January 24, 2001

Mr. M. Reddemann
Site Vice President
Kewaunee and Point Beach Nuclear Plants
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT - NRC INSPECTION REPORT 50-266/00-17(DRP); 50-301/00-17(DRP)

Dear Mr. Reddemann:

On December 31, 2000, the NRC completed a baseline inspection at your Point Beach Nuclear Plant. The results of this inspection were discussed with Mr. A. Cayia and members of your staff on January 3, 2001. The enclosed report presents the results of that inspection.

This 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 personnel.

Based on the results of this inspection, the inspectors identified three issues of very low safety significance (Green). These issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Point Beach facility. Additionally, the inspectors identified one issue that was determined to be a No Color finding.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you chose to provide one, 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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Roger D. Lanksbury, Chief
Reactor Projects Branch 5

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

Enclosure: Inspection Report 50-266/00-17(DRP);
50-301/00-17(DRP)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-266; 50-301
License Nos:	DPR-24; DPR-27
Report No:	50-266/00-17(DRP); 50-301/00-17(DRP)
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Plant, Units 1 & 2
Location:	6610 Nuclear Road Two Rivers, WI 54241
Dates:	November 10 - December 31, 2000
Inspectors:	J. Lara, Senior Resident Inspector, Kewaunee R. Powell, Resident Inspector K. Riemer, Regional Inspector D. Pelton, Operator Licensing Examiner M. Kunowski, Regional Inspector
Approved by:	Roger D. Lanksbury, Chief Reactor Projects Branch 5

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 05000266-00-17, IR 05000301-00-17, on 11/10-12/31/2000, Nuclear Management Company, LLC, Point Beach Nuclear Plant, Units 1 & 2. Adverse weather protection, non-routine plant evolutions, refueling and outage, cross-cutting issues.

The inspection was conducted by the resident inspectors and regional inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the Significance Determination Process does not apply are indicated by "no color" or by the severity level of the applicable violation.

A. Inspector-Identified Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for an inadequately written work instruction that did not provide for appropriate isolation of inverter 1DYO3 which resulted in de-energization of the Unit 1 white instrument bus and a subsequent plant transient.

  This finding was of very low safety significance because all mitigation systems remained operable and barrier integrity was not challenged. (Section 1R14.2).

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for an inadequate procedure which resulted in an inadvertent decrease in reactor coolant system inventory during reactor coolant pump coupling while in cold shutdown.

  The finding was of very low safety significance because residual heat removal was not impacted and the amount of water that could have been drained from the reactor coolant system was limited by system configuration and alignment. (Section 1R20).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for an inadequate procedure that specified actions that inappropriately de-energized heat trace circuits for safety-related equipment when the intent was only to bypass alarms.

  The finding was of very low safety significance because safety-related equipment was not actually rendered inoperable. (Section 1R01).

Cross-cutting Issues: Human Performance

- No Color. The inspectors determined that a negative performance trend had developed in several cornerstone areas with procedure inadequacy being the common element based on two examples identified during this reporting period and two previously identified examples of inadequate procedures. All four examples related to the licensee development, technical review, and approval of procedures.

  While the risk of the individual examples was very low, the licensee had failed to ensure that procedures were correct prior to being approved for use. These findings collectively indicated a problem with the licensee's human performance in the area of procedure development, technical review, and approval. (Section 4OA4).

REPORT DETAILS

Summary of Plant Status: Unit 1 operated at 100 percent throughout the inspection period except for November 20-22, 2000, when power was reduced to 90 percent during repairs to an offsite power line.

Unit 2 began the inspection period in its Cycle 24 refueling outage (which began on October 13). On December 14, Unit 2 was made critical after completion of outage activities. Approximately an hour later, the reactor tripped when control power fuses blew for one of the intermediate range nuclear instruments.

Unit 2 was again made critical later that day after corrective actions. On December 16, the Unit was synchronized to the offsite electrical distribution grid. On December 17, with the reactor at 28 percent power, the main turbine tripped during testing of a recently installed voltage regulator when a trip setpoint for the generator stator ground system was inadvertently reached during the testing. As designed, the reactor did not trip.

Following corrective actions for the turbine trip, Unit 2 was re-synchronized to the grid and reactor power was maintained less than 50 percent for the conduct of core physics testing, secondary-side chemistry holds, and emergent repairs to the steam generator level control system and for leaks from several pump casing bolts on the “A” steam generator feedwater pump. Early on the morning of December 20, with reactor power at 63 percent and increasing at 2 percent per hour, the reactor tripped (as designed) after the main turbine tripped because of the actuation of an overcurrent protective relay associated with the main electrical output transformer.

Subsequent to the completion of corrective actions for the December 20 turbine trip, Unit 2 was made critical around midday on December 21 and was synchronized to the grid later that day. Reactor power was increased until 100 percent power was reached on December 26. Reactor power remained at 100 percent until the end of the inspection period, except for a few hours on December 31, when power was reduced to 94.5 percent for condenser steam dump testing.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspection verified the design features and implementation of the licensee’s procedures to protect mitigating systems from adverse weather (extreme cold) effects. The inspectors focused on the Unit 1 and Unit 2 safety injection systems and the G-03 and G-04 emergency diesel generators (EDGs). During this inspection, the inspectors verified that cold weather protection features such as heat tracing (the facade freeze protection system) and space heaters were monitored and functional; that plant features and procedures for use of the ultimate heat sink (Lake Michigan) were appropriate; that operator actions specified in the licensee's cold weather preparation procedures ensured the readiness of essential systems; and that problems that could affect the functioning of mitigating systems and their support systems during extended cold weather were entered into the licensee's corrective action program for evaluation and resolution. As part of this effort, the inspectors accompanied a non-licensed operator during the operator's check of the preparedness of plant areas and equipment for cold weather in November, and reviewed the licensee's response to a sudden onset of severe cold weather around December 5 and 6. The inspectors also reviewed the following documents:

- Condition Report (CR) 99-2654, Discrepancy in current readings for reactor water storage tank and reactor makeup water tank heaters
- CR 00-0089, Safety injection piping in containment tendon gallery not insulated or heat traced
- CR 00-0213, Low forebay level due to ice buildup on intake structure
- CR 00-4046, Numerous problems were encountered with freeze protection
- Operating Instruction (OI) 38, "Circulating Water System Operation," Revision 21
- OI 106, "Facade Freeze Protection," Revision 15
- Plant Modification No. 99-019, "PBNP [Point Beach Nuclear Plant] Freeze Protection Upgrade"
- Inspection Report 50-266/99019; 50-301/99019, Section O2.2, <u>Cold Weather Preparations</u>
- Periodic Check (PC) 49, "Cold Weather Preparations," Revision 1
- PC 49, Part 3, "Auxiliary Building Ventilation," Revision 8
- PC 49, Part 4, "Auxiliary Building Miscellaneous and Facades," Revision 14
- PC 49, Part 5, "Cold Weather Checklist: Outside Areas and Miscellaneous," Revision 13
- "Cold Weather Preparations," Organizational Assessment (Quality Assurance) Surveillance Report S-P-99-16
- Abnormal Operating Procedure (AOP) 13A, "Circulating Water System Malfunction," Revision 9
- Work Order (WO) Plan 9926728, Install design parameters and functional testing of [facade freeze protection circuits] 2FFCP-01A and 2FFCP-01B

b. Findings

One finding of very low safety significance was identified by the inspectors for an inadequate procedure for the operation of the full-scale replacement modification of the freeze protection system. The sudden, severe cold weather in early December resulted in the freezing of water in several small diameter pipes and tubing of nonsafety-related equipment and caused the licensee to hasten the post-modification fine tuning of the recently installed, complete replacement of the facade freeze protection (heat trace) system. In response to the severe cold weather, the licensee identified that several heat trace circuits had been unintentionally de-energized when operators had bypassed several system alarms. This finding, that bypassing alarms also de-energized the heat trace circuits, if left uncorrected, could have resulted in the freezing of safety-related equipment, if alarms had occurred associated with that equipment and had been bypassed, and could have rendered that equipment inoperable for accident mitigation.

This finding did have a credible impact on safety. However, since no heat trace circuits for safety-related equipment were de-energized and no safety-related equipment actually froze, the finding was considered to be of very low safety significance (Green). Criterion V, “Instructions, Procedures, and Drawings,” of Appendix B of 10 CFR Part 50, required that activities affecting quality be prescribed by documented instructions, procedures, or drawings. Procedure OI-106, “Facade Freeze Protection,” Revision 15, was inadequate in that it specified actions that inappropriately de-energized heat trace circuits for safety-related equipment when the intent was only to bypass alarms. The inadequate procedure is being treated as a Non-Cited Violation (NCV 50-266/00-17-01(DRP); 50-301/00-17-01(DRP)), consistent with Section VI.A.1 of the NRC’s Enforcement Policy. An inadequate understanding of the operation of the new freeze protection system was the apparent cause of the inadequate procedure. Unlike the old system, bypassing alarms in the new facade freeze protection system also de-energized the heat trace circuits. As documented in CR 00-4046, the procedure was revised to correct the inadequacy.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial walkdown of the Unit 2 safety injection system to verify that valves were in the proper position to perform their safety-related function. Instrumentation valve configurations and appropriate meter indications were also observed. The inspectors also evaluated other conditions such as component material condition, adequacy of housekeeping, and proper component labeling. This system was selected based upon its high risk significance and change in plant conditions associated with the end of outage activities. The inspectors reviewed Operations Checklist (CL) 7A, “Safety Injection System Checklist Unit 2,” Revision 15, as part of the inspection.

b. Findings

The inspectors identified 16 valves that were not in the position specified by CL 7A. The valves were locked shut vice shut, but not locked, as required by procedure. The inspectors previously identified a similar issue in Inspection Report 50-266/00-09(DRP); 50-301/00-09(DRP). Because the condition was identified late in the inspection period, the inspectors considered this issue to be an unresolved item (URI 50-301/00-17-02(DRP)) pending additional inspection effort and review of the risk significance.

1R05 Fire Protection

a. Inspection Scope

The inspectors walked down the following areas to assess the overall readiness of fire protection equipment and barriers:

- Unit 2 Containment 8' Elevation, Fire Zone 608
- Unit 2 Containment 21' Elevation, Fire Zone 611
- Unit 2 Containment 46' Elevation, Fire Zone 615
- Unit 2 Containment 66' Elevation, Fire Zone 618

Emphasis was placed on the control of transient combustibles and ignition sources; the material condition of fire protection equipment; and the material condition and operational status of fire barriers used to prevent fire damage or propagation. Area conditions/configurations were evaluated based on information provided in the licensee’s “Fire Protection Evaluation Report,” August 1999.

The inspectors verified that fire hoses and portable fire extinguishers were installed at their designated locations, were in satisfactory physical condition, and were unobstructed. The inspectors also verified the physical location and condition of fire detection devices.

Additionally, the inspectors periodically walked down the following areas identified as “fire sensitive” during the Unit 2 outage to verify licensee identified compensatory actions and restrictions were maintained:

- Valve Pit/Sump Pump Room, Fire Zone 101
- Corridor [-5'-3" Sub-Basement], Fire Zone 113
- Component Cooling Water Pump Room, Fire Zone 142

The inspectors reviewed the following documents during the inspection:

- Fire Emergency Plan 4.7, “Containment Unit 2,” Revision 4

- PC-72, Part 1, “Monthly Surveillance of Controlled-Side Fire Extinguisher Equipment,” Revision 6

- Point Beach Form PBF-9700, “PBNP Shutdown Safety Assessment and Fire Condition Checklist,” Revision 4

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors performed quarterly observations of licensed operator simulator training. On November 21, the inspectors observed licensed operator training on recent revisions to the following procedures:

- Operating Procedure (OP) OP 1C, “Low Power Operation to Normal Power Operation,” Revision 73

- Emergency Operating Procedure (EOP) 0 Unit 2, “Reactor Trip or Safety Injection,” Revision 35

- EOP 0.1 Unit 2, “Reactor Trip Response,” Revision 22

- EOP 3 Unit 2, “Steam Generator Tube Rupture,” Revision 29

The inspectors verified crew performance in terms of clarity and formality of communication; the ability to take timely action in the safe direction; the prioritizing, interpreting, and verifying of alarms; the correct use and implementation of procedures, including alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; and the group dynamics.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule requirements to ensure that component and equipment failures were identified, entered into the calculations for unavailability, and scoped within the maintenance rule and that select structures, systems, or components were properly categorized and classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors also verified that issues were identified at an appropriate threshold and entered into the corrective action program. Additionally, the inspectors verified licensee changes to performance criteria were reflected in the licensee's probabilistic risk assessment.

Specific components or system problems evaluated were:

- Power-Operated Relief Valves

- Diesel Generator Room Heating and Ventilation

The inspectors reviewed various corrective action program documents (CRs), in addition to the following documents:

- Calculation 98-0169, "PRA Assessment of MR APC and RPC," Revision 1 [Probabilistic risk assessment of maintenance rule availability performance criteria and reliability performance criteria]

- "1999 Annual Report for the Maintenance Rule," dated March 30, 2000.

- Nuclear Power Business Unit Procedure (NP) 7.7.4, "Scope and Risk Significant Determination for the Maintenance Rule," Revision 6

- NP 7.7.5, "Determining, Monitoring and Evaluating Performance Criteria for the Maintenance Rule," Revision 6

- Design Basis Document DBD-09, "Reactor Coolant System," Revision 1

Finally, the inspectors reviewed CR 00-4058, "Less Than Adequate Performance Criteria for Maintenance Rule," which was initiated as a result of this inspection activity and was reviewed as part of the inspection scope.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk, scheduling, configuration control, and performance of maintenance associated with planned and emergent work activities and verified that scheduled and emergent work activities were adequately managed. In particular, the inspectors reviewed the licensee's program for conducting maintenance risk safety assessments and verified that the licensee's planning, risk management tools, and the assessment and management of shutdown risk were adequate. The inspectors also verified that licensee actions to address increased shutdown risk during periods when equipment was out-of-service for maintenance, such as establishing compensatory actions, minimizing the duration of the activity, obtaining appropriate management approval, and informing appropriate plant staff, were accomplished when shutdown risk was increased due to maintenance on risk-significant SSCs. The following specific activities were reviewed:

- The inspectors reviewed the maintenance risk assessment for work planned for the week of November 13, 2000. This included work associated with the Unit 2 core reload, rod latching, and vessel head set.

- The inspectors reviewed the maintenance risk assessment for work planned for the week of November 27, 2000. This included work associated with the Unit 2 planned heatup and subsequent cooldown due to a secondary coolant (feedwater) leak.

- The inspectors reviewed the maintenance risk assessment for WO 9925139, "B-03 Infrared Buswork Inspection." The work created an increased risk configuration due to the Train "A" motor-driven auxiliary feedwater (AFW) pump, P38A, being out-of-service.

Additionally, the inspectors reviewed the following documents:

- NP 10.3.6, "Outage Safety Review and Safety Assessment," Revision 5

- NP 10.2.1, "Outage Planning, Scheduling, and Maintenance," Revision 10

- NP 10.3.7, "On-Line Safety Assessment," Revision 4

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 Unit 2 Secondary Coolant (Feedwater) Leak

a. Inspection Scope

The inspectors reviewed plant data and interviewed operators to evaluate the performance and response to a leak from steam generator 2B feedwater check valve 2CS-476BB identified during plant heatup on November 29, 2000. The inspectors verified actions were taken in a timely manner in accordance with AOP 2A, "Secondary Coolant Leak Unit 2," Revision 6.

b. Findings

No findings of significance were identified.

.2 Unit 1 120-Volt (V) Alternating Current Vital White Instrument Bus De-energization

a. Inspection Scope

The inspectors reviewed plant data and interviewed operators to evaluate the performance and response to an unplanned de-energization of the white channel instrument bus during emergent maintenance activities on December 2, 2000. The bus de-energization caused a mismatch between turbine first stage pressure and reactor power which caused control rod motion at maximum speed in the reactor core. The inspectors verified that control room reactor operator actions were taken in a timely manner in accordance with AOP 6C, "Uncontrolled Motion of RCCA(s) Unit 1," Revision 9 and AOP 24, "Response to Instrument Malfunctions," Revision 0.

b. Findings

An inadequate work plan caused Unit 1 control rods to rapidly move into the reactor core. The inspectors noted that prior to the transient initiation, the power for the white channel instrument busses had been switched from the safeguards supply to a non-safeguards supply because of maintenance problems earlier that evening. This resulted in Unit 1 entering an 8-hour limiting condition for operation, in accordance with Technical Specification 15.3.7.B.1.j. In an attempt to isolate inverter 1DY03 during the subsequent troubleshooting, a worker removed non-safeguards power from the instrument bus resulting in de-energization of the bus and the control rod movement.

The inspectors concluded that WO Work Plan WIT 167057 was inadequate in that it did not identify the appropriate isolation for maintenance activities on 1DY03. This finding had a credible impact on safety in that it initiated a primary plant transient and increased the likelihood of a reactor trip; however, since mitigation systems remained operable and barrier integrity was not challenged, the finding was considered to be of very low safety significance (Green). The inadequacy of the written instruction constituted a violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." However, because of the very low safety significance of the item and because the licensee had included this item in their corrective action program (CR 00-4026), this procedure violation is being treated as a Non-Cited Violation (NCV 50-266/00-17-03(DRP).)

.3 Unit 2 Reactor Trip During Initial Startup

a. Inspection Scope

The inspectors reviewed plant data and interviewed operators to evaluate the performance and response following an automatic reactor trip of Unit 2 during startup, from the refueling outage, on December 14, 2000, due to a blown control power fuse on intermediate range neutron detector 2N35. The reactor trip occurred about an hour after the reactor was taken critical. The inspectors verified that plant equipment operated as designed following the reactor trip. The inspectors also reviewed the licensee's post-trip review, conducted in accordance with licensee procedure NP 5.3.3, "Incident Investigation and Post-Trip Review," Revision 1.

b. Findings

No findings of significance were identified.

.4 Unit 2 Turbine Trip at 28 Percent Power

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the trip of the Unit 2 main turbine during testing of a recently replaced voltage regulator on December 17. The inspectors conducted the review to determine if equipment and personnel responded appropriately and if procedures were adequate and adhered to. As designed, the reactor, which was at 28 percent power, did not trip. The licensee's determined that the engineers who wrote the test procedure and reviewers and approvers of the procedure were unaware that the testing would result in the actuation of the stator ground relay trip circuit. Previous testing of the old voltage regulator had not resulted in relay actuation because the regulator's design did not allow testing as extensive as of that conducted on the new regulator. Corrective actions for this issue were being tracked in CR 00-4151.

b. Findings

No findings of significance were identified.

.5 Unit 2 Turbine and Reactor Trip at 63 Percent Power

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the trip of the Unit 2 main turbine and reactor on December 20. Unit 2 power was at 63 percent and increasing at the time of the trip. The inspectors conducted the review to determine if equipment and personnel responded appropriately and if procedures were adequate and adhered to. As part of the inspection effort, the inspectors interviewed plant staff and reviewed control room logs, plant process computer printouts, main control board indications and switch positions, and procedure NP 5.3.3, “Incident Investigation and Post-Trip Review,” Revision 1. The licensee’s initial evaluation identified that a crimped wire connection in a junction box on the “C” phase of the Unit 2 main transformer had failed causing the turbine trip which caused the reactor trip. A review by the licensee of outage activities identified that the box had been inspected in November for indications of oil and water leaks. It was not known if the inspection activities could have loosened the connection or if the inspection should have identified the failed connection. The licensee’s investigation and root cause evaluation of the trip was being conducted as part of CR 00-4185.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors verified that the subject operability evaluations addressed the applicable current licensing basis requirements and commitments, and provided an adequate basis for justifying operability. In those cases where an adequate written basis for justifying operability was not provided, the inspectors performed an independent review of the condition to assess whether a reasonable presumption of operability existed. The independent review included a discussion with licensee personnel and reviews of design and licensing basis documentation. The inspectors reviewed the following three operability evaluations:

- CR 97-0422, “Seismic II/I Issue (Demineralized Water Piping) for the CCW HX’s [component cooling water heat exchangers],” Operability Evaluation, Revision 0

- CR 98-18798, “Bearing Coolers for the Turbine-Driven Auxiliary Feedwater Pump (AFWP) Turbines Do Not Meet Design Pressure Requirements,” Operability Evaluation, Revision 1

- CR 00-3035, “Broken Weld on SI Recirc Line,” Operability Evaluation, Revision 0

b. Issues and Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

.1 Power-Operated Relief Valve 2RC-430

a. Inspection Scope

The inspectors reviewed post-maintenance testing activities following completion of maintenance on power-operated relief valve 2RC-430 to ensure that the testing was adequate for the scope of maintenance work which had been performed and that the testing acceptance criteria were clear and demonstrated operational readiness consistent with design and licensing basis documents. Specifically, the inspectors reviewed the following documents:

- Point Beach Design Basis Document DBD-09, “Reactor Coolant System,” Revision 1
- WO 9926019, “Adjust ‘To Move’ Pressure to 61 PSIG W/RV [Pounds Per Square Inch - Gauge With Reactor Vessel] Head Removed”
- WO 2000-097, “2RC-430 Stroke Time Verification Following Maintenance”
- Inservice Test IT-205, ”Pressurizer Power-Operated Relief Valves and Block Valves (Cold Shutdown) Unit 2,” Revision 22

b. Findings

No findings of significance were identified.

.2 Unit 2 New Analog Rod Position Indication Modules

a. Inspection Scope

The inspectors observed and reviewed post-maintenance testing activities following replacement of Unit 2 control rod position indication cards with new analog rod position indication modules to ensure that the testing was adequate for the scope of maintenance work which had been performed and that the testing acceptance criteria were clear and demonstrated operational readiness consistent with design and licensing basis documents. The inspectors also verified that all testing prerequisites were satisfied, the test was performed as written, and the test acceptance criteria were satisfied. Specifically, the inspectors reviewed the following documents:

- Instrumentation and Control Procedure 2ICP 04.029-3, “Analog Rod Position Channel Outage Functional Test and Linearization Checks,” Revision 1

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors observed work activities associated with the Unit 2 refueling outage which began on October 13, 2000. The inspectors assessed the adequacy of operations activities during the plant startup, and other outage-related activities, such as configuration management, clearances and tagouts, and core reload operations. Additionally, the inspectors also performed reviews and inspections of refueling operations for risk management, conformance to approved site procedures, and compliance with Technical Specifications. The following major activities were observed:

- outage planning meetings
- fuel handling activities
- chemical and volume control system fill and vent
- plant heatup
- unit startup
- other general outage activities, including foreign material exclusion controls and safety shutdown assessments

The following documents were reviewed:

- Refueling Procedure RP 1C, “Refueling,” Revision 45
- OP 4A, “Filling and Venting Reactor Coolant System,” Revision 51
- OP 4B, “Reactor Coolant Pump Operation,” Revision 39
- OP 1A, “Cold Shutdown to Hot Shutdown,” Revision 70
- OP 1B, “Reactor Startup,” Revision 39

b. Findings

During coupling of the Unit 2 reactor coolant pump RCP 2P-1B, performed in accordance with routine maintenance procedure RMP 9002-4, “Reactor Coolant Pump Uncoupling and Coupling,” Revision 24, the Unit 2 control (reactor) operator identified a decrease in reactor vessel level. Licensee investigation identified that the RCP 2P-1B seal injection vent valve (2CV-302B) was tagged open which created a drain path from the reactor coolant system (RCS) to the backseat seal leakage collection rig and subsequently to containment sump “A.” After identifying the source, operators untagged and shut 2CV-302B, terminating the leak path. Reactor vessel level dropped approximately one percent during the event.

The inspectors determined that the governing procedure for the activity, RMP 9002-4, was inadequate in that it did not provide direction or caution for isolation or removal of the backseat seal leakage collection rig prior to coupling the RCP. The inspectors performed a risk significance screening of the decrease in RCS inventory in accordance with NRC Inspection Manual Chapter 0609, “Significance Determination Process”, Appendix G, “Shutdown Operations.” The inspectors concluded that this finding had a credible impact on safety and that although the licensee failed to implement procedures to avoid perturbations in RCS level control during the RCP coupling activity, the finding was considered to be of very low safety significance because residual heat removal was not impacted and the amount of water that could have been drained from the reactor coolant system was limited by system configuration and alignment.

The inadequacy of procedure RMP 9002-4 was considered a violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which required, in part, that activities affecting quality, such as the RCP coupling, be prescribed by documented instructions, procedures or drawings appropriate to the circumstances. Procedure RMP 9002-4 was not appropriate in that it created an inadvertent decrease in RCS inventory. However, because of the very low safety significance of the item and because the licensee had included this item in their corrective action program (CR 00-3891), this procedure violation is being treated as an NCV (NCV 50-301/00-17-04(DRP)).

1R22 Surveillance Testing

.1 AFW System and Anticipated Transient Without Scram Mitigating System Actuation Circuitry (AMSAC) Testing

a. Inspection Scope

The inspectors reviewed and observed testing of the AFW system in response to steam generator low-low level signal, low AFW pump suction pressure, and AMSAC actuation for Unit 2. The inspectors reviewed the following documents:

- Operations Refueling Test ORT 3C, “Auxiliary Feedwater System and AMSAC Actuation,” Revision 3
- Point Beach Final Safety Analysis Report, Section 7.4, “Other Actuation Systems,” dated June 2000
- Point Beach Final Safety Analysis Report, Section 10.2, “Auxiliary Feedwater System,” dated June 2000
- CR 98-0338, “Independent Test of P-28A&B Contacts in Start Circuits of P38A&B”

- CR 98-3936, “AF-4020 and 4022 Close Inhibit Circuit”

- NP 1.2.6, “Infrequently Performed Tests or Evolutions,” Revision 7

The inspectors reviewed the test procedures for appropriateness, attended the Infrequently Performed Tests or Evolutions briefing, observed significant parts of the performance of the test, and verified that procedure adherence was consistent with regulatory requirements and standards. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; that all testing prerequisites were satisfied; and that test data were complete and appropriately verified. Following completion of the test, the inspectors verified that equipment was returned to a condition in which it could perform its safety-related function.

b. Findings

No findings of significance were identified.

.2 Primary Sampling

a. Inspection Scope

The inspectors observed the performance of the following surveillance tests:

- Chemistry Analytical Methods and Procedures (CAMP) 600.2, “Primary Side Sampling Procedures: Hot Leg and Pressurizer Liquid Sampling - Concurrently,” Revision 1

- CAMP 600.6, “Primary Side Sampling Procedures: Mixed Bed Inlet/Outlet Sampling,” Revision 1

- CAMP 600.11, “Primary Side Sampling Procedures: Sampling RCS For Dissolved Gas Samples Using Hot Leg Sample Lineup,” Revision 14, Temporary Change No. 2000-0339

For each surveillance test, the inspectors reviewed the test procedures for appropriateness, observed all or significant parts of the performance of the test, and verified that work practices and procedure adherence were consistent with regulatory requirements and standards. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; that all testing prerequisites were satisfied; and that test data were complete and appropriately verified. Following completion of the test, the inspectors verified that the test equipment was removed and that the equipment was returned to a condition in which it could perform its safety-related function.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

.1 Safety System Functional Failure PI

a. Inspection Scope

The inspectors reviewed reported third quarter 2000 data for the Safety System Functional Failure PI for Unit 1 and Unit 2 using the PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 0.

The inspectors reviewed Licensee Event Reports (LERs) and operator log entries to identify the number of safety system functional failures that occurred during the previous four quarters and compared that number to the number in the PI. The inspectors also reviewed the licensee's basis for excluding events and conditions identified in LERs from reporting as a safety system functional failure.

b. Findings

No findings of significance were identified.

.2 RCS Specific Activity PI

a. Inspection Scope

The inspectors reviewed the first through third quarter 2000 for the RCS Specific Activity PI for Unit 1 and Unit 2 using the PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0.

The inspectors reviewed chemistry department data to determine the value for RCS specific activity. The inspectors verified the PI value by independent calculation.

b. Findings

No findings of significance were identified.

.3 (Closed) URI 50-266/00-07-02(DRP);50-301/00-07-02(DRP): Emergency power - safety system unavailability performance indicators. The inspectors had identified that first quarter 2000 data had been reported in a train format consistent with historic data reporting made to the Institute of Nuclear Power Operation (INPO). The NRC guidance for PIs allowed using data gathered for INPO for periods before 2000, but did not address converting the train format from INPO standards to NRC-endorsed NEI standards. Resolution of the appropriate train format was provided by NEI 99-02

Frequently Asked Question 149 and the licensee submitted data consistent with the guidance.

4OA3 Event Follow-up

.1 (Closed) LER 266/2000-010-00: Manual Reactor Trip Due to Concerns for Diver Safety. On October 27, 2000, Unit 1 was manually tripped when communications with a diver conducting underwater inspections of the Unit 2 circulating water pump area were lost. Following the reactor trip, the main steam isolation valves were shut and the Unit 1 circulating water pumps were secured. After securing the pumps, all divers were removed from the water. The licensee determined that the cause of the loss of communications was due to an open wire in the communications line.

The inspectors responded to the reactor trip as documented in Section 1R14 of NRC Inspection Report 50-266/00-14(DRP); 50-301/00-14(DRP). Based on the inspectors' previous observations and a review of this LER during the current inspection, the inspectors determined that plant systems and components performed as designed, no human performance errors complicated the event response, and no emergency core cooling systems were challenged. The inspectors did note, however, that the event did constitute a scram with loss of normal decay heat removal and was the third such event for Unit 1 in the last 12 quarters. The "scrams with loss of normal decay heat removal" PI had, therefore, crossed the threshold from Green to White.

.2 (Closed) LER 301/2000-004-00: Unplanned Emergency Safety Features Actuation During Safeguards Bus Restoration. On November 5, 2000, an inadvertent start of the G03 emergency diesel generator (EDG) occurred during restoration of the Unit 2 4160-V busses 2A04 and 2A06. The G03 EDG received a fast start signal when fuses for the 2A06 undervoltage relay circuit were installed while the bus was still de-energized. The event did not have a credible impact on safety since mitigation systems were not adversely affected.

The failure of operators to properly sequence the installation of the undervoltage relay circuit fuses resulted in an unplanned engineered safety features actuation. The failure to properly sequence the fuse installation was a violation of 10 CFR 50, Appendix B, Criterion V, which required activities affecting quality be accomplished in accordance with prescribed procedures. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

.3 (Closed) LER 301/2000-005-00: Unplanned Emergency Safety Features Actuation During Calibration and Testing of Safeguards Bus Relays. On November 10, 2000, a loss of voltage condition was created on the Unit 2 Train "B" 4160-V bus, 2A06, during performance of Procedure 2RMP 9056-2, "Calibration and Testing of Safety-Related Protective Relays A06." In accordance with the procedure, a technician applied a multi-meter across two test points to check for continuity. The action energized an auxiliary relay in the under voltage circuitry which then opened the normal 2A06 supply breaker which created a loss of voltage condition on 2A06. Safety systems functioned as designed: G-03 EDG and G-04 EDG automatically started with G-04 EDG sequenced onto 2A06. The event did not have a credible impact on safety since mitigation systems functioned as designed and were not adversely affected.

The inspectors concluded that Procedure 2RMP 9056-2 was inadequate in that it created a loss of voltage condition on the Unit 2 Train “B” safety-related 4160-V bus which resulted in an unplanned engineered safety features actuation. The inadequacy of the procedure constituted a violation of 10 CFR Part 50, Appendix B, Criterion V, which required activities affecting quality be prescribed by procedures appropriate to the circumstances. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action in accordance with Section IV of the NRC’s Enforcement Policy.

4OA4 Cross-cutting Issues

The inspectors determined that a negative performance trend had developed in several cornerstone areas with procedure inadequacy being the common element. The determination was based on two examples identified during this reporting period (NCV 50-266/00-17-01; 50-301/17-01 and NCV 50-301/00-17-04) and two previously identified examples (NCV 50-266/00-13-01; 50-301/00-13-01 and NCV 50-301/00-14-01) of inadequate procedures. The four examples were: (1) following the full-scale replacement modification of the freeze protection system, the licensee approved procedure specified actions that inappropriately de-energized heat trace circuits for safety-related equipment when the intent was only to bypass alarms (NCV 50-266/00-17-01; 50-301/00-17-01), (2) on November 21, 2000, a decrease in RCS inventory inadvertently occurred when a drain path from the RCS to the backseat seal leakage collection rig and subsequently to containment sump “A” was created by plant personnel using an inadequate maintenance procedure during RCP coupling activities (NCV 50-301/00-17-04), (3) on October 11, 2000, the inspectors identified that the procedure for hydrostatic testing of the Unit 2 Train “A” residual heat removal system did not require the correct automatic overpressure protection device (NCV 50-301/00-14-01), and (4) a licensee-approved emergency operating procedure specified actions which created the potential for operators to prematurely secure containment spray prior to reaching the analyzed draw down level of the refueling water storage tank for adequate iodine removal from containment atmosphere (NCV 50-266/00-13-01; 50-301/00-13-01).

The causal relationship of these procedure errors was a failure of the licensee to ensure that procedures were correct and adequate prior to being approved for use. In each of the examples, procedural errors or omissions occurred during the development of the procedure and were not identified during the procedure review and approval process. The licensee’s human performance failures with respect to procedure development, review, and approval is considered a substantive cross-cutting issue not captured in individual findings, and is a finding characterized as No Color (FIN 50-266/00-17-05(DRP); 50-301/00-17-05(DRP)).

4OA5 Other

The inspectors reviewed the Institute of Nuclear Power Operations final report for an evaluation conducted the weeks of March 13 and 20, 2000.

4OA6 Meetings, Including Exit

On January 3, 2001, the inspectors presented the inspection results to Mr. A. Cayia and other members of licensee management. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

A. Cayia, Plant Manager
J. Gadzala, Licensing Manager
V. M. Kaminskas, Maintenance Manager
R.G. Mende, Director of Engineering
B. J. O'Grady, Operations Manager
M. E. Reddemann, Site Vice President
D. D. Schoon, System Engineering Manager
S. J. Thomas, Radiation Protection Manager

NRC

B. A. Wetzel, Point Beach Project Manager, NRR

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-266/00-17-01 50-301/00-17-01	NCV	Procedure specified actions inappropriately de-energized heat trace circuits for safety-related equipment when the intent was only to bypass alarms (Section 1R01)
50-301/00-17-02	URI	16 valves on Unit 2 SI system were locked closed instead of just locked as specified by checklist (Section 1R04)
50-266/00-17-03	NCV	Work plan did not specify appropriate actions to isolate inverter (Section 1R14.2)
50-301/00-17-04	NCV	Unplanned reactor vessel level decrease during coupling of reactor coolant pump (Section 1R20)
50-266/00-17-05 50-301/00-17-05	FIN	Cross-cutting issue for procedure inadequacies (Section 4OA4)

Closed

50-266/2000-010-00	LER	Manual Reactor Trip Due to Concerns for Diver Safety (Section 4OA3.1)
50-301/2000-004-00	LER	Unplanned Emergency Safety Features Actuation During Safeguards Bus Restoration (Section 4OA3.2)

50-301/2000-005-00	LER	Unplanned Emergency Safety Features Actuation During Calibration and Testing of Safeguards Bus Relays (Section 4OA3.3)
50-266/00-07-02 50-301/00-07-02	URI	Emergency power - safety system unavailability performance indicator format inconsistency (Section 4OA1.3)
50-266/00-17-01 50-301/00-17-01	NCV	Procedure specified actions inappropriately de-energized heat trace circuits for safety-related equipment when the intent was only to bypass alarms (Section 1R01)
50-266/00-17-03	NCV	Work plan did not specify appropriate actions to isolate inverter (Section 1R14.3)
50-301/00-17-04	NCV	Unplanned reactor vessel level decrease during coupling of reactor coolant pump (Section 1R20)
50-266/00-17-05 50-301/00-17-05	FIN	Cross-cutting issue for procedure inadequacies (Section 4OA4)

Discussed

50-301/00-14-01	NCV	Testing procedure did not specify the correct automatic overpressure protection device (Section 4OA4)
50-266/00-13-01 50-301/00-13-01	NCV	Emergency operating procedure directed the premature securing of containment spray (Section 4OA4)

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
AMSAC	Anticipated Transient Without Scram Mitigating System Actuation Circuitry
AOP	Abnormal Operating Procedure
CAMP	Chemistry Analytical Methods and Procedures
CL	Operations Checklist
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
FIN	Finding
INPO	Institute of Nuclear Power Operations
LER	Licensee Event Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NP	Nuclear Power Business Unit Procedure
NRC	Nuclear Regulatory Commission
OI	Operating Instruction
OP	Operating Procedure
PC	Periodic Check
PI	Performance Indicator
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
RHR	Residual Heat Removal
RMP	Routine Maintenance Procedure
SI	Safety Injection
URI	Unresolved Item
V	Volt
WO	Work Order