Director
R. Milner, Business Planning Manager
G. Packard, Operations Manager
G. Sherwood, Engineering Programs Manager
F. Forrest, Nuclear Oversite Manager
C. Jilek, Maintenance Rule Coordinator
T. Kendall, Engineering Senior Technical Advisor
B. Kopetsky, Security Coordinator
F. Flentje, Senior Regulatory Compliance Engineer
R. Ladd, Fire Protection Engineer
M. Ray, Emergency Planning Manager
L. Peterson, Design Engineer Manager
C. Sizemore, Training Manager
W. Smith, Production Planning Manager
C. Hill, Assistant Operations Manager
P. Smith, Licensed Operator Requalification Training Group Lead
J. Strharsky, Planning and Scheduling Manager
S. Pfaff, Site Assessment Manager
D. Schuelke, Radiation Protection Manager
P. Wild, Design Engineering Projects Supervisor
L. Hawki, Engineering Supervisor
J. McNamara, Engineering Supervisor
J. Tabat, Responsible Engineer, Reactor Vessel Head Project
B. Jensen, Level III
R. Turner, Inservice Inspection Coordinator
S. Forsha, Engineer, Nuclear Oversight

Nuclear Regulatory Commission

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000266/2005004-01; 05000301/2005004-01	URI	Reactor Vessel Head Drop Analysis (Section 1R02)
05000301/2005004-02	NCV	Inadvertent Loss of Decay Heat Removal Capability (Section 1R14.1)
05000266/2005004-03; 05000301/2005004-03	FIN	Adverse Trend of Failure to Ensure Causal Evaluations for Conditions Adverse to Quality for which Operability Recommendations were Performed (Section 1R15.1)
05000266/2005004-04; 05000301/2005004-04	NCV	Failure to Implement Adequate Corrective Actions to Preclude Repetition of a Significant Condition Adverse to Quality (Section 1R20.1)
05000301/2005004-05	NCV	Inadvertent Refueling Water Storage Tank Inventory Loss (Section 1R22.1)
05000266/2005004-06; 05000301/2005004-06	URI	Event Notification for Unanalyzed Condition Due to Appendix R Safe Shutdown Strategy Deficiency
05000266/2005004-07; 05000301/2005004-07	URI	Event Notification for Unanalyzed Condition Where Fire Organizational Plan No Longer Aligned with Safe Shutdown Analyses
05000301/2005004-08	NCV	Multiple Design Calculation Errors of Very Low Safety Significance (Section 4OA5)

Closed

05000301/2005004-02	NCV	Inadvertent Loss of Decay Heat Removal Capability (Section 1R14.1)
05000266/2005004-03; 05000301/2005004-03	FIN	Adverse Trend of Failure to Ensure Causal Evaluations for Conditions Adverse to Quality for which Operability Recommendations were Performed (Section 1R15.1)
05000266/2005004-04; 05000301/2005004-04	NCV	Failure to Implement Adequate Corrective Actions to Preclude Repetition of a Significant Condition Adverse to Quality (Section 1R20.1)
05000301/2005004-05	NCV	Inadvertent Refueling Water Storage Tank Inventory Loss (Section 1R22.1)
05000301/2005004-08	NCV	Multiple Design Calculation Errors of Very Low Safety Significance (Section 4OA5)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

SCR 2004-0317; 10 CFR 50.59/72.48 Screening for MR 03-056 - Replacement of Unit 2 Reactor Vessel Closure Head; dated February 3, 2005
EVAL 2004-006; 10 CFR 50.59 Evaluation for MR 03-056 - Replacement of Unit 2 Reactor Vessel Closure Head; dated February 17, 2005
SCR 2004-0305; 10 CFR 50.59/72.48 Screening for MR 03-057 - Replacement of the Unit 2 Reactor Head Metallic Reflective Insulation; dated December 13, 2004
SCR 2004-0318-05; 10 CFR 50.59/72.48 Screening for MR 03-058 - Reactor Vessel HAUP Package - Unit 2; dated June 1, 2005
SCR 2004-0314; 10 CFR 50.59/72.48 Screening for MR 03-059 - Unit 2 Containment Equipment Hatch Shield Wall Modification; dated February 11, 2005
SCR 2004-0196; 10 CFR 50.59/72.48 Screening for MR 03-060 - Analog Rod Position Indication Cable and Connector Modification; dated February 24, 2005
SCR 2004-0197; 10 CFR 50.59/72.48 Screening for MR 03-061 - CRDM Cable and Connector Modification; dated February 22, 2005

1R04 Equipment Alignment

DBD-13; Spent Fuel Pool Cooling and Filtration; Revision 3; October 20, 2004
Drawing West 110E018 Sh 4; Auxiliary Cooling System; Revision 42
FHAR FZ 681; Fire Area A01-F; Gas Turbine - G05; April 2004

1R05 Fire Protection

FHAR FZ 611; Fire Area A46; April 2004
FHAR FZ 615; Fire Area A46; April 2004
FHAR FZ 583; Fire Area A01-E; Barrier Rating Requirements and as Built Data; April, 2004
FHAR FZ 583; Fire Area A01-E; Fire Zone Data; U2 Turbine Building General Area - 8 ft.; April, 2004
FHAR FZ 681; Fire Area A01-F; Fire Hazards Analysis Report; Gas Turbine - G05; April 2004
FHAR FZ 552; Fire Area A38; Fire Hazards Analysis Report; SW Pump Room; April 2004
Drawing WE PBC 218 Sh. 1; Fire Protection for Site Plan; Revision 8; August 11, 2001

1R08 Inservice Inspection Activities

PBNP Ultrasonic Calibration Record; RHE-N4-IRS; dated April 11, 2005
PBNP Ultrasonic Pressure Vessel Examination Record; RHE-N4-IRS; dated April 11, 2005
PBNP Ultrasonic Calibration Record; AC-06-SI-2001-18; dated April 13, 2005
PBNP Ultrasonic Piping Examination Record; AC-06-SI-2001-18; dated April 13, 2005
PBNP Ultrasonic Calibration Record; RPV-14-683; dated April 17, 2005

PBNP Ultrasonic Pressure Vessel Examination Record; RPV-14-683; dated April 18, 2005
PBNP Remote Visual Examination Record; Reactor Pressure Vessel Bottom Mounted Instrumentation Nozzles; dated April 4, 2005
PBNP Procedure NDE-757; Visual Examination For Leakage of Reactor Pressure Vessel Penetration; Revision 4
PBNP Procedure NDE-176; Manual Ultrasonic Examination of Regenerative Heat Exchanger Welds and the Corner Region of the Nozzle Inside Radius Sections; Revision 4
PBNP Procedure NDE-168; Manual Ultrasonic Examination of RPV Flange to Upper Shell Weld; Revision 8
PBNP Procedure NDE-173; PDI Generic Procedure for the Ultrasonic Examination of Austenitic Piping Welds; Revision 6
Record of Certification NDE Personnel; William Jensen (Point Beach) and Steven Williams (Lambert Macgill Thomas)
Indication Disposition Report 2003-0008 Engineering Evaluation; dated October 22, 2003
Liquid Penetrant Examination Record for FW-16-FW-2001 28 BC; dated October 21, 2003
Work Order 0300621; Remove and Replace RWST Level Equalizing Line; dated October 28, 2003
Work Order 0206562; Replace P-15A SI Pump to RC Loop A Cold Leg SI Check Valve SI-00845A; dated December 26, 2003
CAP 061099; Boric Acid Indication Recording in PC-24, Containment Inspection CL; dated December 16, 2004
CAP 060731; 2RC-431A (2PI-456A) Indication Was Not Readable; dated November 23, 200
CAP 054371; Boric Acid Leak Eroding Pipe and Valve Support; dated March 1, 2004
CAP 060681; RC-431A Bellows Leakage and Bellows Gauge Fitting Leakage; dated November 21, 2004
CAP 060791; Instrument Mounting Degraded Due to Boric Acid Leak; dated November 30, 2004

1R11 Licensed Operator Qualifications

Simulator Exercise Guide PB-LOR-053-001E; Cycle 05-03 As Found; Revision 1

1R12 Maintenance Effectiveness

Performance Criteria Assessments for CV Since 4/1/2003
Maintenance Rule Unavailability Data Sheet; Unit 2; System CV; Data between April 1, 2003 and April 1, 2005
Maintenance Rule Unavailability Data Sheet; Unit 1; System CV; Data between April 1, 2003 and April 1, 2005
Work Orders for CV with M, F or C Between April 1, 2003 and May 1, 2005
Function List for CV Chemical and Volume Control Sorted for Maintenance Rule Coordination File T7.2.6; System CV; PB1
Maintenance Rule (a)(1) System Action Plan CL and Approval; System CV; November 8, 2004

DBD-04; Chemical and Volume Control System Design Basis Document; Revision 3; November 17, 2004
CA025716; Resolution of CV a(1) Status Unknown - Resolution of 2P-2C Tripping Issue; June 28, 2002
Function List for DG DIESEL GENERATOR sorted for Maintenance Rule
Documentation of Maintenance Rule Performance Criteria; dated July 18, 2000
Performance Criteria Assessments For RH (2 years); dated May 2005
Function List for RH Residual Heat Removal (LPSI); dated May 2, 2005
Work Orders for RH Residual Heat Removal (2 years); dated May 2005
Maintenance Rule Unavailability Data Sheet (2 years); dated May 2005
System Health Report Residual Heat Removal/LHSI System (RH)
Documentation of Maintenance Rule Performance Criteria - DG
Maintenance Rule (a)(1) System Action Plan CL and Approval for Diesel Generators, 9/10/03
Work Orders for DG with M, F or C in MPFF field initiated or completed between 4/1/23 and 5/1/05.
Point Beach Maintenance Rule Unavailability Data Sheets between 4/1/2003 and 4/1/2005 for DG: PB0-G01, G02, G03, and G04
Performance Criteria assessments for DG since 4/1/2003
NP 7.7.4, Scope and Risk Significant Determination for the Maintenance Rule, Rev. 7
NP 7.7.5, Determining, Monitoring and Evaluating Performance Criteria for the Maintenance Rule, Rev. 15
NP 7.7.6, Work Order Review and MRPFF Determination for the Maintenance Rule, Rev. 5
NP 7.7.7, Guideline for Maintenance Rule Periodic Report, Rev. 2
NP 7.7.9, Facilities Monitoring Program, Rev. 3

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

E-1 Report; Work Week Schedules
NP 10.3.6; Outage Safety Review and Safety Assessment; Revision 11
NP 10.3.7; On-Line Safety Assessment; Revision 8
LT-86 Online Near Critical Path; FL-86 Online Near Critical Path Set; June 6, 2005
Work Activity Risk Assignment; LT-81 Human Error Risk Bar chart; June 6, 2005
Work Week Additions/Deletions; June 9, 2005
PBNP Shutdown Safety Assessment and Fire Condition CL; Unit 2; June 9, 2005
Safety Monitor 3.5a; Unit 1; June 9, 2005
Working Schedule; LT-03 Standard Daily Publication; FL-03 Execution Publication Set; June 7, 2005

1R14 Personnel Performance Related to Non-Routine Plant Evolutions and Events

Operations Event Clock Reset; Unit 2 RHR Flow Lost During Hanging of Danger Tags; April 19, 2005
CAP063933; Action Identified Out of Loss of ‘B’ RHR Train Footpath Event Investigation; April 21, 2005
New Danger Tagging Requirements; Revision 1; April 25, 2005
NP 5.3.3; Incident Investigation and Post-Trip Review; Revision 5; September 27, 2004
CAP063862; Possible Unplanned Orange Path Entry for Core Cooling; April 19, 2005

CAP063905; RHR Flow Control Valve Total Isolation Question; April 20, 2005
CAP063906; Question Unit 2 RHR Pump Performance During April 19 Loss of Force Cooling Event; April 20, 2005
CAP063907; Question/Challenge to How RHR HX Bypass Flow Control Valve is Operated; April 20, 2005
CAP063860; Unit 2 RHR Flow Secured During Hanging of Tagout; April 19, 2005
NP 1.9.15; Tagging Procedure; Revision 25; October 20, 2004
CAP064307; U2R27 Contingencies for an RHR Train OOS; April 25, 2005
SEP-1, Unit 2; Degraded RHR System Capability; Revision 4; Attachment B, Local RHR Valve Alignment; December 22, 2003
SEP-1, Unit 2; Degraded RHR System Capability; Revision 4; December 22, 2003
Drawing West 110E018 Sh 1; Auxiliary Coolant System; Revision 57
RP1A Preparation for Refueling; Revision 66; April 26, 2005
RCE 278; Unit 2 RHR Flow Lost During Hanging of Tag Out; May 23, 2005
CAP064739; Control Room to Field Communication; May 23, 2005
CAP064662; Control Board Awareness Related to Inadvertent Start of 2P-10B RHR Pump; May 18, 2005
CAP064038; Isolation of SW Flow to Unit 1 CCW HX's; April 25, 2005

1R15 Operability Evaluations

CAP063321; Relay 2-271X2/B04; Operational Decision Making Issue Evaluation Document; April 5, 2005
OPR 132; Document No. 6090-FT; Page 88 of 103; Part 6.9; Load Transient Test
OPR000132; Unanticipated Load on G-04 Emergency Diesel Generator; Revision 0; April 5, 2005
TS 9; Control Room Heating and Ventilation System Monthly Checks; Revision 27; December 15, 2003
OPR000127; Local control using AOP-10A could result in pressurizer indicated level dropping off scale in the event of a control room fire
OPR000125; Formal documentation of Battery Charger TSR change
CAP064515; Error in initiating temp change 2005-015 for AOP-10A, Revision 39
CAP062734, AOP-10A does not ensure pressurizer level will remain on scale
CAP061059; Appendix R Pressurizer Level Criterion May Not Be Met Under Certain Conditions
OPR000137; Increased Valve Weights in SW Return Piping JB-02 From CCW Heat Exchangers: ISW-00322, HX-12A CC HX Outlet; SW-00360, HX-12B CC HX Outlet; SW-00315, HX-12C CC HX Outlet; 2SW-00307, HX-12D CC HX Outlet; May 3, 2005
Speed Number 2005-028; Replacement Valve for 150# Class, 12" Powell Globe Valve Made by Weir Valve Using Powell Design; Revision 1; May 4, 2005
Sargent & Lundy Issue Summary; Project No. 11165–046; Calc No. WE-300035; May 4, 2005
WE-300035-01-1; SW Return from Unit 1 Containment Penetrations and CCW Heat Exchangers to the 20" JB-2 Discharge Header (Pipe Classes HB-19 & JB-2); PB1; May 9, 2005
CAP063341; Electrical Penetration 2Q-03 Exceeds Admin Limit
PBNP FSAR; Containment System Structure; Page 5.1-83 of 113; Figure 5.1-3; August 2004

NPM 2004-0538; Plant Health Committee “Subcommittee” Meeting Minutes;
August 17, 2004

1R16 Operator Workarounds

Operator Work Arounds Summary - Category 2, 5/23/2005
Operator Work Arounds Summary - Category 3, 5/23/2005
Open Operator Work Arounds and Challenges, 5/23/2005
Open Control Board Stickers, 5/23/2005
Instruments Out of Service, 5/23/2005
Open Control Board Deficiencies, 5/23/2005
Pri 3 Work Orders, 5/23/2005
Operator Work Around Aggregate Impact, 5/24/2005
Total Operator Burden Summay, Lit Annunciators, Operator Work Arounds & Challenges, and Temporary Modifications summaries, January 2004 to April 2005
Operator Work Around Meeting Minutes, September 20, 2004, through March 21, 2005
NP 2.1.4, Operator Burndens, Rev. 5

1R19 Post-Maintenance Testing

Drawing PBC-248; Standard Anchor Point for Personnel Full Arrest or Restraint; Revision 1; June 22, 2002
IM-0540; Installation and Operating Instructions for SM/LA-E3330 and SM/LA-R3300 Series Fail Safe Actuators; Revision C; July 2, 1997
Work Order 0501557; W-15 Control Room Washroom Exhaust Fan Damper; April 18, 2005
Work Order 0501557 Addendum 1; VNCR-06748; W-15 Control Room Washroom Exhaust Fan Damper; April 18, 2005
Work Order 0415163; PWR From 2X-04 LV Station Aux Transformer Incoming; January 7, 2005
Work Order 0415163 Addendum 1; PWR From 2X-04 LV Station Aux Transformer Incoming; April 9, 2005
Work Order 0415163; 2X-04 LV Station Aux Transformer Incoming Line; Breaker Swap During U2R27; January 16, 2005
RMP 9201; Control and Documentation for Troubleshooting and Repair Activities; Revision 2; December 15, 2004
TS 9; Control Room Heating and Ventilation System Monthly Checks; Revision 27; December 15, 2003
ABB Power T&D Company Inc. Instruction Leaflet 41-753.1L; Type MG-6 Multi-Contact Auxiliary Relay
ICP 9.13; Bench Testing of RX Safeguards, RX Protection System, or Other Miscellaneous Relays; Revision 7; January 18, 2005
Work Order 0501407; 480 Safeguards Load Center; Replace MG-6 Auxiliary Relay
Drawing WEST 499B466 SH.311; Elementary Wiring Diagram; 2B-04; 480V; Undervoltage Scheme
Drawing WEST 499B466 SH.387A; Elementary Wiring Diagram Load Center 2B04
PBTP 133; Post Maintenance Test of 2-271X2/B04; Revision 1; Unit 2; April 28, 2005
ORT-59; Train ‘A’ Spray System CIV Leakage Test (U-2); PB2; Revision 27; April 25, 2005

ORT-59; Train 'A' Spray System CIV Leakage Test; PB2; Revision 23; April 22, 2002
ORT-59; Train 'A' Spray System CIV Leakage Test; Unit 2; Revision 25; October 20, 2003
MDB 3.2.6 2B42; 480 V AC MCCs; Unit 2; Revision 16; August 26, 2004
Work Order 0414540; MR 01-128*N; 2SI-850B, 2P-10B RHR Pump Sump B Suction MCC Bucket 2B52-421B Replacement; Unit 2
Work Order 0414541; MR 01-128*N; 2SI-852B, Low Head SI Core Deluge Isolation MCC Bucket 2B52-421F Replacement; Unit 2
Work Order 0414544; MR 01-128*N; 2SI-896B 2P-15B; SI Pump Suction Isolation MCC Bucket 2B52-422C Replacement; Unit 2
Work Order Work Order 0414546; MR 01-128*N; 2SI-878B 2P-15A; SI Loop B Isolation MCC Bucket 2B52-422J Replacement; Unit 2
Work Order 0414548; MR 01-128*N; 2SI-851B; Containment Sump B Isolation MCC Bucket 2B52-423C Replacement; Unit 2
Work Order 0414564; MR 01-128*N; 2RC-516 Power Relief Isolation MCC Bucket 2B52-427F Replacement; Unit 2
Work Order 0414566; MR 01-128*N; 2AF-4006 AFP Suction From SW Supply MCC Bucket 2B52-427M Replacement; Unit 2
IT-205; PORV and Block Valves (Cold Shutdown); Unit 2; Revision 28; May 2, 2005
Work Order 0414704; 2RC-430 Repair Valve/Suspect Leakage; May 8, 2005
2-SI-21X; SI Auxiliary; W/D E-2094; Sh.74L.A2.2C167; Model BFD84S
Work Order 0501865001; Train B SI Auxiliary Relay; Unit 2;SI-21X; May 17, 2005
ICP 9.13; Bench Testing of RX Safeguards, RX Protection System, or Other Miscellaneous Relays; Revision 7
Work Order 0501865; Train B SI Auxiliary Relay; Unit 2; May 13, 2005
WEST 110E163-2; SH.7B; Engineered Safety Features (ESF) Systems Train B Reactor Safeguards System Unit 2; Revision 11

1R20 Refueling and Outage Activities

OP 4D; Part 1; Draining The RCS; Revision 65; February 28, 2005
PBNP Work Order Initiation Tag 219705; 2LT-495; April 5, 2005
PBNP Work Order Initiation Tag 219706; 2LT-497; April 5, 2005
Calculation/Addendum 1999-0103; RCP Rotating Element Strongback Hold Down Beam (W10 x 49); October 16, 1999
CAP063501; Scaffold Construction Impedes EOP-1.3 Temp Shielding Access; April 8, 2005
CAP063122; 2P-2C Charging Pump Seal Leakage Excessive; March 29, 2005
CAP063562; Strong Back 1-1/2 Inch Spacer Blocks not used in RMP 9002-21; April 11, 2005
CAP063314; Electrical Penetration Failed Leak Rate Test; April 4, 2005
CAP063323; Unanticipated Load on G-04 EDG; April 5, 2005
ARB 2C03 2D 3-4; 2P-29 AFP Low Suction Pressure Trip; Revision 6; March 21, 2005
ARB C01 A 4-9; Aux Feed Pump Suction Pressure Low; Revision 6; August 3, 2000
NF-NMC-04-170; Figure 1, Point Beach Unit 2, Cycle 28, Region and Fuel Assembly Locations; November 5, 2004
PBF-5105; SNM and Other Device Physical Inventory; Unit 2 Reactor Core Map; Cycle 28; Revision 1; December 21, 2004
BALCM Appendix B; Boric Acid Examination Guidelines; Revision 1

2-PT-RCS-1; RCS Pressure Test - Inside/Outside Containment; Unit 2; Revision 0
CAP064689; ASME Section XI Pressure Test Required Hold Time Not Met; Unit 2; May 19, 2005
CA062767; Valve Failures During ORT-59 (Cont. Spray ORT) 2SI-862G and 2SI-868A; Unit 2; April 26, 2005
CAP063032; CL-1E Not Updated for U2R27; March 24, 2005
CAP063088; Kewaunee Nuclear Plant NRC Resident Questions Regarding U2R27 Shutdown Safety; March 28, 2005
RCE 254; Potential Loss of Hot Leg Vent Path During Nozzle Dame Installation; Revision 2; April 9, 2004
RCE 276; Planned Orange Path Inappropriately Included in the U2R27 Outage Schedule; May 24, 2005

1R22 Surveillance Testing

ORT 3B; SI Actuation With Loss of Engineered Safeguards AC (Train B); Unit 2; Revision 36; March 7, 2005
ORT 3B; SI Actuation with Loss of Engineered Safeguards AC (Train B); Unit 1; Temp Change No. 2005-20
ORT 6; Containment Spray Sequence Test Unit 2; Revision 21, March 14, 2005
CAP064494; Multiple procedure issues with ORT 6 (Unit 2)
CAP064491; IT-515B and ORT-6 conflict
ORT-59; Train ‘A’ Spray System CIV Leakage Test; PB2; Revision 23; April 22, 2002
ORT-59; Train ‘A’ Spray System CIV Leakage Test; Unit 2; Revision 25; October 20, 2003
CAP064046; Valve Failures During ORT-59 (Cont. Spray ORT) 2SI-862G and 2SI-868A; April 25, 2005
CAP064059; 2SI-868B Failed Secondary Boundary Test; April 25, 2005
CAP064066; During the Performance of ORT-59 some of the valves exceeded their admin limit; April 26, 2005
ORT-59; Train ‘A’ Spray System CIV Leakage Test; Unit 2; Revision 27; April 24, 2005
IT-535D; Leakage Reduction and Preventative Maintenance Program Train ‘B’ HHSI and RHR “Piggyback” test (Refueling) Unit 2; Revision 6; May 9, 2005
IT-535C; Leakage Reduction and Preventative Maintenance Program Train ‘A’ HHSI and RHR “piggyback” test (Refueling) Unit 2; Revision 7; May 9, 2005
ORT-3C; Aux Feed System and AMSAC Actuation; Unit 2; Revision 7, May 10, 2005
O-IT-006B; Containment Spray Pump and Valve Testing; April 1, 2005
IT-6B; Containment Spray Pumps and Valves (Cold Shutdown) Unit 2; Revision 10; May 17, 2005
IT-02A; High Head SI Pumps and Valves (CSD) Unit 2; Revision 14; May 17, 2005
WEST 110E035; SI System; Sheet 3; Revision 44

1R23 Temporary Plant Modifications

Westinghouse Drawing 1C51803; Read Assembly Upgrade Package Coolant Gas Vent Subassembly; Revision 1
Westinghouse Drawing 10007D26; Head Assembly Upgrade Package RVLIS Assembly; Revision 1
Westinghouse Drawing 1C51800; HAUP Reducing Orifice Coupling; Revision 0

Westinghouse Drawing 6474E32; RVH Coolant Gas Vent RVLIS Assembly; Revision 1
Calculation 2005-0022; Estimate of Injection Flow Rates for U2R27 RVH Drop Event
ECN 2005-0082; Affected Document(s) Revision: Drawing SK-TM-2005-03; Revision 0; May 10, 2005
AOP-8J; RVH Drop; Revision 0; May 13, 2002
ECN 2005-0084; Affected Document(s) Revision: Drawings SK-TM-2005-008-7, Revision 0; SK-TM-2005-008-8, Revision 0
SEP 2.3; Cold Shutdown LOCA; Revision 12; May 13, 2005
Calculation 2005-0023; Decay Heat, Water Makeup Requirement, Time to Boil, and Time to Core Uncovery for the U2R27 RVH Set; May 10, 2005
TM 2005-008; Unit 2 Hose Routing Contingency from Containment Spray Test Flanges to Replacement RV Head with 10 CFR 50.59/72.48 Screening Evaluations; May 10, 2005
CAP064706; Concerns with Viability of Temp Mod 2005-008 (Hose Conn. To RX Vessel Head); May 20, 2005
SEP 2; Shutdown LOCA Analysis; Revision 3; September 23, 2002
SCR 2005-0122; TM 05-008 Unit 2 Hose Routing Contingency from Containment Spray Test Flanges to Replacement RV Head; May 11, 2005
Bill of Materials; Temporary Modification TM 2005-008
NPM 2005-0312; Conceptual Design Review Board for TM 2005-008 (U2 Hose Routing Contingency from Containment Spray Test Flanges to Replacement RV Head); May 11, 2005
2RMP 9071-2; A-06 4160/480 Degraded and Loss of voltage Monthly Surveillance Test; Revision 14

2OS1 Access Control to Radiologically Significant Areas

2005 Point Beach Mid-Cycle, Integrated Self-Assessment; dated February 14 - 25, 2005
Airborne Radioactivity Survey; U2 CTMT Lower Cavity; dated April 9, 2005
Apparent Cause Evaluation (ACE) 001742; Adverse Trend in Procedure Use and Adherence in RP Department; dated May 21, 2004.
CAP 056828; Adverse Trend in Procedure Use and Adherence in RP Department; dated May 19, 2004.
CAP 058476; HP 2.6 Procedure Violation; dated August 13, 2004
CAP 059159; Benchmarking in RP; dated September 13, 2004
CAP 062161; Security Officers Receiving Unusual Dose Readings; dated February 18, 2005
CAP 062253; RCE 268 Scored at 72 by CARB; dated February 23, 2005
CAP 062435; Human Performance Clock not Improving in Radiation Protection Group; dated March 2, 2005
CAP 062633; Less than Adequate Corrective Actions Cause Repeat Occurrence of Deficiency; dated March 10, 2005
CAP 062922; Respiratory Protection Equipment Issued without Using HPIP 4.58; dated March 21, 2005
CAP 063613; Radworker Signed onto RWP with Wrong Badge; dated April 12, 2005
HP 2.6; Locked and VHRA Key Control; dated April 1, 2005
HPIP 4.40; TEDE ALARA Reviews; dated February 5, 2002
NP 4.2.1; ALARA Program; dated March 30, 2005

NP 4.2.19; General Rules for Work in a Radiologically Controlled Area; dated April 28, 2004
NP 4.2.20; RWP; dated September 22, 2000
Radiological Surveys; Location: Unit 2 Seal Table; dated April 12, 2005
RCE 000268; HP 2.6 Procedure Violation; dated December 17, 2004
RWP 05-213; Seal Table Activities; dated March 11, 2005
RWP 05-224; RCP Maintenance; dated March 11, 2005
RWP 05-227; Containment - Mechanical Valve Maintenance; dated March 18, 2005
RWP 05-230; Regen HX Cubicle Maintenance; dated March 11, 2005
RWP 05-248; PAB General Maintenance; dated March 11, 2005
RWP 05-249; PAB Mechanical Valve Maintenance; dated March 30, 2005
RWP 05-260; U2 Rx Head Replacement Activities; dated March 30, 2005
SnapShot Self-Assessment Report; Plant Operations/Radiation Protection, NRC RP Outage Access Control/ALARA; dated April 4, 2009

2OS2 As Low As Is Reasonably Achievable (ALARA) Planning And Controls

CAP 062473; Spring 2005 RV Head Replacement Inadequate RP and Related Outage Preps; dated March 3, 2005
Job File 120; RCP Seal Removal and Decon; dated March 2004
Job File 179; Radiation Protection Reactor Head Replacement; dated 2005
Job File 181; Reactor Head Replacement Cutting, Welding, and Burning Guideline; dated 2005
Level 3 Pre-Job ALARA Review Nos. 2005-001; 2005-009; 2005-010; and 2005-011
RWP 05-217; Split Pin Activities; dated March 17, 2005

4OA2 Identification and Resolution of Problems

AR Number: CAP055204; 2T34A SI Accumulator Level Instruments Returned to Service Without Proper Post Maintenance Testing; Attachment P, Effectiveness Review; 12/16/04
CA057853; (TRP) RCE251 - CA#9 - Electrical and Mechanical Maintenance Procedure PMT Review; June 1, 2004
CAP064647; Repeat and Rework Required on 2LT-938 and 2LT-939; May 18, 2005
EFR057297; (CARB) RCE251 CATPR#2 Effectiveness Review - Tool Pouch Maintenance; April 29, 2004
EFR057296; RCE251 Effectiveness Review of CA57290; April 29, 2004; Due Date, September 15, 2005
CA057295; RCE251 CA#8 - Transmitter Training; April 29, 2004
CA057294; RCE251 CA#7 - Unit 2 SI Accumulator Level Transmitter Configuration; April 29, 2004
CA057293; RCE251 CA#6 - 2T-34A and B Level Transmitter Loop Drawing; April 29, 2004
CA057291; RCE251 CATPR#2 - NP 10.2.9 Revision; April 29, 2004
CA057290; (CARB) RCE251 CATPR#1 - I&C Calibration Procedures; April 29, 2004
CA056999; RCE251 Interim Corrective Action - U1R28 Instrument Cal Procedure Review; April 9, 2004
CA056975; RCE251 - Interim Corrective Action - 0I-100 Revision; April 8, 2004

CA056949; RCE251 Interim Action - NP 10.2.9 Tool Pouch Maintenance Procedure Revision; April 8, 2004
CA056932; RCE251 Interim Corrective Action - I&C PMT Expectations; April 7, 2004
CA056975; RCE251 - Interim Corrective Action - OI-100 Revision; April 8, 2004
CA055415; Organizational Response to Unit 2 SI Accumulator Level Transmitter Issues; April 7, 2004
CA056651; Vendor Questions the Application and Configuration of His Equip Installed at PB; March 23, 2004
CA056782; Nuclear Safety Significance of A SI Accumulator Level Out of Specification High; March 31, 2004
CAP051028; Unit 2 Accumulator Level Instruments Found Out of Tolerance; October 14, 2003
NP 10.2.9; Tool Pouch Maintenance; Revision 3; July 14, 2004
RCE 251, 2T-34A SI Accumulator Level Instruments Returned to Service Without Proper Post Maintenance Testing; Revision 1; May 7, 2004
HPIP 11.54; Control Room F-16 Filter Testing; September 17, 2004
NMC letter to NRC; Generic Letter 2003-01: Control Room Habitability - Supplemental Response and Commitment Change; September 29, 2004
NP 1.1.12; Operational Decision-Making Issue Evaluation Process; October 13, 2004
Nuclear Plant Memorandum NPM 2004-836; Minutes from the December 1, 2004 CARB [Corrective Action Review Board] Meeting (Team B); December 1, 2004
CA060496; CATPR [Corrective Action to Prevent Recurrence] 1 From RCE000270 Revise HPIP 11.54; November 17, 2004
CA060498; Corrective Action 7 From RCE000270 - Revise Tech Spec Bases; November 17, 2004
CA060501; CATPR 3 For RCE000270 - Develop Tabletop Critique of Event and Present to Site Management; November 17, 2004
CA060506; CA #10 from RCE000270 - Develop Formal Guidance on How to Respond; November 17, 2004
CA060507; CA 12 for CC4 from RCE000270 - Revise NP 1.1.7; November 17, 2004
CA060509; CATPR 2 from RCE000270 - Require More Frequent Preventive Maintenance; November 17, 2004
CA060510; CA 1 from RCE000270 - Create Pre-Job Brief to Identify Internal OE [Operating Experience] for PM [Preventive Maintenance] Evolution; November 17, 2004
CAP058833; F-16 CR Filter Flow Low Out of Specification per HP 11.54; August 27,2004
CAP058976; Discrepancy Between CREFS SR 3.7.9.6 and FSAR 14.3.5 Assumption for Flow; September 3, 2004
CAP060646; RCE 270 Personnel Interview Inadequacy; November 19, 2004
CAP061388; Draft Corrective Action of RCE 270 Not Included for CARB Review; January 10, 2005
CE014450; Condition Evaluation - Redundant NPs Concerning Issue Management; August 18,2004
CE014962; Corrective Action 6 From RCE000270 - Determine Whether Mod to System is Necessary; November 17, 2004
CE015161; Draft Corrective Action of RCE 270 Not Included for CARB Review; January 12, 2005
Procedure Change Request PCR05924; Redundant NPs Concerning Issue Management; September 17, 2004

RCE000270; F-16 CR Filter Flow Low Out of Specification per HP 11.54;
August 31, 2004
CAP065291; Engineering Evaluation Needed for 2LT-934, 935, AND 938;
June 23, 2005
Operational Decision Making Issue Evaluation Document; CAP064638 May 25, 2005
CAP064638; Independent Verification of the Unit 2 ‘B’ SI Accumulator Level;
May 17, 2005
CA063122; Independent Verification of the Unit 2 ‘B’ SI Accumulator Level;
May 19, 2005
CAP065355; NRC Resident’s Questions Concerning the U2 SI Accumulator Levels;
June 26, 2005
CAP065291; Engineering Evaluation Needed for 2LT-934, 935 and 938; June 23, 2005

4OA5 Other Activities

RMP 9118-1; Containment Building Crane OSHA Operability Inspections; Revision 4
RMP 9118-2; Containment Building Crane Inspections; Revision 2
SLP-2; Items Lifted by Containment Polar Crane Unit 2 with Approval Sheets; Revision 15
Bigge Calculation 2103-C9.1; Point Beach Unit 2 Runway Loading Effects on Floor Elev. 21"; Revision 0
MRS-SSP-1663; RRVH Pre-Outage Movements to Auxiliary Building Truck Bay and Shipping Container Removal - PB2; Revision 1
MRS-SSP-1694; CRDM Cooling System Flow Test at PB Unit 2; Revision 1
MRS-SSP-1703; Radiation Shield Assembly Installation at PB Unit 2; Revision 1
MRS-SSP-1704; RRVH RVLIS/RCGVS Installation at PB Unit 2; Revision 1
MRS-SSP-1706; Install/Remove Bigge Runway System at PB Unit 2; Revision 1
MRS-SSP-1710; RVH Prepare for Disposal and Rig Out of CTMT at PB Unit 2; Revision 1
MRS-SSP-1711; RVH Load Out at PB Unit 2; Revision 1
MRS-SSP-1712; RRVH Transport and Rig into Containment at PB Unit 2; Revision 1
MRS-SSP-1714; HAUP Component Installation at PB Unit 2; Revision 1
NSD-EIS-97-003; CRDM and Analog Rod Position Indication Testing; Revision 3
Bigge Procedure 2100-P7; Procedure for Load Tests of Kewaunee and Ginna Upending/Downending Frames and Spreader Bar SB-224 as Lift Rigging in Vertical Position; Revision 1
Bigge Rigging Personnel Certificates from Training
03E39 - Dwg 42; Sheet 3 of 7; Elevation - Placing Old Head on Stand Handle Duratek Container Components; Revision 0
03E39 - Dwg SK30-1a-Aug 16; Sheet 1 of 2; Runway Plan at Hatch RRVH Project; Revision A
03E39 - Dwg SK30-2a-Aug 16; Sheet 2 of 2; Runway Elevations at Hatch RRVH Project; Revision A
03E39 - Dwg SK04-02-13-01 - Sheet 1 of 3; Plot Plan RRVH Project; Revision 1
03E39 - Dwg SK04-02-13-01 - Sheet 2 of 3; Equipment Hatch Plan View RRVH Project; Revision 0
03E39 - Dwg SK04-02-13-01 - Sheet 3 of 3; Equipment Hatch Section RRVH Project; Revision 0

NP 7.4.3; Post-Maintenance and Modification NDE Requirements for Power Piping; Revision 5
RESP 3.1; Primary System Tests; Revision 21
MHI Procedure UGS-L5-040144; Hydrostatic Test Procedure; Revision 1
IWP 03-058*A; Installation of New RV Head and HAUP Unit 2; Revision 0
MR 03-056; Mod Description
MR 03-058; Mod Description
AOP-0.1; Declining Frequency on 345 Kilo Volt Distribution System; April 18, 2005
Emergency Contingency Action ECA-0.1 Unit 1; Loss of All AC Power Recovery Without SI Required; May 16, 2005
Emergency Contingency Action ECA-0.0 Unit 1; Loss of All AC Power; March 10, 2005
Emergency Contingency Action ECA-0.0 Unit 2; Loss of All AC Power; March 10, 2005
NP 2.1.5; Electrical Communications, Switchyard Access and Work Planning; December 8, 2004
NP 10.2.2; Scheduling, Planning and Implementing On-Line Work; April 13, 2005
NP 10.3.6; Shutdown Safety Review and Safety Assessment; November 24, 2004
NP 10.3.7; On-Line Safety Assessment; May 11, 2005
OI 110; Gas Turbine Operation; January 6, 2005
OP 2A; Normal Power Operation; April 21, 2005
TS 81; Emergency Diesel Generator G-01 Monthly; May 19, 2005
Point Beach Form PBF-1658; Safety Monitor Change Notice; February 11, 2005
Point Beach Form PBF-9814; Risk Management Actions for Unplanned Red or Yellow CDF Level; November 12, 2003

Replacement Reactor Vessel Closure Head (71007)

Modification No. 03-056; Point Beach Unit 2; RVH Replacement Project; Revision 0
Design Specification No. 414A83; Point Beach Units 1 and 2, Replacement Reactor Vessel Head (RRVH); Revision 2
Design Specification No. 414A86; Point Beach Units 1 and 2, CRDM Model L106A; Revision 2
Design Specification No. NSD-ENG-DS-327; Point Beach Units 1 and 2, Core Exit Thermocouple Nozzle Assembly (CETNA); Revision 0
Document L5-01DL511; Point Beach Unit 2, Replacement Reactor Vessel Closure Head, Design Report L5-01DL510; Revision 2, Addendum; Revision 1
Document L5-01DL510; Point Beach Unit 2, Replacement Reactor Vessel Closure Head, Design Report; Revision 2
WCAP-16266-P; Point Beach Unit 2, Replacement Reactor Vessel Closure Head - Design Report; Revision 1, Addendum 2; December 2004
WCAP-16266-P; Point Beach Unit 2, Replacement Reactor Vessel Closure Head - Design Report; Revision 1, Addendum 1; November 2004
WCAP-16266-P; Point Beach Unit 2, Replacement Reactor Vessel Closure Head - Design Report; Revision 1
Calculation Note No. CN-RCDA-03-135; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - Analysis Procedure; Revision 2
Calculation Note No. CN-RCDA-04-63; Point Beach Units 1 and 2, RRVCHs, ASME Section XI Code Reconciliation; Revision 2
Calculation Note No. CN-RCUFW-04-7, Point Beach Unit 2, Replacement Head Project - Closure Head Flange ASME Code and Leakage Evaluation; Revision 0

Calculation Note No. CN-RCDA-03-118; Kewaunee, Replacement Reactor Vessel Closure Head, Finite Element Model of CRDM Head Adapters; Revision 0
Calculation Note No. CN-RCDA-03-119; Kewaunee, Replacement Reactor Vessel Closure Head, ANSYS Thermal and Structural Analyses of CRDM Head Adapters; Revision 1
Calculation Note No. CN-RCDA-03-120; Kewaunee, Replacement Reactor Vessel Closure Head, CDRM Head Adapter ASME Code Evaluation; Revision 0
Calculation Note No. CN-RCDA-04-44; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - CRDM Head Adapter ASME Code Evaluation; Revision 1
Calculation Note No. CN-RCDA-04-78; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - Closure Head Lifting Lug Stress Analysis; Revision 3
Calculation Note No. CN-RCDA-04-02; Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head - Vent Pipe ASME Code Evaluation; Revision 1
Calculation Note No. CN-PAFM-04-59; Point Beach Unit 2, Replacement Reactor Vessel Closure Head, Fracture Evaluation; Revision 0
Document No. KBS-20040428; Point Beach Unit 2, Justification for Nonconformance Reports of Replacement Reactor Vessel Closure Head; Revision 3
Document No. L5-01DE906; Kewaunee and Point Beach Units 1 and 2, Replacement Reactor Vessel Closure Head, Primary Stress Evaluation for Head Adapter; Revision 1
Document No. L5-01DL501; Point Beach Unit 2, Basic Sizing Report of Reactor Vessel Closure Head; Revision 3
Document PB-KCS-04-0009; Point Beach Units 1 and 2, CRDM, Design Report PB KCS-04-0002; Revision 2, Addendum; Revision 1
Document PB-KCS-04-0002; Point Beach Units 1 and 2, CRDM, Design Report; Revision 2
WCAP-16267-P; Point Beach Units 1 and 2, Replacement CRDM, Design Report; Revision 0, Addendum 2; December 2004
WCAP-16267-P; Point Beach Units 1 and 2, Replacement CRDM, Design Report; Revision 0, Addendum 1; November 2004
WCAP-16267-P; Point Beach Units 1 and 2, Replacement CRDM, Design Report; Revision 0
Calculation Note No. CN-ENG-04-6; Point Beach Units 1 and 2, CRDM - Analysis Procedure; Revision 1
Calculation Note No. CN-ENG-04-37; Point Beach, CRDM - ASME Section XI Reconciliation; Revision 1
Calculation Note No. CN-ENG-04-33; Point Beach, CRDM - Tentative Pressure Thickness Calculations Per NB-3324; Revision 0
Calculation Note No. CN-ENG-04-20; Point Beach, CRDM - Pressure Housing ASME Qualification; Revision 1
Document No. PB-KCS-04-003; Point Beach Unit 2, Justification for Nonconformance Reports of Replacement CRDM; Revision 3
Document No. PB-KCS-04-008; Point Beach Units 1 and 2, CRDM, Additional Reconciliation of Design Report with Latest Drawing Revision; Revision 2
MHI Drawing L5-01DL101; Replacement Reactor Vessel Closure Head, Closure Head General Assembly ½; Revision 2
MHI Drawing L5-01DL102; Replacement Reactor Vessel Closure Head, Closure Head General Assembly 2/2; Revision 1

MHI Drawing L5-01DL103; Replacement Reactor Vessel Closure Head, Closure Head Welding ½; Revision 0
MHI Drawing L5-01DL104; Replacement Reactor Vessel Closure Head, Closure Head Welding 2/2; Revision 1
MHI Drawing L5-01DL107; Replacement Reactor Vessel Closure Head, Closure Head Penetration Position ½; Revision 1
MHI Drawing L5-01DL108; Replacement Reactor Vessel Closure Head, Closure Head Penetration Position 2/2; Revision 1
MHI Drawing L5-01DL109; Replacement Reactor Vessel Closure Head, Closure Head and Adapter Housing Assembly; Revision 2
MHI Drawing L5-01DL110; Replacement Reactor Vessel Closure Head, Instrumentation Port Head Adapter ½; Revision 7
MHI Drawing L5-01DL111; Replacement Reactor Vessel Closure Head, Instrumentation Port Head Adapter 2/2; Revision 7
MHI Drawing L5-01DL124; Replacement Reactor Vessel Closure Head, Closure Head Name Plate; Revision 0
MHI Drawing L5-01DL151; Replacement Reactor Vessel Closure Head, Closure Head Marking Procedure Drawing; Revision 1
MHI Drawing L5-01DL171; Replacement Reactor Vessel Closure Head, As-Built Drawing (RV Closure Head) 1/3; Revision 3
MHI Drawing L5-01DL172; Replacement Reactor Vessel Closure Head, As-Built Drawing (RV Closure Head) 2/3; Revision 4
MHI Drawing L5-01DL173; Replacement Reactor Vessel Closure Head, As-Built Drawing (RV Closure Head) 3/3; Revision 4
MHI Drawing L5-03BJ101; Reactor Vessel Closure Head, General Assembly Drawing; Revision 4
MHI Drawing L5-03BJ102; CRDM, General Assembly; Revision 2
MHI Drawing L5-03BJ103; CRDM, Pressure Housing Assembly; Revision 2
MHI Drawing L5-03BJ211; CRDM, Rod Travel Housing; Revision 3
MHI Drawing L5-03BJ212; CRDM, One-Piece Latch Housing; Revision 4
Reactor Vessel Closure Head and CRDMs; ASME Normal Pressure and Temperature Component Certification; Mitsubishi Heavy Industries, Ltd; dated November 25, 2004

<u>Head Assembly Upgrade Package</u> (71007)

Modification No. 03-058; Point Beach Unit 2; Install HAUP; Revision 0
Design Specification No. 418A27; Point Beach Units 1 and 2, HAUP; Revision 5
Calculation Note No. CN-RVHP-04-3; Point Beach HAUP Plenum Stress Qualification; Revision 3
Calculation Note No. CN-RVHP-04-3; Point Beach HAUP Plenum Stress Qualification; Revision 6
Calculation Note No. CN-RVHP-04-6; Point Beach HAUP Cable Bridge Structural Analysis; Revision 2
Calculation Note No. CN-RVHP-04-8; Point Beach HAUP Weight, Center of Gravity and Levelness Calculation; Revision 3
Calculation Note No. CN-RVHP-04-9; Point Beach HAUP Head Lift Rig, Leg Extension and Spreader Evaluation; Revision 2
Calculation Note No. CN-RVHP-04-10; Point Beach Units 1 and 2 HAUP - Missile Impact Analysis; Revision 1

Calculation Note No. CN-RVHP-04-12; Point Beach HAUP Structural Analysis; Revision 0
Calculation Note No. CN-RVHP-04-16; Point Beach HAUP Structural Analysis of the RVLIS and RCGVS Supports; Revision 2
Calculation Note No. CN-RVHP-04-44; Point Beach Unit 1 and 2 HAUP Cooling Shroud Structural Analysis; Revision 0
Calculation Note No. CN-RVHP-04-44; Point Beach Unit 1 and 2 HAUP Cooling Shroud Structural Analysis; Revision 3
Calculation Note No. CN-RVHP-04-50; Point Beach APRI Top Plate Stress Qualification; Revision 1
Calculation Note No. CN-RVHP-05-6; Point Beach Units 1 and 2 RCGVS and RVLIS ASME Section XI Code Reconciliation; Revision 0
Drawing 1C51902; Point Beach Units 1 and 2, HAUP, Ring Girder Support Column - Missile Shield; Revision 0
ECN-05-261; Weld Size Change to Drawing 1C51902; Revision 0
Serial No. NES-CSSTK; Penn State Tool & Die - Welding Procedure; Revision 3
CAP015659; Polar Crane Capacity Questioned During Head Lift; January 23, 1995
CAP064605; Nuclear Safety Concern Raised Regarding Going to Reduced RV Inventory; May 16, 2005
CAP064345; Preliminary Lessons Learned - Exigent LAR for RVH Lift; May 4, 2005
CAP063860; Unit 2 RHR Flow Secured during Hanging of Tagout; April 19, 2005
CAP063731; Administrative Deficiencies with Heavy Load Procedures; April 14, 2005
CAP063730; Review of Station’s Response to NUREG-0612; April 14, 2005
CAP063450; NUREG-0612 Information Not Fully Incorporated into FSAR; April 7, 2005
CA062521; NUREG-0612 Information Not Fully Incorporated into FSAR; April 9, 2005
CAP063989; Z-013 Polar Crane Bridge Failure; April 23, 2005
CAP063967; Reactor Upper Vessel Internals Drop Analysis; April 22, 2005
CAP063972; AOP-13C Entry; April 22, 2005
CAP063983; Safety Issue Raised During Split Pin Preparation Work at the Refueling Cavity; April 22, 2005
CAP063536; Unable to Meet NUREG-0612 Phase II Requirements for Head Drop Analyses; April 9, 2005
CAP027500; Scaffolding in Unit 1 Containment Struck by Polar Crane; April 18, 1994
OE047478;Electrolytic Capacitors Leak Due to Containment Pressure; June 15, 2000
CAP062420; Due Date on DIA Between PBNP & Whiting Services Was Not Met; March 1, 2005
ACE000736; Load Being Lifted Out of the U2 RV Cavity Dropped; May 6, 2002
ACE000935; Unsafe Rigging Practices; September 23, 2002
CE006862; Scaffolding in Unit 1 Containment Struck by Polar Crane; April 19, 1994
CE013891; Lack of Understand/Commitment/Adherence to Formal Administrative Controls; April 23, 2004
CAP003089; Polar Crane Main Hook Damaged Pipe Insulation Above B-RCP Cubicle; April 28, 2002
CAP003163; Load Being Lifted Out of the U2 RV Cavity Dropped; May 3, 2002
CAP006816; Welding Requirements Not Met - SGT; November 2, 1996
CAP015653; Crane Hooks and Rigging Devices Do Not Meet Safety Manual 25.5.3 Requirement; January 17, 1995
CAP009767; Unit 2 Polar Crane Found to Have A Ground Fault; December 12, 1998
CAP059870; U2R27 Review of OE for Containment Cranes; October 14, 2004

CAP060827; U2R27 Polar Crane Upgrade Schedule; December 1, 2004
CAP045808; NRC Information Notice 96-26; Recent Problems With Overhead Cranes; April 30, 1996
CAP055909; NP 7.2.10 Engineer Eval Prep, Review, and Approval Not Used for U1 Polar Crane; April 20, 2004
CAP060205; Continued Quality Issues Associated with Vendor Supplied Crane Drawings; October 28, 2004
CAP061584; No-Sway System for Z-14 and Z-15 Cranes Not Usable in Bridge Mode; January 20, 2005
CAP 063508; SLP 2 Did Not Support Planned Lifts for Assembly of Temp Gantry Crane; April 9, 2005
ACE000712; Polar Crane Main Hook Damaged Pipe Insulation Above B-RCP Cubicle; April 30, 2002
CAP063546; Drawing and wiring Discrepancies in Breaker Cubicle 2A52-47; April 10, 2005
CAP061113; Wooden Equip Storage Crates Located in FZ 238 (Gas Stripper Rm); December 16, 2004
CAP058817; FSAR Documentation on AFW may be Inaccurate; August 27, 2004
CAP061109; Procedural Guidance Needed to Comply with TMI Commitment; December 16, 2004
CAP050483; FP Supply to 1/2P-29 (TDAFP) Bearing Coolers During SBO; September 28, 2003
CAP058817; FSAR Documentation on AFW may be Inaccurate; August 27, 2004
CEO014533; FSAR Documentation on AFW may be Inaccurate; August 31, 2004

Implementation of RVH Modification

Unit 2 RVH Background Information
10 CFR 50.59/72.48 Screening; SCR 2005-0125; Revision 10 to SLP 8 Synthetic Webbing Sling Sizing
Derived Licensing Basis for Control of Heavy Loads Literature
Work Order 0415356; 2R-1 NDE BMI Penetrations - U2R27
RCE Charter; CAP63088; RCE 276
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LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
AISC	American Institute of Steel Construction
ALARA	As Low As Reasonably Achievable
AMSAC	ATWS (anticipated transient without scram) mitigating system actuation circuitry
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
CAP	Corrective Action Program Document
CFR	Code of Federal Regulations
CL	Checklist
CRDM	Control Rod Drive Mechanism
CCW	Component Cooling Water
DBD	Design Basis Document
FSAR	Final Safety Analysis Report
HAUP	Head Assembly Upgrade Package
HPIP	Health Physics Implementing Procedure
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
MCC	Motor Control Center
NCV	Non-Cited Violation
NP	Nuclear Plant Procedures Manual Procedure
NRC	Nuclear Regulatory Commission
OPR	Operability Evaluation
OSP	Offsite Power
PBNP	Point Beach Nuclear Plant
PWR	Pressurized Water Reactor
RCE	Root Cause Evaluation
RCGVS	Reactor Coolant Gas Vent System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMP	Routine Maintenance Procedure
RPT	Radiation Protection Technician
RRVH	Replacement Reactor Vessel Head
RVH	Reactor Vessel Head
RVLIS	Reactor Vessel Level Indication System
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
SEP	Shutdown Emergency Procedure
SI	Safety Injection
SW	Service Water
TS	Technical Specifications
TSO	Transmission System Operator
U2R27	Unit 2 Refueling Outage 27
URI	Unresolved Item
VHRA	Very High Radiation Area