U.S. NUCLEAR REGULATORY COMMISSION

REGION III

REPORT NO. 50-266/301-95013

FACILITY

Point Beach Nuclear Plant Units 1 and 2

License No. DPR-24; DPR-27

LICENSEE

Wisconsin Electric Power Company
231 West Michigan Street - P379
Milwaukee, WI 53201

DATES

October 4, 1995, Through November 16, 1995

INSPECTORS

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12/26/95
Date

AREAS INSPECTED

Routine, unannounced inspections of operations, engineering, maintenance, and plant support were performed. Safety assessment and quality verification activities were routinely evaluated. Follow-up inspection was performed for non-routine events. A special inspection of the licensee's radiological controls was also performed this period.

EXECUTIVE SUMMARY

Operations

- Overall, activities associated with the outage were performed well. Day-to-day performance of operators was very good. However, some events identified weaknesses in the communications and integration of information between various organizations. These weaknesses contributed to the lack of engineering assessment of a valve time delay relay in the containment spray system that was found outside the acceptance range during testing and a reactor coolant system spill during Unit 2 reactor coolant pump maintenance (Sections 1.1, 2.4, 2.5, 4.1.1 and 4.1.2).

- The inspectors were concerned with the events surrounding the performance of the Unit 2 "Safety Injection Actuation With Loss Of Engineered Safeguards AC" concurrently with safeguards system testing for Unit 1. The Unit 2 test procedure and the 10 CFR 50.59 safety evaluation for this test did not allow these tests to be performed at the same time. This is of concern because performing these two tests concurrently required the Unit 1 operator to pay increased attention to the steam generator levels during the portions of ORT 3A that ran the motor driven auxiliary feedwater pumps in addition to maintaining an awareness of the testing being performed by I&C technicians and its potential effect on Unit 1 (Section 1.2).

- Concerns were again noted with weak foreign material exclusion (FME) controls around the "residual heat removal (RHR) pump suction from the containment sump" valves. During past outages, the inspectors noted similar weaknesses with FME controls around these valves.

Maintenance

- Two work orders did not contain foreign material exclusion (FME) controls as required by the Nuclear Procedure governing FME controls resulting in a non-cited violation. Problems with FME controls have been identified by the inspectors previously (Section 2.3).

- A spill of approximately 40 gallons of reactor coolant occurred during reactor coolant pump maintenance due to inadequate isolation of a vent valve (Section 2.4).

- The inspectors continued to identify ladders and other portable equipment left unattended and unsecured in safe shutdown areas. This issue has been noted by the inspectors in several previous inspections (Section 2.1).

Engineering

- Phase 3C Emergency Diesel Generator (EDG) tie-in activities showed engineering effectively implemented lessons learned from past outages (Section 3.0).

- Overall the control of contractors performing engineering activities this outage improved. However, instances occurred that indicate the licensee should continue to place emphasis in this area (Section 3.1).

Health Physics

Overall, the radiation protection program was a strength during the Unit 2 refueling outage as evidenced through low station dose, low contamination levels throughout the station, and a strong ALARA program (Section 4.1). However, the following weaknesses were noted:

- Weak contamination controls were observed during the first two weeks of the outage (Section 4.1.1).

- A high radiation barrier violation occurred when two contractors entered a high radiation area in containment without signing the appropriate radiation work permit or obtaining the proper dosimetry (Section 4.1.2).

Summary of Open Items

Violations: One identified in Section 1.2.
Unresolved Items: Not identified in this report.
Inspector Follow-up Items: One identified in Section 1.3.
Non-cited Violations: Two identified in Sections 2.3 and 4.1.2.

INSPECTION DETAILS

1.0 OPERATIONS

NRC Inspection Procedure 71707 was used in the performance of an inspection of ongoing plant operations.

1.1 Routine Outage and Non-Outage Activities

During this period, inspectors observed routine control room operations and shutdown operations for the Unit 2 refueling. On October 7, Unit 2 came off-line to begin refueling outage U2R21. Operator performance prior to and during the outage was good except for the following:

- The inspectors had concerns with weak FME controls around the valves RH-850 A & B. The valves provide RHR pump suction from the containment sump in the event of a loss-of-coolant accident. During a routine containment inspection the inspectors noted that the screen which covers the valves during normal operation to prevent FME from entering the RHR system had been removed and an FME boundary established. However, material staged around and in the boundary did not meet the requirements of NP 8.4.10, "Foreign Material Control." This issue is of particular concern because inspectors identified and documented a similar concern during Unit 1 refueling outage in IR 50-266/301-95004. This is an example that management expectations concerning FME controls are either not fully understood or implemented by operations personnel. The inspectors will continue to monitor the operators use of FME controls around the spent fuel pool and the reactor cavity during core reload operations for Unit 2.

- During the performance of ORT 6, "Containment Spray Sequence Test Unit 2," 2SI-836A "Spray Additive Tank Train "A" Outlet Valve" opened earlier than allowed by the acceptance range. The time delay relay for this valve "2/926A" was adjusted and retested without any system or component engineering input. Although this was allowed by procedure, no analysis of the "as found" condition was performed by engineering because they were never informed of the condition. This condition was identified by the inspectors during review of ORT 6. Subsequent review of the issue identified no safety concerns with the "as found" condition of the relay. However, this issue is of concern to the inspectors since conditions outside of the acceptance range could involve safety issues and these may not be identified if engineering does not review unacceptable test data. Licensee has updated procedure to require engineering department be notified when the relay is not in the acceptance range.

1.2 <u>Performance of ORT 3A, Revision 30, "Safety Injection Actuation With Loss Of Engineered Safeguards AC Unit 2" Concurrent With Unit 1 Safeguards Testing</u>

On November 7, 1995, operations performed portions of ORT 3A which tested the ATWS Mitigating System Actuation Circuit (AMSAC), Automatic Actuation of the Auxiliary Feed (AF) System on Steam Generator (SG) "LO-LO" Level, and Automatic Trip of 2P-29 and P-38B, AF Pumps, on Low Suction Pressure. Concurrent with ORT 3A, an I&C technician was performing 1ICP-02.001 "Reactor Protection And Emergency Safety Features Analog Quarterly Surveillance Test" for Unit 1. ORT 3A required in the "Initial Conditions" section, Step 3.8, that "All safeguards systems-related work or testing on either or both units is suspended for the duration of this test." The 10 CFR 50.59 Safety Evaluation 95-113 for ORT 3A also stated no safeguards systems work or testing restrictions during the test as the basis for no increase in the probability of occurrence or consequences of an accident previously evaluated in the FSAR.

Performance of Unit 1 safeguards testing concurrent with ORT 3A was discussed in the morning meeting and was agreed to by the DSS, the ORT 3A test director and test coordinator and the DCS present. Also, the I&C technician's analog testing was noted during the pre-job brief to ORT 3A. However, this discussion did not result in a change to the ORT 3A requirements or a reevaluation of Safety Evaluation 95-113.

The inspectors had the following concerns:

- Despite the Safety Evaluation 95-113 and ORT 3A, Step 3.8 restrictions, no formal temporary change was made to ORT 3A nor was a 10 CFR 50.59 review or safety evaluation performed to allow Unit 1 Safeguards Testing to be conducted concurrent with ORT 3A.

- No one from operations and engineering involved in ORT 3A questioned whether the temporary change had been made to ORT 3A to allow concurrent testing or if this affected the 10 CFR 50.59 Safety Evaluation even after the inspector raised concerns about this issue while observing this portion of ORT 3A.

- Performing these two tests concurrently required the Unit 1 operator to pay increased attention to the steam generator levels during the portions of ORT 3A that ran the motor driven auxiliary feedwater pumps in addition to maintaining an awareness of the testing being performed by I&C technicians and its potential effect on Unit 1.

This is a violation of TS 15.6.8.1 which states in part, "The plant shall be operated and maintained in accordance with approved procedures. <u>Major</u> procedures ... shall be provided for the following operations where these operations involve nuclear safety of the plant: ... 7. Surveillance and Testing of safety related equipment." Contrary to the technical specification requirements, 1ICP-02.001 was performed concurrently with ORT 3A (VIO 301-95013-01(DRP)).

1.3 Operability Determinations

The inspectors performed a review of operability determinations made by the licensee in accordance with NP 10.3.2, Justification for Operation (JCO). A concern was identified with JCO 92-003-01, which justified both units to operate at a Fire Power T_{avg} of 570°F in lieu of 573.9°F as described in the FSAR. The inspectors were concerned because no safety evaluation (SE) was performed in accordance with 10 CFR 50.59 to determine if the deviation presented an unreviewed safety question. JCO 92-003-01, is still in effect; however, License Amendment 146, dated October 27, 1993, approved a T_{avg} of 570°F for Unit 2, and the safety analysis accepted by the NRC for the amendment also provided justification for the lower T_{avg} in Unit 1; therefore the inspectors do not have a current safety concern. However between November 1992 and October 1993, no formal SE had been performed in accordance with 10 CFR 50.59. An evaluation was performed by Westinghouse in October 1992 that concluded that both units could be operated at 570°F but the evaluation did not address all of the questions posed by 10 CFR 50.59. Specifically, whether a new accident had been introduced or if the safety margin had been reduced by the change. This issue will be tracked as an open item until the inspectors determine why a safety evaluation in accordance with 10 CFR 50.59 of this JCO has not been performed to date and what, if any, changes the licensee plans to make to the FSAR (IFI 50-266/301-95013-02 (DRP)).

In addition the inspectors are concerned that the licensee does not have adequate procedures in place to ensure that JCOs receive SEs in accordance with 10 CFR 50.59, which could lead to the licensee operating outside of the licensing basis for the plant. The licensee has stated they plan review NP 10.3.2, "Justification for Continued Operation," and make changes as necessary, to ensure SEs are performed when appropriate. The inspectors will continue to review future JCOs to ensure problems of this type do not recur.

1.4 Inspection of the Safety Injection System

The inspectors independently verified that the portion of the Safety Injection System (SI) inside containment, not normally accessible during power operations, was at a good level of readiness to perform its safety function. This portion of the SI system was in good material condition. The plant's Piping and Instrumentation Diagram (P&ID) PB 02 MSIL0134 matched the SI system's as-built configuration. No violations or deviations were identified.

2.0 MAINTENANCE

NRC Inspection Procedures 62703 and 61726 were used to perform an inspection of maintenance and testing activities.

2.1 Maintenance Activities

- The inspectors observed Unit 2 reactor trip breaker maintenance. Overall performance of the activities was good. The mechanics had a strong technical knowledge of the breakers and good engineering support was noted.

- During routine inspections of outage activities, the inspectors noted several ladders and other portable equipment left unattended and unsecured in safe shutdown areas. In addition, the Operations Department made similar observations. This problem has been documented in past inspection reports during the past year. The licensee has a program in place to ensure that portable items are adequately controlled or restrained to prevent seismic interaction hazards to safety-related equipment; however, it does not appear to be adequately implemented. Therefore, additional management attention may be required to ensure this problem is resolved.

- During the outage, the licensee took actions to remove boric acid build up on packing glands and pump seals. During the past year, maintenance work requests had been written for most of the affected valves and pumps because of the leakage; however, the boric acid buildup on the components prevented analysis of the severity of the leakage. Cleaning the affected areas will assist the licensee to ensure that the components with the most significant leakage get repaired in a timely manner. The inspectors view this as a positive effort.

2.2 Containment Accident Fan Cooler Repairs

During the refueling outage the licensee initiated repairs to the service water supply to the 2HX-15B, containment accident fan cooler. The repair involved cutting out a service water pipe elbow that exhibited extensive exterior corrosion and welding in a new elbow. Upon completion of the weld, visual and radiograph inspections were performed. However, the documentation for the visual inspection does not state the results of the inspection. The inspectors previously noted weaknesses involving the documentation of NDE results during the weld repair of glycol cooling to emergency diesel generators G03 and G04.

The documentation discrepancies appear to be a result of weak licensee procedures and communication of expectations between the Maintenance Department and the NDE examiners. The inspectors discussed this concern with the licensee. The licensee stated that the procedure governing the documentation of NDE results is being reviewed and will be updated as necessary to more clearly define expectations for documenting inspection results.

The inspectors will continue to randomly review NDE results to ensure this issue has been resolved.

2.3 FME Controls

During this inspection period the inspectors noted the following weaknesses in FME controls.

- The inspectors noted while watching retubing of sensing lines for the P-38A Auxiliary Feedwater Pump Discharge Pressure Indicator Switch that WO 9506768 did not require FME controls. NP 8.4.10, Step 2.0 notes that "Exclusion of Foreign Material from Plant Components and Systems" applies to all work on safety-related piping systems and components. The inspectors did not have any concerns with foreign material exclusion control maintained by the I&C performing the work. The inspectors discussed this situation with the I&C manager. He acknowledged that FME controls should have been in place for the retubing procedure. Subsequently, a condition report (CR) was written to note this condition. The inspectors will follow-up on the corrective action generated by this CR and improvements in FME controls.

- Work to replace the 2-½", service water elbow was performed under Work Order (WO) 9409483. However, the WO did not contain any requirements to maintain foreign material exclusion (FME) in accordance with NP 8.4.10, Step 7.3.1, which states in part that "Does this work activity involve opening a safety-related system or component? If yes, FME requirements need to be defined and included in the work package." Step 7.3.2, states in part that "Does this work activity involve opening a piping system or component greater than two inches? If yes, FME requirements need to be defined and included in the work package. In either case the work package should have included FME controls. Contrary to NP 8.4.10, WO 9409483 did not contain any FME controls. This is a violation of 10 CFR 50, Appendix B, Criterion V, which states that activities affecting quality shall be prescribed by documented procedures of the type appropriate to the circumstances and shall be accomplished in accordance with these procedures. However, this issue was identified by the licensee and appropriate actions have been taken; therefore this meets the criteria of NUREG 1600, Criterion VII, Paragraph B.1, and will not be cited.

For the past year inspectors have discussed weaknesses in the implementation of FME controls with the licensee. Since the adequacy of FME controls continues to be cyclical, additional management attention in this area is warranted. The inspectors will continue to monitor the licensee's progress in this area.

2.4 Unit 2 Reactor Coolant Pump Maintenance

Near the end of the last Unit 2 operating cycle, the licensee identified increased shaft vibration on the "A" reactor coolant pump (2P-1A). Due to the vibration the licensee decided to perform pump seal and motor maintenance during the U2R21 refueling outage in lieu of waiting until Fall 1996 as scheduled.

During uncoupling of the pump from the motor, the pump coupling did not release from the motor coupling as expected. The couplings had to be jacked apart before the pump shaft could be lowered onto its backseat. Once the motor and seal assembly were removed from the pump the licensee refit the pump and motor couplings and inspected for interference. It was discovered that an interference fit existed between the two couplings and again had to be jacked apart, this time with a force of approximately 52 ft-lbs (6600 lbs. weight equivalency). The licensee used hand polishing techniques to correct the interference problems.

The licensee also discussed this issue with the pump vendor. The vendor stated that a possible problem with the diffuser adapter could have existed. Specifically, one or more of the bolts used to attach the adapter could have failed or loosened. Although this problem could, over time, cause pump degradation, it would not cause catastrophic pump failure or pose a safety concern. The licensee may inspect the diffuser adapter and impeller during U2R22, in Fall 1996.

Seal and motor inspections did not reveal any firm evidence as to the cause of the shaft vibration. However, one shoe on the lower radial bearing of the motor was found to be slightly below tolerance. The licensee feels this was the most likely cause for the vibration. The inspectors did not have any concerns with the licensee's evaluation of the potential causes for the shaft vibration.

On October 12, 1995, while preparing to disassemble the RCP seal, the licensee experienced a spill of approximately 40 gal. of reactor coolant. The spill occurred because a vent valve from the seal was not properly isolated when the refueling cavity was being flooded for fuel movement operations. Although the safety significance of this event was low, the inspectors are concerned that it happened for the following reasons:

- A similar event occurred in March 1995, when inadequate valve isolation caused a spill from the number 2 seal standpipe (see IR 50-266/301/95004, Section 2.b). In both cases, inadequate communications and coordination of work activities between maintenance and operations were significant contributors to the spills.

- It became evident to the inspectors during the followup of this event that maintenance and operations both showed a lack of responsibility for the spill.

The inspectors discussed these issues with licensee management who had similar concerns. In response, licensee management have counselled the operations and maintenance staff on the need to accept ownership of events of this type and prevent their occurrence. The inspectors will continue to monitor the licensee's work planning activities.

2.5 Unexpected Unit 2 Reactor Trip Signal Generated During Reactor Protection and Safeguards Analog Racks Temperature Measurement

A Unit 2 reactor trip signal was generated on October 9, 1995, after an I&C technician had left one channel of Delta Temperature in "test" after performing calibrations on that channel and subsequently placed a second channel into "test." This situation occurred because another I&C technician was performing work which affected the "Reactor Protection and Safeguards Analog Racks Temperature Measurement" testing was stopped to consult with supervision to determine the proper course of action for both technicians. After the supervisor determined that it would be appropriate to continue with analog rack testing, the I&C technician forgot the first channel was in "test" and placed the second channel into "test" making up the trip logic. Unit 2 was in cold shutdown with the reactor trip breakers open, so no movement of rods occurred. This situation is very similar to the unexpected reactor trip signal generated on Unit 1 in the Spring of 1995.

The inspectors discussed this issue with the I&C Manager. The individual responsible was counseled by I&C management. I&C management is also investigating whether the procedures performing analog testing while the reactor trip breakers are open should require only working one channel at a time and not allow bringing in the reactor trip signal. The inspectors were concerned that this had not been addressed as part of the corrective action for the Unit 1 occurrence.

2.6 Inadvertent Start of a Motor Driven Auxiliary Feedwater Pump (P38B) During Testing

On October 26, 1995, during performance of ICP 5.58B-1 "Safeguards Timing Relays Calibration Unit 2 Train B," the I&C technician misaligned Unit 2 Time Delay Relay (TDR) 20 while installing it into its base. Each relay has a 4-by-4 pin matrix that mates with a 4-by-4 socket base. This error placed a voltage potential across two other safeguards time delay relay coils and caused pump P38B to start after the associated relay timed out. The Unit 2 "B" train Residual Heat Removal pump (2P10B) would also have auto-started had the electrical power and automatic controls not been isolated at the time. No auxiliary feedwater (AFW) was delivered to either units' "B" steam generator because the AFW discharge isolation valves were not actuated by this event. The misaligned relay was removed from the incorrect position, tested, and installed in its proper alignment. The AFW pump was secured and restored to a standby condition.

The inspectors discussed this event with I&C management. The inspectors were concerned with the fact that the I&C technician performed this work without having adequate lighting present at the job site.

2.7 Licensee Action on Previously Identified Items

(Closed) Inspection Followup Item (266/301/94011-02): Effects of Dead Zebra Mussels on the Service Water System.

The inspectors previously had concerns that once the licensee began chlorination of the intake system that large zebra mussel kills could introduce large amounts of dead mussels into the service water system. The licensee began chlorination at the intake structure this summer. During the Unit 2 outage the licensee removed approximately 13 cubic yards of mostly dead zebra mussels from the intake surge chamber; however, only small amounts made their way into the forebay. In addition, no significant amounts of dead mussel or remnants were found in the service water system since intake chlorination began. It is unclear why so many zebra mussels were found in the surge chamber and not beyond. The inspectors will continue to follow any migration of zebra mussels into plant equipment; however, this item is considered closed.

3.0 ENGINEERING

NRC Inspection Procedures 37551 and 92700 were used to perform an onsite inspection of the engineering functions. Phase 3C Emergency Diesel Generator (EDG) tie-in activities which tied in EDG G-02 and provided cross-tie capability of EDG G-03 to 2A-06 "Unit 2 "B" train" went very well. Engineering effectively implemented lessons learned from past outages including contractor control problems for this project.

3.1 Control of Contractors

Overall the control of contractors performing engineering activities this outage improved. The inspectors observed various portions of licensee activities to tie-in emergency diesel generator (EDG) G02, to safeguards bus 2A05. Overall the project went very well. Significant improvements were noted in the licensee's control of contractors performing this work. Lessons learned during past EDG tie-ins appear to have been effectively implemented. However, instances occurred that indicate the licensee should continue to place emphasis in this area. Specifically, during routine inspections in the containment, the inspectors twice noted contractors performing seismic qualification inspections in accordance with NRC Bulletin 79-14, standing on piping that was 2" to 2-½". In one instance, the contractors thought they were standing on larger diameter pipe because of the large amounts of insulation on the pipe. The inspectors discussed these issues with the licensee who agreed they did not meet their expectations.

3.2 Oil Leakage From The Inboard Bearing of The Unit 1 "B" Safety Injection Pump 1P-15B

On October 5, 1995, the oiler for 1P-15B emptied in approximately five minutes after the pump had been started for PC-9 part 5 test run. This oiler had emptied previously during the performance of IT-01 on

August 14, 1995. After the oiler emptied during IT-01, a maintenance work request tag was written on the oiler and an operability determination was made. After the oiler emptied during PC-9 part 5, pump 1P-15B was declared inoperable and engineering was called in for assistance. A standpipe was connected to the bearing housing and the oil level was verified to be within the range recommended by the vendor.

Inspection of the bearing housing drain plug revealed that the plug was loosely threaded into the housing. The oiler assembly was also replaced. After reinstalling the drain plug with pipe dope and tightening and replacing the oiler, the pump was run for ten minutes and no change in oiler level was noted.

The inspectors reviewed the operability determination provided by the licensee after the pump was declared inoperable and repaired. Although the inspectors had no concerns with the licensee's operability determination, the inspectors are concerned that more timely action was not taken to address the MWR tag written after IT-01 in August and the oil leaks beneath the bearing housing.

3.3 Licensee Action on Previously Identified Items

(Closed) Unresolved Item 301/93014-02: Disabling SI Pumps Prior to Cold Shutdown

The inspectors noted inconsistencies between plant practice and the Westinghouse analysis for a shutdown LOCA in modes 3 or 4 (WCAP 12476). Both SI pumps were being removed from service prior to the plant being cooled below 350°F. As discussed in Inspection Report 50-266/301/95004, plant operating procedures have since been revised to preclude disabling both SI pumps while in hot shutdown. However, procedures continued to allow disabling a single pump in hot shutdown.

The NRC Staff recently declined to review the analysis in WCAP 12476 because of the very low probability and consequences of a LOCA during modes 3 and 4. Therefore, operation with a single SI pump under these shutdown conditions is acceptable.

(Closed) Violation 266/94002-02: Diesel Generator Inoperability

Maintenance had been improperly performed on the G01 emergency diesel generator causing it to fail on February 3, 1994. Exciter leads had been positioned in a manner that interfered with generator rotation. Post maintenance testing failed to identify the inadequate maintenance.

As corrective action, procedure NP 8.1.3, "Post Maintenance Testing," was revised to require manual rotation of motors and generators to check for freedom of rotation and clearance from obstruction. Procedure RMP 43, "Diesel Annual Inspection," was revised to require manually rolling the generator to verify clearance between the rotor and stator if any leads were lifted during maintenance. The inspectors reviewed these procedures and had no further concerns.

(Closed) Violation 266/301/94013-02: Breach of Containment Integrity During Valve Testing

On July 13, 1994, containment integrity requirements were not met during routine inservice testing of containment sump recirculation valves. Such testing had been routinely performed since 1977. It was not previously known that this testing produced a configuration that violated containment integrity.

As corrective action, the test procedures were revised to require stationing a dedicated operator at the affected valves during testing. The inspectors reviewed these procedures and had no further concerns.

(Closed) LER 266/94-004: Unexpected Automatic Reactor Scram During Hot Control Rod Drop Testing

This event occurred during the 1994 Unit 1 refueling outage, resulting in shutdown bank A control rods dropping from 20 steps. Details appear in Inspection Report 266/305/94008. The cause was attributed to inadequate work control and poor communications. Corrective action included additional training for operators on the specifics of this event and enhancements to the work control system.

(Closed) LER 266/301/94-011: Redundant Decay Heat Removal Requirements Not Met During Refueling Shutdown

This report describes a condition where both residual heat removal pumps were simultaneously secured for a period of about two minutes. The cause was due to an inadequate test procedure which directed securing the running pump prior to starting the other train pump. As corrective action, the procedure was revised to ensure that the second pump is started prior to securing the operating pump. The inspectors reviewed the revised procedure and had no further concerns.

(Closed) LER 301/94-005: Violation of Technical Specification Table 15.3.5-2, Overtemperature δT Minimum Degree of Redundancy Not Achieved

Following replacement of Unit 2 nuclear instrumentation detector N42, improper indication of delta flux ($\delta\phi$) was observed during reactor startup testing. Signals from the upper and lower N42 detectors were found reversed. As documented in Inspection Report 266/305/94022, this violation was not cited because the criteria for mitigation of enforcement sanctions were satisfied.

Initial Investigation Inconclusive

An initial wiring check in the instrument control cabinets indicated that the N42 upper and lower detector leads were properly connected to the appropriate meters. Since Unit 2 was at power, detector connections inside containment could not be checked.

To correct the improper $\delta\phi$ indication, N42 leads were switched in the control cabinet on the premise that the connections were reversed at the detector following its replacement.

Subsequent Investigation Contradicts Initial Findings

During the current Unit 2 refueling outage, technicians checked the cable connections of the N42 detectors inside containment. Unexpectedly, all connections were found properly made up. This prompted technicians to trace the entire circuitry to identify the location of the swapped leads.

The cause was found to be mislabeled connector points in the control cabinets, resulting in the detector leads being reversed. These connector points had been mislabeled since initial plant construction. Since the plant's electrical schematics were based on the erroneous labeling they also were wrong. Technicians had connected the new N42 detector leads in accordance with the erroneous schematics.

The other three Unit 2 instrument connector points were also mislabeled. However, their associated detectors were properly connected. A check of Unit 1 cabinets revealed that Unit 1 labeling was correct.

Since the wiring error was found in the control cabinet where the initial investigation was performed, vice at the detector, it appeared that a more aggressive effort on the part of technicians during the initial investigation could have led to quicker identification of this problem.

Corrective Action

As long term corrective action, Unit 2 nuclear instrument labels and schematics were corrected. A revision was initiated to procedure ICP 10.15, "Replacement of NIS Detector Assembly," to require electronic cable length measurement of each detector lead. This step will verify which detector each lead is connected to. A revision was initiated to procedure RESP 5.1, "Reactor Engineering Tests From 0 percent to 30 percent Power," to provide additional guidance on detector operability testing, including comparison of all four $\delta\phi$ indications. The inspectors reviewed the proposed procedure revisions and had no further concerns.

(Closed) LER 266/301/94-007: Breach of Containment Integrity During Valve Testing

This report describes the July 13, 1994, condition where containment integrity requirements were not met during routine inservice testing of containment sump recirculation valves. Such testing had been routinely performed since 1977. It was not previously known that this testing produced a configuration that violated containment integrity. Because this event reoccurred August 12, 1994, the violation was cited in Inspection Report 266/301/94013.

<u>(Closed) LER 266/94-008</u>: Breach of Containment Integrity During Valve Testing

This report describes the August 12, 1994, condition where containment integrity requirements were not met during routine inservice testing of containment sump recirculation valves. Because this event was a reoccurrence of the July 13, 1994, event discussed above the violation was cited in Inspection Report 266/301/94013.

<u>(Closed) Unresolved Item (266/301/95004-04):</u> Concerns relative to fuse control.

The previous concern was the lack of fuse control and the possibility that fuses of two different interrupting capacity (50,000 amp and 10,000 amp) were used in a safety related breaker. The inspectors determined through interviews and review of procedures that the fuses in question had not been used as first thought. The Point Beach procedure PBNP 3.1.12 Revision 0 December 23, 1993, "Fuse Replacement," in use at the time has been subsequently changed twice and the new procedure is NP 8.4.13, Rev. 1, September 29, 1995. This procedure, if properly implemented, appears adequate. Also, there was a concern that fuses installed prior to the December 1993 issuance of the original fuse replacement procedure, were not being reviewed. The licensee is checking fuses for the correct size and type as maintenance is performed on all breakers. This issue is closed.

4.0 PLANT SUPPORT

NRC Inspection Procedures 71750 and 83750 were used to perform an inspection of Plant Support Activities.

4.1 <u>Radiological Protection</u>

The radiation protection program continues to be a strength as evidenced through low station dose, low contamination levels throughout the station, and a strong ALARA program.

Outage dose as of 11/06/95 was 68 rem (0.68 Sv) which was below the outage goal of 120 rem (1.20 Sv). Estimated end of year dose was near the annual goal of 175 rem (1.75 Sv). The number of personnel contamination events (PCEs) in the outage was considerably below the goal of 54 PCEs (29 as of 11/06/95). Licensee actions to reduce the number of PCEs (after the unanticipated high number of events from the previous outage) appeared successful.

The ALARA program continued to be a strength. A sample review of pre-job and in-process ALARA reviews found them to be thorough and conservative. The Health Physics department was involved early in the work planning process which, in conjunction with ALARA job reviews, appeared to provide effective ALARA planning. The license completed several aggressive ALARA initiatives, including reactor head shielding during head lifts, and permanent shielding for the fuel transfer tube in both units, with anticipated dose savings in excess of 600 mrem (6.00 mSv) per outage.

No violations or deviations were noted in these areas.

4.1.1 Radiation Worker Practices

While performing routine containment inspections the inspectors noted weak contamination controls during first two weeks of the outage. At the time of the observations the containment was considered a controlled area. These included poor contamination control practices such as:

- Protective clothing (i.e., gloves and shoe covers) periodically scattered throughout the Containment Building. It was not always apparent to the inspectors whether these items were unused or used.
- Individuals touching their head and face with potentially contaminated gloves.
- Poor unsuiting practices were observed including removing shoe covers with bare hands, and gloves and shoe covers being dropped next to repositories in lieu of being placed in them.
- Potentially contaminated items inadvertently placed across contaminated area boundaries.

In addition, during inspection activities in the Auxiliary Building, the inspectors observed the drip bag beneath 2 RH-624 "2 HX-11A RHR Heat Exchanger Outlet Control Valve" with approximately one quart of water present. This bag had no drain line installed at the bottom. The inspectors discussed this condition with operations, health physics technicians and the General Supervisor-Health Physics. The inspectors questioned the General Supervisor-Health Physics about inspection of drip bags especially involving bags which may contain leakage due to changing plant conditions. The "A" train of RHR cooling was running to support decay heat removal for the plant.

Approximately two weeks later, the inspectors observed SF-27 "P-12A & B SFP Cooling Pumps To U-6 SFP Demineralizer" with several gallons of water present in the drip bag and the bag about ready to overflow. This drip bag also did not have a drain line installed at the bottom. Drain lines were subsequently installed in the bottom of both drip bags after the inspectors identified the conditions. This last observation by the inspectors was of particular concern because of the previous discussions with health physics and operations regarding water in spray bags. These conditions left uncorrected can result in overflowing of the drip bags and spread of contamination in uncontaminated areas.

No significant personnel contaminations could be attributed to the poor contamination control practices. However, management should review these concerns to ensure that they do not lead to the spread of contamination during work evolutions involving higher contamination levels such as dry cask storage activities and steam generator replacement.

4.1.2 High Radiation Barrier Violation

On November 3, 1995, the licensee identified that two contractors entered a high radiation area in containment without signing the appropriate radiation work permit or obtaining the proper dosimetry which was required for the entry. The workers were in the area for less than one minute and did not receive any significant radiation exposure. Poor communication, to the workers by their supervision, of a change in radiological conditions in the containment appears to be the main contributing cause of the event. The individuals and their supervision were counseled on the licensee's expectations for work in changing radiological environments. This is a violation of Technical Specification 15.6.11, which describes the radiation protection program and requirements at PBNP. However, this violation is not being cited because the criteria specified in NUREG 1600, Criterion VII, Paragraph B.1 were met.

5.0 PERSONS CONTACTED AND MANAGEMENT MEETINGS

The inspectors contacted various licensee operations, maintenance, engineering, and plant support personnel throughout the inspection period. Senior personnel are listed below.

At the conclusion of the outage radiological protection (RP) inspection on October 27, 1995, the RP inspectors met with licensee representatives (denoted by •) and summarized the scope and findings of their inspection activities.

At the conclusion of the inspection on November 20, 1995, the inspectors met with licensee representatives (denoted by *) and summarized the scope and findings of the inspection activities. The licensee did not identify any of the documents or processes reviewed by the inspectors as proprietary.

* G. J. Maxfield, Plant Manager
*•A. J. Cayia, Production Manager
•J. E. Anthony, Quality Assurance Manager
*•F. A. Flentje, Administrative Specialist
W. B. Fromm, Sr. Project Engineer - Plant Engineering
•P. B. Tindall, Health Physics Manager
* C. M. Gray, Duty Shift Superintendent
*•T. C. Guay, Health Physics Supervisor

L. D. Halverson, Site Services Manager
F. P. Hennessy, Manager - Chemistry
W. J. Herrman, Sr. Project Engineer - Construction Engineering
N. L. Hoefert, Manager - Production Planning
* T. J. Jessesky, Sr. Project Engineer - Quality Verification
* J. A. Palmer, Manager - Maintenance
S. A. Patulski, Nuclear Engineering Manager
J. C. Reisenbuechler, Manager - Operations
*•D. D. Schoon, Regulatory Services Manager
* J. G. Schweitzer, Maintenance Manager
* R. D. Seizert, Training Manager
* G. R. Sherwood, Manager - Instrument & Controls
T. G. Staskal, Sr. Project Engineer - Performance Engineering
M. F. Baumann, Manager - Nuclear Fuels Services
* P. D. Bronk, Project Manager - ISFSI
K. R. Aundensun, Project Manager - ISFSI
* P. W. Huffman, Project Engineer - System Engineering