June 6, 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 50-266/01-08; 50-301/01-08

Dear Mr. Reddemann:

On May 8, 2001, the NRC completed an inspection at your Point Beach Nuclear Plant. The enclosed report documents the inspection findings which were discussed on May 8, 2001, with you and other 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 personnel. Specifically, this inspection focused on radiation protection practices during outage periods, inservice inspection activities, refueling activities, and other routine resident inspections

No findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html the Public Electronic Reading Room).

Sincerely,

Original signed by
Roger D. Lanksbury

Roger D. Lanksbury, Chief
Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 50-266/01-08; 50-301/01-08

See Attached Distribution

cc w/encl: R. Grigg, President and Chief
Operating Officer, WEPCo
R. Anderson, Executive Vice President
and Chief Nuclear Officer
J. Gadzala, Licensing Manager
D. Weaver, Nuclear Asset Manager
F. Cayia, Plant Manager
J. O'Neill, Jr., Shaw, Pittman,
Potts & Trowbridge
K. Duveneck, Town Chairman
Town of Two Creeks
D. Graham, Director
Bureau of Field Operations
A. Bie, Chairperson, Wisconsin
Public Service Commission
S. Jenkins, Electric Division
Wisconsin Public Service Commission
State Liaison Officer

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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:	50-266/01-08; 50-301/01-08
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Plant, Units 1 & 2
Location:	6610 Nuclear Road Two Rivers, WI 54241
Dates:	April 1 through May 8, 2001
Inspectors:	P. Krohn, Senior Resident Inspector R. Powell, Resident Inspector M. Holmberg, Reactor Inspector M. Mitchell, Radiation Specialist R. Schmitt, Radiation Specialist D. Chyu, Reactor Inspector
Approved by:	Roger D. Lanksbury, Chief Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000266-01-08(DRP), IR 05000301-01-08(DRP), on 04/01-05/08/2001, Nuclear Management Company, LLC, Point Beach Nuclear Plant, Units 1 & 2. Fire Protection.

This report covers a 5-week routine resident inspection, a review of inservice inspection activities, a radiation protection outage ALARA (as-low-as-is-reasonably-achievable) inspection, and a review of fire protection open items. The inspections were conducted by resident and specialist inspectors. Three Unresolved Items (URIs) were identified, one each in the areas of the maintenance rule, risk assessment, and inservice inspection. No findings or violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process.” The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at **http://www.nrc.gov/NRR/OVERSIGHT/index.html.** 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

None.

B. Licensee-Identified Findings

Violations of very low significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appeared reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Unit 1 began the inspection period at 100 percent power. Unit 1 was shutdown on April 6, 2001, for refueling outage U1R26 and remained shutdown for the duration of the inspection period. Unit 2 was at or near 100 percent power throughout the inspection period.

1. REACTOR SAFETY

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

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

.1 Unit 2 ‘A’ Train Emergency Diesel Generator (EDG)

a. Inspection Scope

The inspectors performed a partial system walkdown of the Unit 2 ‘A’ Train EDG (G-02) while the backup EDG, G-01, was out-of-service for 480-volt bus, 1B-03, work. The inspectors used licensee checklists (CLs) CL 11A G-02, “G-02 Diesel Generator Checklist,” Revision 21, and CL 10D, "Fuel Oil System," Revision 17, during the walkdowns. The inspectors also utilized selected portions of system electrical, fuel oil, lubricating oil, and starting air drawings to accomplish the inspection.

The inspectors walked down G-02 to verify the correct position of control switches, breakers, louvers, dampers, and valves associated with G-02 and ventilation, heating, fuel oil transfer, and engine control power alignments associated with G-02 support systems. The inspectors also performed walkdowns in the control room to verify appropriate switch positions and valve configurations. Finally, the inspectors evaluated other elements such as material condition, housekeeping, and component labeling.

b. Findings

No findings of significance were identified.

.2 Unit 1 Safety Injection (SI) System

a. Inspection Scope

The inspectors performed a partial walkdown of the Unit 1 SI 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 Unit 1 refueling outage activities. The inspectors reviewed Operations CL 7B, “Safety Injection System Checklist Unit 1,” Revision 14, as part of the inspection.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

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

- Unit 1 Containment, 66' level, Fire Zone 520
- Unit 1 Containment, 46' level, Fire Zone 516
- Unit 1 Containment, 21' level, Fire Zone 511
- Unit 1 Containment, 8' level, Fire Zone 505

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,” dated August 1999.

The inspectors looked at fire hoses, sprinklers, and portable fire extinguishers to verify that they were installed at their designated locations, were in satisfactory physical condition, and were unobstructed. The inspectors also evaluated 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 1 refueling 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

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed thermal performance test data for the Unit 1 'A’ and 'B' Component Cooling Water Heat Exchangers (CCHXs). The thermal performance tests were completed at the beginning of the Unit 1 refueling outage when the decay heat load on the components was relatively high. The inspectors reviewed test data and design basis requirements to verify that the CCHXs were capable of performing their safety-related function. Specifically, the inspectors examined performance trends, acceptance criteria, test data, and testing protocol. The inspectors reviewed the test procedure and system drawings to verify that the valve and heat exchanger alignments were adequate to support measurements of heat exchanger thermal performance. The inspectors reviewed component cooling water and service water flow rates through the CCHXs to confirm that the parameters were within limits defined in the test protocol. The inspectors also performed independent calculations to ensure that turbulent flow conditions in the CCHX tubes were met, a requirement in the test protocol to ensure that test parameter uncertainty analysis assumptions were satisfied. Finally, the inspectors reviewed the frequency of CCHX inspections to verify that inspection intervals were sufficient to detect CCHX degradation prior to the loss of heat removal capabilities below design values. Documents used in the review are listed at the end of the report.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's inservice inspection program for monitoring degradation of the reactor coolant system (RCS) boundary and the risk significant piping system boundaries based on review of records and in-process observation of non-destructive examinations.

The inspectors reviewed two modifications, disposition of four recordable indications identified during previous nondestructive examinations, and reviewed radiographic records for three Code welding activities. In addition, the inspectors observed the following nondestructive examinations:

- ultrasonic examination of the pressurizer spray nozzle inner radius;
- ultrasonic examination of the pressurizer relief line nozzle inner radius;
- ultrasonic examination of Weld 13 on pipe FW-16-FW-1002;
- magnetic particle examination of Weld 13 on pipe FW-16-FW-1002;
- dye penetrant examination of Weld 15R1 on pipe RC-02-LD-1001;
- dye penetrant examination of Weld 16B on pipe RC-02-BP-1003; and
- dye penetrant examination of Weld 32B on pipe CVC-02-LD-1001.

The records reviewed and activities observed were evaluated for conformance with requirements in the 1986 Edition, No Addenda, of the American Society of Mechanical Engineers Code, Section III, Section V, Section IX, and Section XI. In addition, requirements of 10 CFR 50.55a and the 1995 Edition with 1996 Addenda of the American Society of Mechanical Engineers Code, Section XI, Appendix VIII, were used for evaluating the ultrasonic examinations of piping welds.

The inspectors also reviewed a sample of inservice inspection related problems documented in the licensee's corrective action program to assess the appropriateness of the corrective actions. The documents listed at the end of the report were used in the review.

b. Findings

The inspectors identified ultrasonic examination equipment which had not been tested to confirm the essential variable tolerances required by the American Society of Mechanical Engineers Code.

The licensee was required to implement Appendix VIII of Section XI of the 1995 Edition with the 1996 Addenda of the American Society of Mechanical Engineers Code by 10 CFR 50.55a(g)(6)(ii)(C). Article VIII-4110 of Section XI allowed the ultrasonic examination procedure to be modified for substitution or replacement of pulsers, receivers, or search units without requalification, when essential variable tolerances had been met. The control of these essential variables ensured that examinations conducted with replacement equipment were adequate to detect and size flaws in safety-related welds or bolting. The licensee had used three replacement ultrasonic instruments and associated search units to perform examinations of Code components and had not measured the essential variable tolerances. Specifically, the tolerances not measured included the upper, lower, and center frequencies; and the amplitude, rise time, and duration of the pulse of the instruments. For the search units, the tolerances not measured included the center frequency, and waveform duration.

The licensee had been following the industry Performance Demonstration Initiative practice that allowed equipment of the same make and model as that used during the procedure qualification to be exempt from essential variable tolerance testing. The inspectors were concerned that a manufacturer may produce equipment of the same model that varied beyond the essential variable ranges required by Article VIII-4110, which would go undetected due to the lack of confirmatory testing. This issue is an Unresolved Item (URI 50-266/01-08-01; URI 50-301/01-08-01) pending review of licensee corrective actions to resolve the essential variable tolerances for the ultrasonic examination equipment. This issue was documented in the licensee's corrective action program in Condition Report (CR) 01-1455. The licensee's recommended actions included writing a Code Case or seeking NRC approval, via a relief request, to deviate from the Code requirements.

1R12 Maintenance Rule Implementation (71111.12)

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, 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 reviewed station logs, maintenance work orders, (a)(1) corrective action plans, and a sample of CRs to verify the licensee was identifying issues related to the maintenance rule at an appropriate threshold and corrective actions were appropriate. Additionally, the inspectors reviewed the licensee's performance criteria to ensure that the criteria adequately monitored equipment performance and verified that changes to performance criteria were reflected in the licensee's probabilistic risk assessment. Specific components and systems reviewed were:

- Steam Generator Atmospheric Dump Valves
- Auxiliary Feedwater

The inspectors reviewed various corrective action program documents (CRs), including CR 01-1224, "Unavailability Time Not Counted," which was initiated as a result of this inspection activity and was reviewed as part of the inspection scope. Documents used by the inspectors during assessment of this area are listed at the end of the report.

b. Findings

The inspectors identified that the licensee returned the auxiliary feedwater (AFW) system from (a)(1) to (a)(2) status on December 18, 2000, after achieving action plan goals of installing restriction orifices in the AFW motor-driven pump recirculation lines and meeting new pump unavailability performance criteria. The inspectors reviewed station logs, CRs, and work orders to independently determine AFW unavailability hours. The inspectors identified that 6.5 hours of P-38A ('A' Train Motor-Driven AFW Pump, shared by both units) unavailability from November 1999 were not included in the licensee's calculated unavailability hours. With the addition of the 6.5 hours, the inspectors determined that P-38A, and thus AFW, exceeded its maintenance rule performance criteria of 160 hours/2 years in November and December 2000. The inspectors therefore concluded that the goals of the (a)(1) action plan were not met prior to returning the AFW system to (a)(2) status on December 18, 2000. Because the issue was identified late in the inspection period, the inspectors considered this issue to be a URI (URI 50-266/01-08-02; 50-301/01-08-02) pending additional inspection effort and review of the risk significance. The issue was entered into the licensee's corrective program as CR 01-1671.

The inspectors noted that the failure to accurately track unavailability of the 'A' Motor-Driven AFW Pump, P-38A, continued a trend of maintenance rule program deficiencies. In the past 6 months, the inspectors identified the following:

- On December 6, 2000, the inspectors identified that the performance criteria for the RCS did not effectively monitor the risk significant function of pressure relief due to the performance criteria allowing a power-operated relief valve block value to be closed continuously for a 24-month period. The issue was entered into the licensee's corrective action program as CR 00-4058, "Less Than Adequate Performance Criteria for Maintenance Rule."

- On March 1, 2001, the inspectors identified that performance criteria for a maintenance rule system, control room ventilation, were not evaluated on an ongoing basis from January 2000 through the time of inspector discovery on March 1. The issue was entered into the licensee's corrective program as CR 01-0641, "Control Room Ventilation VNCR Maintenance Rule Performance."

- On April 11, 2001, the inspectors identified that 5 hours of unavailability time for the Unit 2 'B' steam generator atmospheric steam dump were not counted against maintenance rule performance criteria. The issue was entered into the licensee's corrective program as CR 01-1224, "Unavailability Time Not Counted."

- On April 27, 2001, the inspectors identified that in addition to P-38A being returned to (a)(2) status prior to meeting (a)(1) monitoring goals, unplanned unavailability in March 2001 caused the system to exceed its established (a)(1) threshold without the licensee recognizing and addressing the threshold crossing in accordance with plant procedures. The issue was entered into the licensee's corrective program as CR 01-1671.

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

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 to verify that scheduled and emergent work activities were adequately managed. In particular, the inspectors reviewed the licensee's program for conducting maintenance risk safety assessments to verify that the licensee's planning, risk management tools, and the assessment and management of online risk were adequate. The inspectors also reviewed licensee actions to address increased online 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, to verify that the actions were accomplished when online risk was increased due to maintenance on risk significant structures, systems, or components. When risk significant equipment was taken out-of-service, the inspectors reviewed selected tagouts to ensure that no unintentional equipment had been removed from service which would increase the assumed risk profile. The following specific activities were reviewed:

- The maintenance risk assessment for work planned the week beginning April 15, 2001. This included the scheduled Unit 1 refueling outage de-energization (drop) of 480-volt safeguards bus 1B03. As a result of the bus drop, the inspectors examined the effect of the loss of shared equipment on the risk profile of Unit 2. Finally, the inspectors reviewed CR 01-1380, "Risk Profile Underestimated," which was initiated as a result of this inspection activity.

b. Findings

During April 19 to 21, 2001, the licensee secured power to Unit 1, 480-volt alternating current safeguards bus, 1B03, as part of U1R26 planned refueling outage activities. The inspectors reviewed the safety-related loads powered from 1B03, focusing on those components that were common to both Unit 1 and Unit 2 operations. Based on the review, the inspectors asked a probabilistic risk assessment engineer if fuel oil transfer pump P-206A, "EDG G-01 Fuel Oil Transfer Pump," had been included in the risk profile of Unit 2, which was operating at 100 percent power. The engineer responded that P-206A had not been included in the Unit 2 risk assessment for the 1B03 bus drop and added that another component, P-035A, “Electric-Driven Fire Pump,” had also been omitted from the assessment. The combined effect of not including P-206A and P-35A in the Unit 2 risk profile raised the core damage frequency from a factor of 6.9 times baseline risk to approximately 13.1 times baseline risk.

Bus 1B03 had been de-energized for approximately 4 hours when the inspectors asked the initial question and the risk assessment errors were identified. Once the P-206A and P-35A errors were identified, the probabilistic risk assessment engineer contacted the shift technical advisor who updated the Unit 2 safety monitor risk profile. The 1B03 work window had been scheduled for 42 hours duration and actually lasted 49.1 hours.

When P-206A and P-35A were included in the Unit 2 risk assessment, the instantaneous core damage frequency increased from 2.63E-4/yr to 5.01E-4/yr (a factor of 1.9). Although the instantaneous damage frequency was relatively high (5.01E-4), the change in core damage probability was very low (1.09E-7) due to the short duration before the error was realized (four hours). Because further review is required, this issue is considered to be a URI (URI 50-301/01-08-03(DRP)) pending NRC review of the risk significance. The inspectors noted that if the 1B03 bus de-energization had proceeded without the error being identified, the unrealized increase in core damage probability would have been 1.34E-6. This issue was entered in the licensee’s corrective action program as CR 01-1380.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors observed and evaluated operator response to increasing counts on the Unit 1 source range detector, N-31, during the U1R26 refueling outage on April 12, 2001. The inspectors reviewed Abnormal Operating Procedure AOP 6E, “Alternate Boration/Loss of Shutdown Margin,” Revision 9, to verify that operator actions were taken in a timely and appropriate manner. Finally, the inspectors reviewed CR 01-1249, “1N31 Source Range Instrument Failed,” which was initiated as a result of the event.

The inspectors also reviewed the actions associated with extinguishing a fire that occurred in the Unit 1 primary containment on the 'A' steam generator manway platform on April 24, 2001. The inspectors reviewed the timeliness and effectiveness of the fire brigade response as well as the amount and type of extinguishing agent used to put the fire out. Documents used in the review are listed at the end of the report.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the operability evaluation listed below to verify that it addressed the applicable current licensing basis requirements and commitments, and provided an adequate basis for justifying operability. Independent reviews were conducted that included discussions with licensee personnel and reviews of design and licensing basis documentation. The inspectors reviewed the operability evaluation associated with CR 01-1232, "Snubber Testing Machine." This CR addressed the improper functioning of the snubber test machine. The machine was determined to be improperly functioning.

c. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed a permanent plant modification that was being made to the plant's intake structure which connects the pump house facility to Lake Michigan, the ultimate heat sink. The modification submerged the intake structure to reduce the likelihood of drowning cormorants (a number of these migratory birds have been drowned in previous years while diving for food near the intake structure) and to improve the effectiveness of ice melt operations in winter when the intake structure and forebay are susceptible to ice formation.

The inspectors compared the planned modification work to design and licensing requirements, specifically the Final Safety Analysis Report system and functional descriptions. The inspectors reviewed the modification package to ensure that it included a safety evaluation. The individual work plan (IWP) was also reviewed to ensure that it conformed to the safety evaluation. Documents used in the review are listed at the end of the report.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed and reviewed post-maintenance testing activities conducted in accordance with RMP 9061, "Hydraulic Snubber Surveillance and Testing," Revision 1, following rebuild of snubbers 1HS-3, "'B' Main Steam Line-West," and HS-601R-92 A2, "SRV [Safety Relief Valve] Discharge Piping," to ensure that the test was adequate for the scope of the maintenance work which had been performed. The inspectors also evaluated the test activities to verify that the test was performed as written; that all testing prerequisites were satisfied; and that the test data were complete, appropriately verified, and met the requirements of the testing procedure.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed work activities associated with the Unit 1 refueling outage, U1R26, which began on April 6, 2001. The inspectors assessed the adequacy of operations activities during the plant cooldown, and other outage related activities such as configuration management, clearances and tagouts, and RCS reduced inventory operations. Additionally, the inspectors reviewed refueling operations for implementation of risk management, preparation of contingency plans for loss of key safety functions, conformance to approved site procedures, and compliance with Technical Specifications. The inspectors also verified compliance with commitments made during licensee response to Generic Letter 88-17, “Loss of Decay Heat.” The following major activities were observed or performed:

- outage planning meetings
- unit cooldown and depressurization
- unit heatup and pressurization
- blocking of SI
- draining the RCS for reduced inventory operations
- adequate reactor vessel level and temperature instrumentation during reduced inventory conditions
- monitoring and verification of nuclear instrument operability during core alterations
- fuel handling activities
- walkdowns of residual heat removal (RHR) systems during times of reduced inventory to ensure decay heat removal capabilities
- walkdowns of the spent fuel pool cooling system after all the fuel had been offloaded from the reactor to the pool
- walkdowns of selected shutdown inventory addition makeup paths
- walkdowns of RCS boundary integrity prior to increasing reactor vessel inventory
- selected replacement of fuel assembly top nozzle blocks
- effect of switchyard maintenance activities on continuity of power to safeguards buses relied upon to maintain operability of RHR systems
- containment closure abilities during periods of core alterations
- walkdowns to ensure all debris which could inhibit mitigating the effects of a design basis accident were removed from the primary containment
- other general outage activities, including foreign material exclusion controls and safety shutdown assessments

Documents used in the review are listed at the end of the report.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Leakage Reduction and Preventive Maintenance Program Seat Leakage Test

a. Inspection Scope

The inspectors reviewed and observed the outage frequency leakage reduction testing of the train ‘A’ RHR system. The inspectors reviewed the test procedure for appropriateness and observed portions of the test to verify procedural compliance, ensure all testing prerequisites were satisfied, and to confirm that test data were appropriately reviewed and met the requirements of the testing procedure. Additionally, the inspectors performed walkdowns to verify proper configuration of the hydostatic pressure test rig, independently performed leakage calculations, and conducted a post-test system alignment review.

Finally, the inspectors reviewed CR 01-1053, “Valves Not Locked as Required” which was initiated as a result of this inspection activity and was reviewed as part of the inspection scope. Documents used by the inspectors during the assessment of this area are listed at the end of the report.

b. Findings

No findings of significance were identified.

.2 Unit 1 SI Actuation With Loss of Engineered Safeguards Alternating Current

a. Inspection Scope

The inspectors reviewed and observed operation of SI sequencing and EDG loading under conditions of loss of engineered safeguards alternating current power concurrent with a manual SI signal for Unit 1, 'A' and 'B' safeguards trains. Observed activities also included automatic load shedding and restoration of vital loads following a manual trip of the diesel generator output breakers for the Unit 1 EDGs: G-01 and G-03. Observations were performed from the control room and locally at selected engineering safeguards equipment locations.

The inspectors attended and reviewed the adequacy of pre-job briefs given to test and operations personnel prior to the performance of the surveillance tests. The inspectors reviewed pre-test equipment alignments to ensure proper system configurations prior to starting the test. Selected danger and caution tags were verified installed as required by the surveillance test procedure. Communication practices, control room decorum, receipt of expected alarms and warning lights, supervisory overview, and the interface between test and on-shift licensed personnel involving plant equipment were observed. The inspectors also observed G-01 and G-03 EDG design basis loading, frequency, and start times. Unit 1 control room crew response to unexpected boration in Operations Refueling Test ORT 3A, Step 5.9, and a failed time delay relay in ORT 3B, Step 5.4, were observed. Documents used in the review are listed at the end of the report.

b. Findings

No findings of significance were identified.

.3 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 a Unit 1 steam generator low-low level signal, low AFW pump suction pressure, and AMSAC actuation signals. The testing was accomplished in accordance with ORT 3C, "Auxiliary Feedwater System and AMSAC Actuation Unit 1," Revision 3. The inspectors reviewed the test procedures for appropriateness, 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 performed walkdowns to verify 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. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiological Boundary Verifications, and Observations of Radiation Worker Performance

a. Inspection Scope

The inspectors performed walkdowns of the radiologically controlled area (RCA) to verify the adequacy of radiological controls and postings. Specifically, the inspectors walked down several radiologically significant work area boundaries (high and locked high radiation areas) in the Units 1 and 2 primary auxiliary building and the Unit 1 containment. In addition, the inspectors performed confirmatory radiation measurements in these areas to verify that access to these areas (Very High Radiation Areas (VHRA), High Radiation Areas (HRA), and Radiation Areas) were properly posted and controlled in accordance with 10 CFR Part 20, licensee procedures, and Technical Specifications. The inspectors also observed radiation workers performing the activities described in Section 2OS2.2, evaluated their awareness of radiological work conditions, and verified the implementation of radiological controls specified in applicable radiation work permits and as-low-as-is-reasonably-achievable (ALARA) plans.

b. Findings

No findings of significance were identified.

.2 Reviews of Radiation Work Permits

a. Inspection Scope

The inspectors reviewed selected routine radiation work permits (RWP) and electronic dosimeter alarm set points for both dose rate and accumulated dose for access to HRAs. The inspectors evaluated established work controls to determine if worker exposures were maintained ALARA.

b. Findings

No findings of significance were identified.

.3 Reviews of Licensee's Programmatic Controls for Highly Activated/Contaminated Materials

a. Inspection Scope

The inspectors reviewed procedure REI [Reactor Engineering Instruction] No. 24.0, "Spent Fuel Pool," Revision 15, to verify that all highly activated/contaminated materials were properly stored and controlled in the spent fuel pool. The inspectors also discussed with the Radiation Protection Manager the licensee's programmatic controls over the highly activated/contaminated materials. The inspectors observed remote top nozzle removal/replacement in the spent fuel pool to verify worker adherence to procedures and appropriate radiological work practices in handling highly radioactive and contaminated material.

b. Findings

No findings of significance were identified.

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

.1 ALARA Planning

a. Inspection Scope

The inspectors reviewed the station's collective exposure histories for 1998 to present, current exposure trends for the ongoing Unit 1 refueling outage (U1R26), and planned and completed radiological work activities for the outage to assess current performance and exposure challenges. The inspectors reviewed the annual exposure data (to date) and the station's three-year rolling average exposure information and compared it with national pressurized water reactor industry data.

b. Findings

No findings of significance were identified.

.2 Job Site Inspections, ALARA Controls, and Radiological Work Planning

a. Inspection Scope

The inspectors observed work activities in the RCA that were performed in radiation areas, HRAs, or VHRAs to evaluate the use of ALARA controls. Work areas were surveyed to verify that radiation levels were consistent with the licensee's survey data and that low dose areas were designated and appropriately used by workers. The licensee's engineering controls were evaluated at selected locations, and the inspectors verified that the controls were consistent with those specified in the ALARA plans. The inspectors also observed and questioned workers at each job location to determine that they had adequate knowledge of radiological work conditions and exposure controls. Specifically, the inspectors reviewed RWPs, ALARA reviews, and surveys and observed pre-job radiological briefings (as applicable) and radiation protection technician performance for the following work activities:

- 1CV-1298 Replacement/Modification (containment),
- Insulation Work/Scaffolding Construction Support,
- Fuel element top nozzle removal/replacement,
- Steam Generator Eddy Current Testing, and
- Reactor Coolant Pump Seal Removal.

b. Findings

No findings of significance were identified.

.3 Source Term Reduction and Control

a. Inspection Scope

The inspectors reviewed the status of the licensee's source term reduction program, focusing on those initiatives taken for the outage which included shutdown chemistry controls (i.e., early boration/hydrogen peroxide addition), hydrolazing and other decontamination work, and installation of permanent and temporary shielding. The inspectors also evaluated other ongoing source term reduction strategies, such as water chemistry control and hot spot reduction initiatives, to verify that a viable source term control program was in place.

b. Findings

No findings of significance were identified.

.4 Verification of Exposure Goals and Exposure Tracking System

a. Inspection Scope

The inspectors reviewed the licensee's outage dose goals. In particular, the inspectors compared the current dose estimates to the licensee's historical performance and reviewed the licensee's assumptions used to estimate current job doses to verify that the licensee had a technical basis for its dose estimates. The inspectors also reviewed the licensee's dose trending/tracking to ensure that the licensee was taking action to identify work activities that were not progressing as expected. In addition, the inspectors reviewed personnel exposures within work groups to identify the reasons for any significant exposure variations.

The licensee established a goal of 75 person-rem for the U1R26 refueling outage. The inspectors compared the current outage dose (45 person-rem) to the expected outage dose (22 person-rem) to assess outage performance in light of ALARA expectations. The licensee's explanation of the changes to outage job scheduling for outage efficiency was reviewed.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems (71121.01 and 71121.02)

a. Inspection Scope

The inspectors reviewed self-assessments and audits, since the last outage, as well as selected outage generated condition reports, which focused on ALARA planning and controls. The inspectors evaluated the effectiveness of the licensees self-assessment process to identify, characterize, and prioritize problems. The inspectors reviewed the licensees ability to identify repetitive problems, contributing causes, the extent of conditions, and corrective actions which would achieve lasting results.

The inspectors reviewed a number of licensee identified deficiencies in "Post-Job ALARA Reviews," from the previous outage, to assess the effective use of pre-job planning.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

.1 Initiating Events

a. Inspection Scope

The inspectors reviewed the following PIs following licensee first quarter 2001 data submission:

- Unplanned Power Changes
- Scrams With Loss of Normal Decay Heat Removal
- Unplanned Scrams

The inspectors used the PI definition and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0. The inspectors reviewed station logs and licensee monthly Operating Data Reports to determine the number of unplanned power changes, unplanned scrams, and critical hours during the previous four quarters. The inspectors independently calculated PI values.

Finally, the inspectors reviewed CR 01-1570, "U2 Number of Hours Critical Concern," 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.

.2 Mitigating Systems

a. Inspection Scope

The inspectors reviewed the Heat Removal System Unavailability (AFW) PI following licensee first quarter 2001 data submission. The inspectors used the PI definition and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0. The inspectors reviewed station logs, CRs, and work orders to determine the number of train unavailability hours and the number of hours the AFW system was required to be available. The inspectors verified PI values by independent calculation.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 Unit 1 Containment Fire

a. Inspection Scope

Early on the morning of April 24, 2001, while Unit 1 was shutdown and defueled for refueling outage U1R26, a fire occurred in the 'A' steam generator vault on the access platform to the primary side manway covers. The fire consumed a bag of rags and testing equipment debris, lasted for approximately 23 minutes, and was extinguished within 3 minutes of the fire brigade arriving on the scene.

Upon notification from the control room staff, the inspectors responded to the site and made an initial tour and extent-of-condition damage survey of containment. During the initial tour, the inspectors examined the effect of the fire on safety-related instrumentation, power supplies, and RCS barrier material. Evaluations of potential spread of contamination and airborne radiological conditions were also made. Subsequent evaluations included consideration of the effects of 70 pounds of dry chemical extinguishing agent used in putting out the fire that was dispersed through large areas of the 'A' steam generator and reactor coolant pump vaults. Consideration was given to the effects of the extinguishing agent on exposed stainless steel pipe and tube surfaces, motor- and air-operated valves, safety-related snubbers, and electrical contacts and components. Since an estimated 130 milligrams of sulfate were found in the 'A' steam generator hot leg channel head, consideration was also given to the effects of the extinguishing agent on steam generator tube internals and other materials internal to the RCS in contact with primary coolant. Finally, the inspectors reviewed the adequacy and timeliness of the “Notification of Unusual Event” emergency declaration made by control room personnel as the result of the fire. Documents used by the inspectors during the assessment of this area are listed at the end of the report.

b. Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 50-266, 50-301/1998-020-00: Unprotected cables in cable spreading room. The licensee identified a group of telephone and flourescent lighting wires that were not installed in conduits or covered cable trays. The licensee’s evaluation of this issue determined that these low voltage cables did not constitute a fire hazard. The inspector reviewed Fire Protection Engineering Evaluation FPEE-1999-00, “Unprotected Telephone wires and lighting fixtures power cords in the Cable Spreading Room,” Revision 0, and agreed with the licensee’s assessment. This item is not a violation of regulatory requirements.

.3 (Closed) LER 50-266, 50-301/1999-030-00: Incorrect assumptions for determining equipment necessary to achieve and maintain hot shutdown conditions following a postulated fire. The licensee identified that the plant safety-related battery chargers should be reclassified as hot shutdown equipment. This was based upon the licensee’s determination that in the event of fire-induced short circuits that de-energized the chargers, the chargers could not be re-energized until the faulted portion of the circuit was jumpered and the control fuses replaced. Initially, the licensee considered the activities necessary to recover the battery chargers to be repair activities not allowed for hot shutdown equipment. Subsequently, the licensee determined that plant abnormal operating procedures, used in the event of a fire, would direct operators to trip an inverter to reduce the loads on the battery. This action would ensure that the battery was capable of supplying necessary loads for a period of time sufficient to achieve safe shutdown. In addition, the licensee identified two additional swing battery chargers and another battery that could be used to supply safe shutdown loads. Based on the licensee's demonstration of the station batteries capability to independently supply safe shutdown loads for a sufficient amount of time necessary to achieve and maintain hot shutdown, this item was not a violation of regulatory requirements. The inspector's review of this LER did not identify any new issues.

.4 (Closed) LER 50-301/1999-002-00: Red channel of steam generator pressure indication passes through fire zone. This licensee-identified finding involved a failure to provide direct reading of steam generator 'B' pressure for post-fire safe shutdown purposes. This item was determined to have very low safety significance (Green).

The licensee, during an Appendix R rebaseline project, determined that the cable for the red channel of instrumentation for steam generator 'B' pressure transmitter 2PT-00483 was routed through Fire Zone 187. However, this parameter was required to be monitored at the alternate safe shutdown location following a fire in Fire Zone 187. The licensee re-routed the cables associated with this instrument out of the fire zone. The inspectors considered the corrective action acceptable. The licensee determined that steam generator pressure could be extrapolated from other parameters such as reactor coolant temperature, RCS pressure, and steam generator level. In addition, steam generator over-pressure protection would be provided by the main steam code safety relief valves.

Since the issue involved degradation of the defense-in-depth elements, the inspectors evaluated the issue using NRC Manual Chapter 0609, "Appendix F, Fire Protection Significance Determination Process (SDP)." Using Phase 1 of the SDP, the issue screened out as Green because it did not affect detection, suppression, fire barriers, or the 20-foot separation requirement. The inspector considered this finding to be of very low safety significance (Green) because other instruments could be used to provide the steam generator 'B' pressure parameter. This issue is dispositioned in Section 4OA7 of this report.

.5 (Closed) LER 50-266, 50-301/1999-004-00 and 01: Fuel oil transfer pump cable in the AFW pump room outside Appendix R design basis. The licensee identified that a postulated fire in the north half of the AFW pump room (designated as a III.G.2 area) could potentially result in the disruption of the electrical power supply to the fuel oil transfer pump which provided a continuous fuel oil supply to the G-01 EDG. The postulated fire could not cause a loss of offsite power event because the cabling associated with offsite power was not routed in the area. Therefore, offsite power would continually be available to supply power to safe shutdown equipment. This issue did not constitute a violation of NRC requirements.

.6 <u>(Closed) LER 50-266/1999-007-00</u>: Cable tray fire stops do not meet Appendix R exemption requirements. A licensee-identified finding was evaluated involving a failure to install fire stops as an approved alternative to the requirements of 10 CFR Part 50, Appendix R, Section III.G.2. This issue was determined to have very low safety significance (Green).

The Unit 1 motor control center room contained redundant trains of charging pump cables with intervening combustibles in the form of other cable trays. 10 CFR Part 50, Appendix R, Section III.G.2.b required that redundant cables within the same fire area be separated by 20-foot space with no intervening combustibles. An exemption to this requirement was granted by the NRC on July 3, 1985, to allow an alternative to the requirement of 20-foot separation. The approved alternative was to install area suppression and to install fire stops in the cable trays traversing the area between the redundant trains of charging pump cables. The licensee identified that three cable tray fire stops were not installed in accordance with the requirements of the approved exemption. As a result, a postulated fire could damage the power cables for all three charging pumps.

Since this issue involved a degradation of defense-in-depth element, the inspectors evaluated the issue using NRC Manual Chapter 0609, "Appendix F, Fire Protection SDP." Phase 1 and 2 evaluations of the fire protection SDP were performed because the issue involved degradation of 20-foot separation requirement and combustibles were located in the combustible-free zone. Because the charging pumps were not credited in the mitigation of a transient, a transient with the loss of power conversion system, or loss of offsite power, the increase in fire risk due to the loss of redundant charging pumps was determined to have very low risk significance (Green). This issue is dispositioned in Section 4OA7 of this report.

.7 <u>(Closed) LER 50-266, 50-301/1999-010-00</u>: Inadequate Appendix R emergency lighting. The licensee identified several locations where emergency light illumination was inadequate. This deficiency was not a degradation of fire protection features or defense-in-depth elements and therefore was not evaluated using NRC Manual Chapter 0609, "Appendix F, Fire Protection SDP." Although this violation should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy, NUREG-1600. This issue was entered in the licensee's corrective action program as CR 99-2546.

.8 <u>(Closed) LER 50-266, 50-301/2000-004-00</u>: Potential loss of process monitoring instrumentation due to a fire in containment. A licensee-identified finding was evaluated involving a failure to provide 20-foot separation between redundant trains of equipment within non-inerted containment as required by 10 CFR Part 50, Appendix R, Section III.G.2.d. This was determined to have very low safety significance (Green).

The licensee identified that a postulated fire at the 21-foot elevation in either Unit 1 or Unit 2 containment could cause a loss of several temperature elements and steam generator level instruments such that the temperature and level indications from the same loop would not be available. This deficiency was due to the redundant instrument cables being located within 20 feet of each other. The licensee determined that at the locations where the redundant instrument cables were within 20 feet of each other, the only combustible material was open cable trays near the ceiling. In addition, the licensee's administrative procedure prohibited any transient combustibles in containment while at power. There were no ignition sources in the areas of concern.

Since this issue involved a degradation of a fire protection feature and defense-in-depth element, the inspectors evaluated the issue using NRC Manual Chapter 0609, "Appendix F, Fire Protection SDP." Since this issue affected the 20-foot separation requirement, Phase 2 of the SDP should be performed. However, since there was no ignition source in the areas of concern the inspectors could not develop a realistic fire scenario. Without a realistic fire scenario, this issue could not be evaluated using Phase 2 of the fire protection SDP and was therefore considered to be of very low risk significance (Green). This issue is dispositioned in Section 4AO7 of this report.

.9 <u>(Closed) LER 50-266; 50-301/1999-008-00</u>: Postulated fire could lead to loss of redundant trains of charging pumps. This licensee-identified deficiency involved a lack of protection for cables associated with volume control tank and reactor water storage tank outlet valves such that charging pumps may not be available to meet the performance goals specified in 10 CFR Part 50, Appendix R, Section III.L.2.b. This was determined to have very low safety significance (Green).

The licensee identified that in eight fire zones, the cables associated with volume control tank and reactor water storage tank outlet valves were routed in the same areas. There would be insufficient time to take manual actions to prevent failure of any running charging pump (credited for post-fire safe shutdown purposes).

Since this issue may have a credible impact on safety and involved a degradation of the defense-in-depth element, the inspectors evaluated the issue using NRC Manual Chapter 0609, "Appendix F, Fire Protection SDP." Phase 1 and 2 evaluations of the fire protection SDP were performed because the issue involved degradation of fire barriers such that no fire barriers existed between the normal and the alternate suction path to the credited charging pump. Because the charging pumps were not credited in the mitigation of a transient, a transient with the loss of power conversion system, or loss of offsite power, the increase in fire risk due to the loss of redundant charging pumps had very low risk significance (Green). This issue is dispositioned in Section 4OA7 of this report.

.10 <u>(Closed) 50-301/2000-001-00</u>: Replacement of charging pump control power fuse outside Appendix R design basis. This licensee-identified deficiency involved a lack of protection for cables associated with the credited charging pump such that the pump may not be free of fire damage as required by 10 CFR Part 50, Appendix R, Section III.G.1. This was determined to have very low safety significance (Green).

The licensee identified that a fire in Fire Zone 142 or 187 could result in the failure of the fuse supplying the control circuit in the direct current controller cabinet for the credited charging pump. The pump must be running for a fault on the cable to open the fuse. If the transfer from remote to local control was made prior to the fuse opening, the charging pump would remain operable. If the fuse was already blown before the transfer was made, the direct current contactor could not be closed to start the pump until the fuse was replaced.

Since this issue may have a credible impact on safety and involved a degradation of the defense-in-depth element, the inspectors evaluated the issue using NRC Manual Chapter 0609, “Appendix F, Fire Protection SDP.” Phase 1 and 2 evaluations of the fire protection SDP were performed because the issue involved degradation of fire barriers such that no fire barriers existed to protect the control circuit of the credited charging pump. Because the charging pumps were not credited in the mitigation of a transient, a transient with the loss of power conversion system, or loss of offsite power, the increase in fire risk due to the loss of redundant charging pumps had very low risk significance (Green). This issue is dispositioned in Section 4OA7 of this report.

4OA6 Meetings

Exit Meeting

The resident inspectors presented the routine inspection results to Mr. M. Reddemann and other members of licensee management at the conclusion of the inspection on May 8, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

Interim Exit Meetings

Senior Official at Exit:	M. Reddemann
Date:	April 20, 2001
Proprietary	No
Subject:	Access Control to Radiologically Significant Areas ALARA Planning and Controls
Change to Inspection Findings:	No

Senior Official at Exit:	N. Hoefert, Engineering Programs Manager
Date:	April 26, 2001
Proprietary	No
Subject:	Inservice Inspection
Change to Inspection Findings:	No

4OA7 Licensee-Identified Violations

The following findings of very low significance were identified by the licensee and were violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations (NCVs).

If you deny these NCVs, you should provide a response with the basis for denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 2055-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 2055-0001, and the NRC Resident Inspector at the Point Beach facility.

NRC Tracking Number	Requirement Licensee Failed to Meet
NCV 50-301/01-08-04	10 CFR Part 50, Appendix R, Section III.L.2.d, requires the process monitoring function be capable of providing direct readings of the process variables necessary to perform and control safe shutdown functions. Contrary to the above, the licensee failed to provide direct readings of steam generator 'B' pressure parameter which was necessary to perform safe shutdown functions. This issue was entered into the licensee's corrective action program as CR 99-2341 and corrected. This is being treated as a NCV.
NCV 50-266/01-08-05	10 CFR Part 50, Appendix R, Section III.G.2.b, requires separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards. An exemption to this requirement was granted by the NRC, dated July 3, 1985, which stated that the approved alternative was to install fire stops in the intervening cable trays. Contrary to the above, the licensee failed to install the fire stops in the Unit 1 motor control center room in a configuration which would prevent propagation of fire from one redundant train of charging pump cables to another. This issue was entered into the licensee's corrective action program as CR 99-2063. This is being treated as a NCV.
NCV 50-266/01-08-06; 50-301/01-08-06	10 CFR Part 50, Appendix R, Section III.G.2.d, requires separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards inside non-inerted containment. Contrary to the above, redundant cables for several temperature elements and steam generator level instruments were located within 20 feet of each other in the Units 1 and 2 containments. This issue was entered into the licensee's corrective action as CR 00-0569 and is being treated as a NCV.

NCV 50-266/01-08-07; 50-301/01-08-07	10 CFR Part 50, Appendix R, Section III.L.2.b, requires the reactor coolant makeup function be capable of maintaining the reactor coolant level within the level indication in the pressurizer for pressurized water reactors. Contrary to the above, in eight fire zones, the cables associated with volume control tank and reactor water storage tank outlet valves were routed in the same fire areas. There would be insufficient time to take manual actions to prevent failure of charging pumps credited for maintaining reactor coolant level. This issue was entered into the licensee's corrective action program as CR 99-2341 and is being treated as a NCV.
NCV 50-301/01-08-08	10 CFR Part 50, Appendix R, Section III.G.1, requires that fire protection features be provided for systems important to safe shutdown so that one train of systems necessary to achieve and maintain hot shutdown conditions is free of fire damage. Contrary to the above, the licensee failed to provide redundant fusing to protect the control cable associated with the credited charging pump which was necessary for hot shutdown condition and was not free of fire damage. The issue was entered into the licensee's corrective action program as CR 00-0022. This is being treated as a NCV.

KEY POINTS OF CONTACT

Licensee

A. Cayia, Plant Manager
F. A. Flentje, Senior Regulatory Compliance Specialist
D. Gehrke, Nuclear Oversight Supervisor
N. L. Hoefert, Engineering Programs Manager
V. M. Kaminskas, Maintenance Manager
J. Lindsay, General Supervisor - Radiation Protection
R. G. Mende, Director of Engineering
J. Michaelson, NDE Supervisor
B. J. O'Grady, Operations Manager
C. Onesti, Health Physicist
L. D. Pepple, Radiation Protection Supervisor
C. T. Prothero, Principle Engineer - Inservice Inspection
M. E. Reddemann, Site Vice President
R. Repshas, Manager - Site Services
D. D. Schoon, System Engineering Manager
D. Shannon, Radiation Protection Supervisor
S. J. Thomas, Radiation Protection Manager
R. Turner, Inservice Inspection Coordinator
T. Webb, Licensing Manager

NRC

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-266/01-08-01; 50-301/01-08-01	URI	Ultrasonic equipment essential variable tolerances (Section 1R08)
50-266/01-08-02; 50-301/01-08-02	URI	AFW system unavailability time not counted (Section 1R12)
50-301/01-08-03	URI	Unit 2 risk profile underestimated (Section R13)
50-301/01-08-04	NCV	Failure to provide direct readings of steam generator 'B' pressure parameter which was necessary to perform safe shutdown functions (Sections 4OA3.4 and 4OA7)
50-266/01-08-05	NCV	Failure to install the fire stops in a configuration which would prevent propagation of fire from one redundant train to another (Sections 4OA3.6 and 4OA7)

50-266/01-08-06 50-301/01-08-06	NCV	Redundant instrument cables were located within 20 feet of each other in the Units 1 and 2 containments (Sections 4OA3.8 and 4OA7)
50-266/01-08-07 50-301/01-08-07	NCV	Postulated fire could lead to loss of redundant trains of charging pumps (Sections 4OA3.9 and 4OA7)
50-301/01-08-08	NCV	Replacement of charging pump control power fuse outside Appendix R design basis (Sections 4OA3.10 and 4OA7)

Closed

50-266/1998-020-00 50-301/1998-020-00	LER	Unprotected cables in cable spreading room (Section 4OA3.2)
50-266/1999-030-00 50-301/1999-030-00	LER	Assumptions for equipment necessary to maintain hot safe shutdown outside Appendix R design basis (Section 4OA3.3)
50-301/1999-002-00	LER	Red channel of steam generator pressure indication passes though fire zone (Section 4OA3.4)
50-266/1999-04-00 & 01 50-301/1999-04-00 & 01	LER	Fuel oil transfer pump cable in the AFW pump room outside Appendix R design basis (Section 4OA3.5)
50-266/1999-007-00	LER	Cable tray fire stops do not meet Appendix R exemption requirements (Section 4OA3.6)
50-266/1999-010-00 50-301/1999-010-00	LER	Inadequate Appendix R emergency lighting (Section 4OA3.7)
50-266/2000-004-00 50-301/2000-004-00	LER	Potential loss of process monitoring instrumentation due to a fire in containment (Section 4OA3.8)
50-266/1999-008-00 50-301/1999-008-00	LER	Postulated fire could lead to loss of redundant trains of charging capacity (Section 4OA3.9)
50-301/2000-001-00	LER	Replacement of charging pump control power fuse outside Appendix R design basis (Section 4OA3.10)
50-301/01-08-04	NCV	Failure to provide direct readings of steam generator 'B' pressure parameter which was necessary to perform safe shutdown functions (Sections 4OA3.4 and 4OA7)
50-266/01-08-05	NCV	Failure to install the fire stops in a configuration which would prevent propagation of fire from one redundant train to another (Sections 4OA3.6 and 4OA7)

50-266/01-08-06 50-301/01-08-06	NCV	Redundant instrument cables located within 20 feet of each other in the Units 1 and 2 containments (Sections 4OA3.8 and 4OA7)
50-266/01-08-07 50-301/01-08-07	NCV	Postulated fire could lead to loss of redundant trains of charging capacity (Section 4OA3.9 and 4OA7)
50-301/01-08-08	NCV	Replacement of charging pump control power fuse outside Appendix R design basis (Sections 4OA3.10 and 4OA7)

Discussed

None

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
ALARA	As-Low-As-Is-Reasonably-Achievable
AMSAC	Anticipated Transient Without Scram Mitigating System Actuation Circuitry
ASME	American Society of Mechanical Engineers
CCHX	Component Cooling Water Heat Exchanger
CFR	Code of Federal Regulations
CL	Checklist
CR	Condition Report
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
HRA	High Radiation Area
ISI	Inservice Inspection
IWP	Individual Work Plan
LER	Licensee Event Report
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NP	Nuclear Power Business Unit Procedure
NRC	Nuclear Regulatory Commission
ORT	Operations Refueling Test
PBNP	Point Beach Nuclear Plant
PI	Performance Indicator
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWP	Radiation Work Permit
SDP	Significance Determination Process
SI	Safety Injection
URI	Unresolved Item
VHRA	Very High Radiation Area

LIST OF DOCUMENTS REVIEWED

<u>1R07</u> <u>Heat Sink Performance</u>

OI 152	HX-012A&B [Unit 1 A and B Component Cooling Water Heat Exchangers] Component Cooling System Heat Exchanger Data Collection Unit 1	Revision 1
Final Safety Analysis Report, Section 9.6	Service Water System	Revision dated June 2000
TIN Number 97-1177	Test Protocol Wisconsin Electric Power Company Point Beach Nuclear Plant HX-12B Component Cooling Water Heat Exchanger	Revision 1
Nuclear Power Business Unit Calculation PGT-99-1416	Point Beach Nuclear Plant Component Cooling Water Heat Exchangers HX 12A and HX 12B Thermal Performance Test Data Evaluation and Uncertainty Analysis	Revision 0
PBNP Drawing 018982	P&ID Auxiliary Coolant System, Sheet 3	Revision E
PBNP Drawing 080034	P&ID Service Water, Sheet 3	Revision E

1R08 Inservice Inspection Activities

99U1-170P001	Ultrasonic Calibration Record, Head to Flange	November 5, 1999
99U1-350P004	Magnetic Particle Examination and Technique Record, Head to Flange	November 4, 1999
99U1-350P021	Magnetic Particle Examination and Technique Record, Closure Stud	November 11, 1999
99U1-352P001	Magnetic Particle Examination and Technique Record, RCP [Reactor Coolant Pump] #1 Seal Bolting	November 5, 1999
99U1-750P009	Visual Examination Record, Valve Bolting 10-in Gate	November 6, 1999
CR 00-252	Weld Surface Examination Records Not Placed in Work Plan	
CR 00-354	Maintenance/NDE [Nondestructive Examination] Concerns	
CR 00-638	Improper Documentation of Corrective Actions for ASME [American Society of Mechanical Engineers] Section XI Leak Test	
CR 00-664	RRM Errors of IWE Exams	
CR 00-784	Weld Exam Performed Correctly but Administratively List Wrong Code	
CR 00-792	Missed VT-2 Examination as Part of WO Process	
CR 00-1423	Drawing Changes Inaccurate	
CR 00-2133	ISI [Inservice Inspection] Weld Document Concern	
CR 00-3322	Safety Injection Piping Not in ISI Program	
CR 00-3777	Classification of Pressurizer Instrumentation	
CR 00-3949	Documentation of NDE not	

CR 01-793	ISI Indications Not Evaluated	
CR 01-1455	Potential Non-Compliance with the Code (ASME Section XI, Appendix VIII)	
CR 01-1446	Loose Parts in Steam Generator 1B	
Drawing ISI-PRI-1104	Primary ISI Isometric PNBP [Point Beach Nuclear Plant] Unit 1 Pressurizer Upper Head	Revision 2
Drawing ISI-PRI-1246	Primary ISI Isometric PNBP Unit 1 Loop B Feedwater Inside Containment	Revision 4
Drawing ISI-PRI-1146	Primary ISI Isometric PBNP Unit 1 Letdown From Loop B Cold Leg	Revision 4
Drawing ISI-PRI-1147	Primary ISI Isometric PBNP Unit 1 Letdown From Loop B Cold Leg	Revision 2
Drawing ISI-PRI-1137	Primary ISI Isometric PBNP Unit 1 Loop B RTD [Resistance Temperature Detector] Bypass	Revision 3
NDE-20	Performance Demonstration Initiative Generic Procedures Table 1 Documents	Revision 0
NDE-161	Manual Ultrasonic Examination of Ferritic and Austenitic Pressure Piping Welds	Revision 17
NDE-172	PDI Generic Procedure For the Ultrasonic Examination of Ferritic Piping Welds	Revision 2
NDE-173	PDI Generic Procedure For the Ultrasonic Examination of Austenitic Piping Welds	Revision 2
NDE-350	Magnetic Particle Examination Alternating Current (AC) Yoke	Revision 18
NDE-451	Visible Dye Penetrant Examination	Revision 17
NDE-750	Visual Examination (VT-1) of Nuclear Power Plant Components	Revision 19

NDE-754	Visual Examination (VT-3) of Nuclear Power Plant Components	Revision 9
NDE-760	Visual Examination of IWE Boundary Components (Metal Containment and Metallic Liners of Concrete Containment)	Revision 2
NDE-163	Manual Ultrasonic Examination of Ferritic Pressure Vessel Welds Greater Than Two Inches in Thickness	Revision 8
Radiographic Examination Record	1SI-857B, Weld 1	November 11, 1999
Radiographic Examination Record	Valve AF-0101A, Welds A&B	November 17, 1999
Radiographic Examination Record	Valve 2CV-1298, Weld 2&3	November 1, 2000
RRM 98-093	Replace the Unit 2 Charging Pump Discharge Valves	July 28, 1998
RRM 99-0047	Remove and Reinstall Valve 1SI-857B	July 20, 1999
WO-9924572	HX-2 Regen HX [Heat Exchanger] Out CHG Isol to RC [Reactor Coolant] Loop A Cold Leg	
WO-9812507	HX-1B SG [Steam Generator] Auxiliary Feedwater First Off Check	
WP-1	Welding Procedure For Carbon Steels Group P-1 to P-1 GTAW-SMAW	Revision 8
WP-2	Welding Procedure For Austenitic Stainless Steels ASME Group P-8 GTAW-SMAW	Revision 6
WP-6	Carbon Steels Group P-1 to P-1 GTAW-Pipe Size Over One Inch OD	Revision 1
	Assessment Quarterly Report and Self-Assessment Report 4Q2000	Not dated

1R12 Maintenance Rule Implementation

Calculation 98-0169	Probabilistic Risk Assessment of Maintenance Rule Availability Performance Criteria and Reliability Performance Criteria	Revision 1
	2000 Annual Report for the Maintenance Rule	March 26, 2001
Section 14.2.4	Point Beach Final Safety Analysis Report, “Steam Generator Tube Rupture”	June, 2000
DBD-07	Main Steam and Steam Dump Systems	Revision 0
Nuclear Power Business Unit Procedure Manual (NP) 7.7.5	Determining, Monitoring and Evaluating Performance Criteria for the Maintenance Rule	Revision 7
	Maintenance Rule (a)(1) Action Plan for the Auxiliary Feedwater System	Revision 1
PBF-7030	Review of Maintenance Rule Performance (Change of Disposition), Auxiliary Feedwater System	December 18, 2000
CR 01-1224	Unavailability Time Not Counted	April 11, 2001
CR 00-4058	Less Than Adequate Performance Criteria for Maintenance Rule	December 6, 2000
CR 01-0641	Control Room Ventilation VNCR Maintenance Rule Performance	March 1, 2001
CR 01-1671	AFW Maintenance Rule Unavailability Hours	May 8, 2001

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

	Weekly Core Damage Risk Profile (Safety Monitor) Unit 2	April 15, 2001
AOP 18A	Train ‘A’ Equipment Operation, Attachment A	Revision 5
	Tag Series List 1R26-005, Unit 1 480V B-3	Revision 0
PBNP IPE [Individual Plant Examination], Section 3.2	Systems Analysis	June 30, 1993
NP 10.3.7	On-Line Safety Assessment, Attachment A	Revision 4
	Selected Safety Monitor Runs for Unit 2 with 1B03 480 Volt Bus Out-of-Service	April 18-20, 2001

CR 01-1380	Risk Profile Underestimated	April 19, 2001

1R14 Personnel Performance During Non-routine Plant Evolutions

AOP 6E	Alternate Boration/Loss of Shutdown Margin	Revision 9
CR 01-1249	1N31 Source Range Instrument Failure	April 12, 2001
PC 74 NNSR	April 24, 2001 Unit 1 Containment Fire Brigade Self-Critique	April 24, 2001

1R17 Permanent Plant Modifications

IWP # MR 01-001	Reconfigure the Intake Crib" with associated drawings	
Safety Evaluation SE 2001-0017	MR 01-001 Intake Crib Reconfiguration	
	Point Beach Safety Evaluation Report dated 7/15/70	
33 CFR Part 64, Section 31	Determination of Hazard to Navigation	

1R20 Refueling and Outage Activities

OP 3C	Hot Shutdown to Cold Shutdown	Revision 83
OP 4D Part 1	Draining the Reactor Coolant System	Revision 55
OP 4F	Reactor Coolant System Reduced Inventory Requirements	Revision 16
OP 5A	Reactor Coolant Volume Control	Revision 32
Final Safety Analysis Report, Section 14.1.4	Chemical and Volume Control System Malfunction	Revision dated June 1999
	Nuclear Power Business Unit Fuel/Insert/Component Movement Authorization for Point Beach Unit	March 16, 2001
Generic Letter 88-17	Loss of Decay Heat Removal	October 17, 1988
Point Beach Response to Generic Letter 88-17	Loss of Decay Heat Removal, Point Beach Nuclear Plant, Units 1 and 2,"	February 2, 1989
NRC Observations on Generic Letter 88-17	Loss of Decay Heat Removal Capabilities, Point Beach Nuclear Plant, Units 1 and 2	October 16, 1989

1ICP 05.064	Reactor Vessel Level Transmitters Outage Calibration	Revision 1
NP 10.3.6	Outage Safety Review and Safety Assessment	Revision 6
U1R26 Outage Safety Assessment	Key Safety Functions	March 26, 2001 and April 11, 2001
	Point Beach Nuclear Plant Shutdown Safety Assessment and Fire Condition Checklist	April 15 and 17, 2001
CR 01-1479	Inadvertent Auxiliary Feedwater AFW Pumps Activation	April 25, 2001
	Point Beach Nuclear Plant Shutdown Safety Assessment and Fire Condition Checklist	April 28, 2001
NPM 2001-0286	Outage Safety Contingency Actions for U1R26 CFCs [Containment Fan Coolers] Out of Service	April 7, 2001
NUMARC [Nuclear Management and Resources Council] 91-06	Guidelines for Industry Actions to Assess Shutdown Management	December 1991
Refueling Procedure RP 1C	Refueling	Revision 46
NP 1.6.6	Work Duration Restrictions	Revision 3
Nuclear Power Business Unit Operations Checklist CL 1E	Containment Closure Checklist - Unit 1	Revision 3

<u>1R22</u> <u>Surveillance Testing</u>

CR 01-1053	Valves Not Locked as Required	
IT 530A	Leakage Reduction and Preventive Maintenance Program Seat Leakage Test of the Train ‘A’ RHR System (Refueling) Unit 1	Revision 7
	CLTR Testing Program Basis Document	Revision 3
OI 65	Post Maintenance Pressure Testing	Revision 20
ORT 3A	Operations Refueling Test, “Safety Injection Actuation With Loss of Engineered Safeguards AC (Train A) Unit 1"	Revision 35

ORT 3B	Operations Refueling Test, “Safety Injection Actuation With Loss of Engineered Safeguards AC (Train B) Unit 1"	Revision 32

20S1 Access Control to Radiologically Significant Areas

Airborne Radioactivity Survey/Isotopic analyses	Containment Samples, No(s). 35-39, 30-68, and 30-69	April 18, 2001
AR #1	Action Item Status Report, High Radiation Area Greater Than 1000 millirem/hour Discovered	April 17, 2001
CR 01-0990	High Radiation Door Propped Open	March 28, 2001
CR 01-1107	High Radiation Area Greater Than 1000 millirem/hour Discovered	April 8, 2001
CR 01-1292	Valve Worker Entered High Radiation Area by Mistake	April 18, 2001
CR 01-1345	Improper Installation of Permanent Lead Shielding	April 18, 2001
CR 01-1346	Valve Team Using Wrong RWP for Work	April 18, 2001
CR 01-1141	At-Power Containment Entry on Wrong RWP	April 10, 2001
Personal Contamination Event (PCE) Reports	Four Contract Employees	April 18,2001
HPIP [Health Physics Implementing Procedure] 1.74	Operation of the Canberra Whole Body Counter	Revision 3, April 13, 2001
NP 4.2.20	Radiation Work Permits	Revision 11, September 22, 2000
REI 24.0	Spent Fuel Pool Instruction/Inventory Sheets	Revision 15, March 25, 1999
Radiation Work Permits	No(s). 15, 103, 117,118, 120,124, 131, and 138	All for current U1R26 outage

2OS2 ALARA Planning and Controls

ALARA (Pre-Job) Reviews, Unit 1	2001-0001 to 2001-0018	For current U1R26 outage

ALARA (Post-job) Reviews, Unit 2	2000-0001 to 2000-0021	For U2R24 outage
CR 00-2571	Locations of Work Areas Not Known	August 25, 2000
CR 00-3480	Worker Sent to Wrong Location	November 11, 2000
CR 00-3815	Unexpected Airborne Activity During Head Lift	November 17, 2000
CR 00-3834	ALARA Weaknesses	November 17, 2000
CR 01-0214	RCS Chemistry Cleanup	January 20, 2001
CR 01-0770	Engineering Work Requests for the ALARA Program	March 12, 2001
CR 01-1318	Outage Dose Exceeding Estimations	April17, 2001
	Dose History/Goals Data	For past outages and current outage U1R26
	Inservice Inspection Plan for U1R26	January 20, 2001
LL-U2R24-RP-001 to 012	Outage Activity Critique, Lessons Learned, Unit #2	October 14, 2000 to November 8, 2000
	Outage (U1R26) Update Bulletin	April 17, 2001
HPIP 3.7	Hydrogen Peroxide Addition to the Reactor Coolant System	Revision 0, April 13, 2000
NP 4.2.1,	Plant ALARA Program	Revision 4, March 22, 2000
NP 5.3.1	Condition Reporting System	Revision 17, November 1, 2000
	Schedule of all Outage Activities	For outage U1R26
	Shutdown Chemistry Meeting Notes	February 2, 2001
	Strategic plan for Radiation Dose Reduction	November 7, 2000
NPM 2001-0095, S-A-RP-01-04	Radiation Protection Self-Assessment: RP Station Documentation Assessment	February 1, 2001
NPM 2001-0145, S-A-P-00-01	Radiation Protection Self-Assessment: High Radiation Area Controls Assessment	February 16,2001

4OA3 Event Follow-up

WR 9937540	1 Reactor Coolant Clean/Smear Piping	
	‘A’ Steam Generator Cubicle Fire, “Chlorides, Fluorides, and Sulfates - Targeted Cleanup List”	April 25, 2001
CAMP 592	Chemistry Analytical Methods and Procedure (CAMP), “Dionex Autoion 400 Analysis of Anions and Organic Acids by High Performance Chromatography	Revision 7
CAMP 209	Chloride and Fluoride: Acceptance Test for Surface Contamination	Revision 6
P.S. 84351 NL	Westinghouse Electric Corporation Letter, “Determination of Surface Chloride and Fluoride Contamination Stainless Steel Materials”, Revision 3	June 15, 1973
NP 3.2.2	Primary Water Chemistry Monitoring Program	Revision 9
Section 3.7	Electric Power Research Institute Pressurized Water Reactor Primary Water Chemistry Guidelines, “Control and Diagnostic Parameters, Frequencies and Limits for Startup Chemistry”	Revision 4, Volume 1
	Wisconsin Energy Corporation Chemical Hazard Evaluation System Product Data Sheet for Purple-K	Revision 4
	Wisconsin Energy Corporation Chemical Hazard Evaluation System Product Data Sheet for Foray	
	Airborne Radioactivity Surveys and Radiological Surveys for Unit 1 Containment Following ‘A’ Steam Generator Platform Fire	April 24, 2001
CR 01-1435	Fire In Containment	April 24, 2001