U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-266/90022(DRP); 50-301/90022(DRP)

Docket Nos. 50-266; 50-301

Licenses No. DPR-24; DPR-27

Licensee: Wisconsin Electric Company
231 West Michigan
Milwaukee, WI 53201

Facility Name: Point Beach Units 1 and 2

Inspection At: Two Rivers, Wisconsin

Dates: October 16 through December 2, 1990

Inspectors: C. L. Vanderniet
J. Gadzala

Approved By: I. N. Jackiw, Chief
Reactor Projects Section 3A

12-24-90
Date

Inspection Summary

Inspection from October 16 through December 2, 1990, (Reports No. 50-266/90022(DRP); No. 50-301/90022(DRP))

Areas Inspected: Routine, unannounced inspection by resident inspectors of outstanding items; operational safety; radiological controls; maintenance and surveillance; emergency preparedness; security; engineering and technical support; and safety assessment/quality verification.
Results: During this inspection period, Unit 1 operated at full power with only requested load following power reductions. Unit 2 completed refueling outage 16 and was started up November 17. The unit achieved a maximum power of 99.7% on November 21 and remained there the remainder of the period. Unit 2 could not reach 100% power due to the extensive steam generator tube plugging during the outage. The Unit 2 outage was carried out well and was ahead of schedule until the unanticipated CRDM seal repairs delayed it.

One violation of NRC requirements was cited in this inspection report:

Reactor vessel water level indication was lost on unit 2 during the outage while in a reduced inventory condition. A mispositioned valve isolated the vessel head vent causing a pressure buildup in the vessel and a false signal

to the variable leg of the level sensor. Additionally, operators used the wrong section of operating procedure OP-5A to raise reactor vessel water level (paragraph 3.f).

A review by the Manager's Supervisory staff, of a temporary change to a procedure, was not done as required by plant procedures. This violation is included in the above mentioned citation as another example of failure to follow procedures (paragraph 9.d).

Issues addressed in this inspection report include:

The status of the plant's corrective actions for a violation involving inadequate corrective actions was reviewed. Weaknesses were found in the plant's implementation of their escalation procedure for overdue open items. This area continues to be closely monitored by the inspector (paragraph 2.e).

Unit 2 experienced a partial inadvertent initiation of the containment recirculation coolers while shutdown during the outage. A technician performing a modification in a safeguards cabinet accidentally bumped the actuation relay for a containment cooler service water return motor operated valve, causing the valve to open. Operators recognized the problem and restored the system to normal (paragraph 3.e).

Three individuals were slightly contaminated while involved in grinding activity on the unit 2 reactor vessel o-ring. Two of these personnel also received minor intakes of radioactive material from the event. Special air samples being taken in support of this work detected the airborne activity and the plant evacuated the workers to prevent further exposure. No exposure limits were exceeded during this event (paragraph 4.a).

Weaknesses were observed in the knowledge and conduct of plant fire watches during a weld repair on a control rod drive mechanism in Unit 2 containment during the outage. The fire watches did not appear adequately briefed and relied only on the normally installed portable fire extinguishers vice having a dedicated portable extinguisher readily available (paragraph 5.a).

Excessive check valve leakage was identified by the plant during conduct of an inter system loss of coolant accident (LOCA) surveillance test on the Unit 2 residual heat removal system during the outage. The leaking valves were repaired and retested satisfactorily (paragraph 5.b).

An inadvertent firearm discharge occurred during shift turnover when a security officer fired a round from his handgun during a routine inspection of the weapon. The bullet impacted a nearby steel plate and shattered. No injuries of any consequence occurred (paragraph 7).

During the Unit 2 outage, 117 tubes were plugged on the two steam generators. This increases the equivalent total plugging (including sleeves) to 10%. Although this is below the plugging limit of 15%, Unit 2 is able to achieve only 99.7% of full power (paragraph 8.b).

A leak on a Unit 2 control rod drive mechanism (CRDM) canopy seal was found during the last outage. The leak was on an interior CRDM whose inaccessibility makes a conventional weld repair impractical. The CRDM vendor was called in to repair the seal with a robotic welder developed for such purposes. The seal was repaired and tested satisfactory (paragraph 8.c).

Weaknesses were noted in the submittal of several licensee event reports (LERs). Reports did not identify the root cause of events and some corrective actions were vague. The utility will submit supplemental reports amplifying the original submittals (paragraph 9.b).

New issues which remain unresolved include:

During an inspection of the service water system, the plant found that silt had accumulated in piping elbows in various portions of the service water system including the suctions to all four auxiliary feedwater (AFW) pumps. Subsequent testing and an evaluation by the plant determined that service water would have been available for AFW suction if called upon. The service water lines were flushed to remove the silt. Development of a preventive maintenance program to deal with silting remains unresolved (paragraph 8.a).

DETAILS

1. Persons Contacted (71707) (30702)

G. J. Maxfield, Plant Manager
*T. J. Koehler, General Superintendent - Maintenance
*J. C. Reisenbuechler, Superintendent - Operations
J. G. Schweitzer, Superintendent - Maintenance
*N. L. Hoefert, Superintendent - Instrument & Controls
W. J. Herrman, Superintendent - Technical Services
T. L. Fredrichs, Superintendent - Chemistry
J. J. Bevelacqua, Superintendent - Health Physics
M. L. Mervine, Superintendent - Training
*R. D. Seizert, Superintendent - Regulatory & Support Services
F. A. Flentje, Administrative Specialist

Other licensee employees were also contacted including members of the technical and engineering staffs, and reactor and auxiliary operators.

*Denotes the personnel attending the management exit interview for summation of preliminary findings.

2. Licensee Action on Previous Inspection Findings (92701) (92702)

a. (Closed) Open Item (266/90010-02; 301/90010-02): Control Board Human Factors Design.

Both unit control boards have valve and pump controllers which are inconsistently labeled. These controllers have a linear scale graduated from 0 to 100 to indicate valve position or pump operation. On most, the 0 position corresponds to the valve being fully shut. On several, however, the convention is reversed. This condition presents the potential for operator confusion and is believed responsible for improper operation of the residual heat removal system in 1989.

Wisconsin Electric is implementing corrective action to include enlarging the size of the "open" and "close" labels on those controllers with the reverse convention scale. For pump controllers, the inappropriate "open" and "close" labels will be replaced with "min" and "max" or "slow" and "fast" labels. The utility considered but rejected a proposal to reverse the paper scale on the reverse convention controller faces so that a 0 indication would always correspond to closed or minimum. The inspector discussed this project's implementation progress with the utility and this item is closed.

b. (Closed) Unresolved Item (266/90019-01; 301/90019-01): Inadvertent Auxiliary Feedwater (AFW) Pump Actuation.

On October 9, 1990, the motor driven AFW pumps were inadvertently started while performing a main steam hydrostatic test on Unit 2 simultaneously with an unrelated maintenance on the Unit 2 "B" train safeguards relay. Inadequacies in the procedure contributed to the operator's failing to verify that the AFW auto initiation logic was not satisfied prior to unblocking the signal. The utility has issued event report 301/90-003 documenting this incident. Corrective actions will be tracked via the event report, therefore this item is closed.

c. (Closed) Unresolved Item (266/90019-02; 301/90019-02): Single Failure Potential on Bus Tie Breakers.

On October 8, 1990, the utility identified a potential for a single failure to cause the tie breakers between the safeguards and non-safeguards electrical busses to accidentally shut. Such an event could cause the emergency diesels to become overloaded. As immediate corrective action, control power fuses for these breakers were removed and their operation is being administratively controlled by the plant. This removes the potential for inadvertent closure. Wisconsin Electric has issued event report 266/90-012 documenting this finding. Permanent corrective actions will be tracked via the event report, therefore this item is closed.

d. (Closed) Violation (301/90019-03): Failure to Issue an Event Report.

Wisconsin Electric did not report a reactor protection system actuation that occurred on October 8, 1989, until this missed report was noted by the inspector. Although the utility has a history of missed commitment dates for event report corrective actions and occasional weaknesses in the material content of some reports, this missed report is considered an isolated incident. The utility has since issued this report which was subsequently reviewed by the inspector. This item is closed.

e. (Open) Violation (266/89033-02; 301/89033-02): Failure to comply with 10 CFR 50 Appendix B, Criterion XVI - Corrective Actions.

The inspector reviewed the plant's progress in implementing Quality Assurance Instruction QAI 16.2, "Open Item Follow-up and Escalation Process for Internally-Identified Deficiencies". The inspector noted several overdue open items which were classified as either priority 1 or 2, that were not escalated as required by the procedure. Several priority 1 items that were escalated, were granted extensions in excess of that allowed in the procedure. The procedure defines priority 1 deficiencies as those involving immediate personnel or nuclear safety issues.

The inspector discussed this concern with utility management. The Quality Assurance group will review the effectiveness of the escalation procedure and consider action to improve compliance with it. This new procedure has been in effect since July 1990 and the majority of corrective actions in response to this violation were to be completed by the end of 1990. This item remains open pending completion of the corrective actions and improved compliance with the escalation procedure.

f. The following items are administratively closed based on a document review of corrective actions and management recommendation for closure:

Violation (266/89004-01; 301/89004-01): Inadequate Design Control

Violation (266/89004-02; 301/89004-02): Lack of Procedures for Analysis of Integral Pipe Attachments per Code Requirements.

3. Plant Operations (71707) [(71710) (93702)]

a. Control Room Observation (71707)

The inspector observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the inspection period. During these discussions and observations, the inspectors ascertained that the operators were alert, cognizant of current plant conditions, attentive to changes in those conditions and took prompt action when appropriate. The inspectors noted that a high degree of professionalism attended all facets of control room operation and that both unit control boards were generally in a 'black board' condition (no non-testing annunciators in alarm condition). Several shift turnovers were also observed and appeared to be handled in a thorough manner.

The inspectors performed walkdowns of the control boards to verify the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components.

On November 30, the inspector noted that a group of indicating lights for the steam dump valves and their associated labels were taped over in preparation for painting the control board. As part of the control room design enhancement project, groups of instruments on the control boards are being identified as belonging to the same system by having the area around them on the control board painted a characteristic color. Although the system involved is not safety related, the inspector nonetheless discussed the propriety of masking over indicating lights on an operating unit with the licensee.

The Plant Manager was observed making periodic tours of the control Room and through the plant. The Vice President, Nuclear, was also observed conducting an extensive tour of the plant.

b. Facility Tours (71707)

Tours of the Unit 2 Containment, Turbine Building, Auxiliary Building, and Service Water Building were conducted to observe plant equipment conditions, including plant housekeeping/cleanliness conditions, status of fire protection equipment, fluid leaks and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance.

During facility tours, inspectors noticed very few signs of leakage and that all equipment appears to be in good operating condition. Plant cleanliness has remained adequate, although certain areas are marginal, such as the boric acid transfer station.

c. Unit 1 Operational Status (93702)

The unit continued to operate at full power during this period with only requested load following power reductions.

At 1655 on October 24, 1990, the licensee discovered a body to bonnet leak on the Pressurizer Steam Space Sample Line Downstream Isolation Valve (1 SC-950B). The leakage was estimated to be approximately 60 drops/minute plus steam vapor. The upstream isolation valve (1 SC-950) was shut to stop the leakage. One SC-950B is accessible and can be repaired while the unit is at power, however, because the leakage has been stopped, repairs are not planned until the next unit outage.

At 1750 on October 24, 1990, the unit 1 reactor operator noticed the turbine EHC panel emergency power supply light come on and go off. Shortly thereafter, the operator noticed a 7 MW increase in the main generator electrical output and immediately reduced load by 7 MW. The Unit 1 EHC system has shown some instabilities recently, however, all of those have resulted in slight drops in generator output. This is the first case of a spurious electrical output increase. Though the increase was slight, the plant is paying close attention to the EHC system and is evaluating the problem.

d. Unit 2 Operational Status (60710) (93702)

The unit began this period in a refueling outage which was extended due to a leak on a control rod drive mechanism seal weld. The inaccessibility of the leak required a special robotic welder never before used in the field. Problems in setting up the welder delayed repair efforts, but eventually the leak was successfully repaired. During a post refueling test, reactor vessel water level indication was lost while raising the level from a reduced inventory condition. This event is discussed in detail in paragraph 3.f.

The unit was taken critical at 1410 on November 17 and placed on-line at 1323 on November 18. Due to the number of steam generator tubes that have been plugged during this and previous outages, the unit is unable to achieve 100% power. Further discussion of the current status of the Unit 2 steam generators in contained in paragraph 8.b. The main turbine governor valves reached full open at 1900 on November 21 with the unit at 99.7% power and has been maintained at this level for the remainder of the period.

e. Partial Inadvertent Initiation of a Safeguards F ture (93702)

On October 31, the licensee notified the NRC via the emergency notification system regarding the partial inadvertent initiation of the containment recirculation coolers.

While pulling wires in a safeguards cabinet during the performance of a modification, a technician apparently bumped safety injection slave relay 2SI-13X, causing it to energize. This resulted in the opening of the isolation valve on the service water return from the containment coolers (2MOV-2907). The additional service water demand resulted in sufficient pressure drop to actuate the containment cooler low water flow alarm. An additional pump was started to maintain service water pressure. Operators recognized the problem and shut the containment cooler isolation valve.

The inspector discussed this event with plant personnel and reviewed the operators' response. No further concerns were raised. The utility submitted event report 301/90-004 detailing this incident.

f. Loss of Reactor Vessel Level Indication While Shutdown (71707)

On November 2, Unit 2 reactor vessel water level was raised from 22% (3/4 height on inlet piping) to 60% in preparation for ORT 3, "Safety Injection Actuation with Loss of AC". Since this test temporarily disables both safety injection pumps, it is required by the procedure that vessel level be greater than 52%.

Unknown to operators was that when the reactor vessel head vent spool piece was reinstalled, one head vent isolation valve (RC 573) remained shut. The result of this was that when vessel level was raised enough to cover the inlet piping (29% indicated level), further level increases caused pressure to rise in the void space at the now isolated top of the vessel. This pressure rise sent a false level signal to the variable leg of the level instrument. This is because the reference leg for vessel level indication is attached to the pressurizer, which became isolated when the water level rose above the top of the inlet piping with the vent valve shut. The indicated level that operators saw was higher than actual level in the vessel.

The shut vent valve was discovered the following day and reopened. Actual vessel level was then raised to 60%. Although indicated vessel level was 59% while the vent valve was shut, actual level was subsequently calculated to have been between 39% and 52%.

Consequently, both safety injection pumps were disabled during ORT 3 with the reactor coolant system still in a mid-loop condition. This is a violation of procedure OP-4F, "Reactor Coolant System Reduced Inventory Requirements" and of procedure ORT 3 (301/90022-01).

The inspector determined that vessel water level could not have gone below 29% without level instrumentation providing a true reading. At this level and below, the coolant piping becomes uncovered, providing a vent path to the pressurizer. Therefore, no danger existed of coolant level dropping below minimum requirements without the operators' knowledge.

A review by the inspector determined that valve RC 573 was shut during performance of procedure RP-1A, "Preparation for Refueling". However, procedure RP-1B, "Recovery from Refueling", fails to direct the restoration of valve RC 573 to the open position. Additionally, two other procedures that involve vessel level changes, OP-4D, "Draining the Reactor Coolant System", and OP-5A, "Reactor Coolant Volume Control", do not require that valves RC 573 or RC 578 (vent isolation valves) be checked open.

The inspector was informed that operators used procedure OP-5A to raise vessel level from 22% to 60%. However, the section of OP-5A that was used explicitly states that its purpose is only for maintaining level at 22% +/- 3%. The licensee is reviewing these procedures to determine the necessary changes to correct them.

These reviews and observations were conducted to verify that facility operations were conducted safely and in conformance with requirements established under technical specifications, federal regulations, and administrative procedures.

4. Radiological Controls (71707)

The inspectors routinely observed the licensee's radiological controls and practices during normal plant tours and the inspection of work activities. Inspection in this area includes direct observation of the use of Radiation Work Permits (RWPs); normal work practices inside contaminated barriers; maintenance of radiological barriers and signs; and health physics (HP) activities regarding monitoring, sampling, and surveying. The inspector also observed portions of the radioactive waste system controls associated with radwaste processing.

From a radiological standpoint the plant is in good condition, allowing access to most sections of the facility. During tours of the facility, the inspectors noted that barriers and signs also were in good condition. When minor discrepancies were identified, the HP staff quickly responded to correct any problems.

a. High Airborne Activity in Containment (71707)

At 2205 on October 24, 1990, the Unit 1 containment was posted as a high airborne activity area as a result of special air samples being taken during the performance of reactor vessel head o-ring replacement. Isotopic analysis of the air sample indicated approximately 7% of MPC was reached for the combined isotopic activity. Based on a review of the work performed and the isotopic composition of the activity, it appears that the airborne activity resulted from the cutting of the old o-ring to facilitate its removal from containment. Three individuals in the vicinity of the work were slightly contaminated and given whole body counts to determine if any uptakes of activity occurred. One of the individuals had a normal count, however, the other two received slightly elevated counts both less than 0.5 MPC hours. The licensee is in the process of conducting a final analysis of this event. The inspector discussed this issue with the licensee and had no further concerns. The containment was reopened at 2344.

All activities were conducted in a satisfactory manner during this inspection period.

5. Maintenance/Surveillance Observation (62703) (61726)

a. Maintenance (62703)

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with technical specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety-related equipment maintenance which may affect system performance.

Portions of the following maintenance activities were observed/reviewed:

- Modification 88-188, Motor-operated valve 2RH-709 upgrade
- Replacement of cell number 35 in station battery D
- Repair of Unit 2 CRDM assembly G-7 lower seal weld.

The inspector interviewed two fire watches assigned to cover this welding and determined that neither had been adequately briefed on their duties and responsibilities. The only fire extinguishers available to the fire watch were those that are normally installed in containment. The inspector discussed this weakness with the licensee, especially since weaknesses in this area have been previously identified. This concern will be discussed at the next periodic management meeting between the licensee and the NRC. Further details regarding this maintenance action are stated in paragraph 8.c.

b. <u>Surveillance (61726)</u>

The inspector observed surveillance testing and verified that testing was performed in accordance with adequate procedures; that test instrumentation was calibrated; that limiting conditions for operation were met; that removal and restoration of the affected components were accomplished; that test results conformed with technical specifications and procedure requirements and were reviewed by personnel other than the individual directing the test; and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspector witnessed and reviewed the following test activities:

- IT-535 (Revision 3): Leakage Reduction and Preventive Maintenance Program Test of the Residual Heat Removal System, Unit 2.

  This is a routine surveillance test conducted during refueling outages per Technical Specification 15.4.4.IV.A.1.(a). The surveillance requires the hydrostatic testing of various portion of the RHR system to 350 psig. While performing this test on the Unit 2, train "B" RHR pump (2P10B) leakage of approximately 6 GPM was identified. The acceptance criteria for the test (Technical Specification 15.4.4.IV.B) states that the maximum allowable system leakage is 2 GPM. The plant determined that the leakage was through the Refueling Water Storage Tank Suction Valve (2 SI-856B). The valve was repaired by replacing the valve disks and a retest was performed with acceptable results before power operation was resumed.

- ORT-3 (Revision 24): Safety Injection Actuation With Loss of Engineered Safeguards AC, Unit 2.

  This is a routine post refueling test which requires a great deal of coordination and communication on the part of the licensee. One minor deficiency was identified by the licensee. This was the failure of the control room ventilation alarm to annunciate on the switching of the control room ventilation to the recirculation mode. This was identified for correction.

- RESP 6.2 (Revision 3): Precision RCS Flow Rate Measurement.

No other discrepancies were noted during the observance of any of the above tests.

6. Emergency Preparedness (71707)

An inspection of emergency preparedness activities was performed to assess the licensee's implementation of the site emergency plan and implementing procedures. The inspection included monthly review and tour of emergency facilities and equipment, discussions with licensee staff, and a review of selected procedures.

The plant conducted a plant evacuation November 30 to exercise the accountability system. Accountability was completed within the required time frame. The inspector observed the drill and was satisfied.

All activities were conducted in a satisfactory manner during this inspection period.

7. Security (71707)

The inspector, by direct observation and interview, verified that portions of the physical security program were being implemented in accordance with the station security plan. This included checks that identification badges were properly displayed, vital areas were locked and alarmed, and personnel and packages entering the protected area were appropriately searched. [The inspector also monitored any compensatory measures that may have been enacted by the licensee.]

Inadvertent Firearm Discharge (71707)

On November 27, a site security officer inadvertently caused a handgun to discharge during a routine inspection of the weapon. The bullet impacted into a nearby metal plate and shattered. One fragment grazed the security officer and caused a minor scratch. No other injuries occurred. The licensee determined that the cause of this event was improper handling of the weapon. The involved individual was counseled and the event was discussed with the remainder of the security force. The inspector discussed this with the licensee and had no further concerns.

All activities were conducted in a satisfactory manner during this inspection period.

8. Engineering and Technical Support (71707)

The inspector evaluated licensee engineering and technical support activities to determine their involvement and support of facility operations. This was accomplished during the course of routine evaluation of facility events and concerns through direct observation of activities and discussions with engineering personnel.

a. Silt Accumulations in Auxiliary Feedwater (AFW) Pump Service Water Suction Supply Lines (71707)

In the licensee's January 12, 1990, response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," (Action Item III) the licensee stated that an evaluation of potential corrosion, erosion, and silting of selected portion of the service water system would be completed by the end of the Unit 2 fall 1990 refueling outage. This evaluation consisted of performing radiography on selected portions of piping and analyzing the results to determine the existence or the degree of piping degradation. While radiographing portions of the emergency service water suction supply to the AFW pumps, the licensee determined that silt had accumulated in piping elbows located upstream of the service water suction isolation valves.

The Point Beach AFW system consists of four pumps, two motor-driven pumps which are shared between units and two unit dedicated steam-driven pumps. The emergency service water piping elbows to each pump had silt accumulations as follows (percent silt indicates percent of piping diameter);

AFW PUMP	AFW supplied to	Percent Silt
steam-driven 1P29	Unit 1 "A&B" S/G	68%
steam-driven 2P29	Unit 2 "A&B" S/G	40%
motor-driven P38A	Both units "A" S/Gs	28%
motor-driven P38B	Both units "B" S/Gs	64%

The plant performed hydrolancing of the service water suctions to remove the accumulated silt. After clearing the 1P29 pump suction the licensee ran a fiber optic boriscope into the service water supply line and discovered the overhead portion of the pipe had a silt accumulation also. The silt accumulation in the overhead piping was approximately 50% and was removed by hydrolancing. This same procedure was repeated on the AFW pumps P38A and P38B however no silt accumulations were found. The licensee radiographed the overhead portion of AFW pump 2P29 and found silt accumulation similar to that found in 1P29. The licensee performed a full flow test of the 2P29 pump service water suction supply line and determined that the silt became suspended in the water and was expelled from the pipe. This means that in the event that if the service water supply had ever been called upon to serve as suction for the AFW pumps the accumulated silt would have passed through the pumps. The licensee has contacted the vendor of the AFW pumps and was told that the pumps were capable of passing the silt without degrading their ability to perform their intended safety function.

The plant also evaluated the possible accumulation of silt in the service water supply to the AFW pump bearing lube oil coolers. This was accomplished by the removal of the cap from the drain connection and the draining of the line. Silt was found in the dead leg from the drain connection to the actual service water supply line however, after a boriscope inspection of the line no further silting was found.

The utility is in the process of determining final corrective actions which will include a means for a regularly scheduled full flow flush of the AFW service water suction supply lines. The plant is also planning to expand the scope of the current service water inspection program. The implementation of this corrective action as well as further examination of the service water system for silt buildup will be followed by the resident staff and remains an unresolved issue (266/90022-02 and 301/90022-02)

b. Steam Generator Tube Plugging (71707)

During the recent refueling outage, eddy current testing on the Unit 2 steam generators revealed the need to plug 117 tubes. This brings the total number of plugged tubes on the two generators to 504 (260 on A and 244 on B). Each generator has 3260 tubes. The equivalent total plugging, including sleeves, is 10%. This remains below the tube plugging limit of 15%, which is based on maintaining the minimum required reactor coolant flow. However, the reduction in heat transfer area prevented the unit from achieving 100% power. The unit was able to reach 99.7% power after startup. Steam generator replacement on this unit is currently planned for 1995. The licensee believes that a larger than expected number of faulty tubes were found during this outage due to the use of a rotating pancake coil to conduct the eddy current testing. This is a relatively new and more sensitive probe than was used during previous testing and thereby able to identify finer defects.

c. Control Rod Drive Mechanism (CRDM) Seal Weld Leak (71707)

A boroscopic examination of the Unit 2 CRDM pressure housing lower canopy seal welds during this outage revealed a leak on assembly G7. The leak was detected by the presence of a small patch of boric acid crystals on the exterior of the seal. This assembly is located at the center of the vessel head and its inaccessibility makes a conventional weld repair impractical. Point Beach contracted Westinghouse to make the necessary repairs using a remote robotic system they have developed. This is the first field application of this repair technology and extensive preliminary work was necessary.

Westinghouse had successfully demonstrated the use of the robotic lower canopy seal welder on a spare four loop (17 by 17) reactor vessel head at the vendor's testing facilities, but never on a two loop reactor vessel head (14 by 14) such as at Point Beach. This required several minor modifications to the welder and its associated equipment to accommodate the more restrictive clearances of a smaller head. The modified equipment successfully prepared the defect for welding and then preformed a weld overlay sealing the leak. The equipment also performed the necessary examination of the new weld remotely thus significantly reducing the potential exposure from a manual repair of the defect.

Point Beach had recently experienced two similar problems on several reactor vessel head penetrations on the Unit 1 CRDM upper pressure housing canopy seal weld assemblies. Other previous leaks have occurred on Unit 2 in a part length CRDM in 1972, two head adapter plug penetrations in 1974 and 1976, and in a Unit 1 CRDM in 1980. These leaks were all on peripheral assemblies and were, therefore, more easily accessible for manual repairs.

All activities were conducted in a satisfactory manner during this inspection period.

9. Safety Assessment/Quality Verification (40500) (90712) (92700)

The licensee's quality assurance programs were inspected to assess the implementation and effectiveness of programs associated with management control, verification, and oversite activities. Special consideration was given to issues which may be indicative of overall management involvement in quality matters such as self improvement programs, response to regulatory and industry initiatives, the frequency of management plant tours and control room observations, and management personnel's attendance at technical and planning/scheduling meetings.

a. Manager's Supervisory Staff Meeting (40500)

The inspector observed session 90-22 of the Manager's Supervisory Staff Meeting. Issues discussed included diesel generator fuel oil cloud point, storage of low level radioactive waste in the steam generator storage facility onsite, and quality control inspection activities. The meeting was conducted in accordance with approved procedures.

b. Licensee Event Report (LER) Review (90712)

The inspector reviewed LERs submitted to the NRC to verify that the details were clearly reported, including accuracy of the description and corrective action taken. The inspector determined whether further information was required, whether generic implications were indicated, and whether the event warranted onsite followup. The following LERs were reviewed:

*266/90-010 Axial Flux Outside Technical Specification Limits (Open)

On August 16, 1990, Unit 1 experienced a load rejection of about 100 MWe due to a malfunction of the turbine governor control. This caused the axial flux differential (AFD) to exceed prescribed limits. The operator used control rods and boration to restore AFD back into the specified band within the allowed 15 minutes using control board indications. The plant process computer, however, indicated that AFD was outside specification for 17 minutes. Since the plant's Reactor Engineering Instructions (REIs) require that the plant computer be used for primary indication, the unit was technically outside the allowable band for longer than the 15 minutes allowed by technical specifications. The faulty governor control has since been repaired.

The inspector reviewed operator logs and the plant computer printout for this event. He also observed control board indications of AFD during rod stroking and noted that the plant computer lags the control board indication by up to several minutes. A discussion with the licensee determined that they are aware of this phenomena. The inspector determined that the operator's actions during the event were correct based on the training received.

The report failed to discuss the inconsistency between operator training for this event and the plant's procedures, which is considered the principal factor in this event. The licensee indicated that they will issue a supplemental report amplifying this inconsistency and specifying whether they will change the procedure or change the training. This item remains open pending issuance of the supplemental report and subsequent review by the inspector.

*266/90-011 (Open) Low NPSH to Containment Spray Pumps with ECCS in Recirculation Mode.

On August 29, 1990, an engineering evaluation determined that, under certain conditions, the residual heat removal pumps cannot provide adequate net positive suction head (NPSH) to the containment spray pumps when the emergency core cooling system (ECCS) is in the recirculation mode. A further evaluation determined that containment spray is not needed in the recirculation mode. Temporary changes have since been made to the emergency procedures directing that containment spray not be used during containment sump recirculation except under specific conditions. These changes are to be made part of a permanent procedure change and the FSAR is to be revised with the most recent evaluation.

The commitment for making the permanent procedure changes and the FSAR update did not specify when this would be completed. The inspector discussed this report weakness with the licensee and will continue to monitor corrective action progress. The adequacy of the utility's technical analysis is being evaluated by the NRC's Office of Nuclear Reactor Regulation and this item will remain open pending completion of that evaluation.

*301/90-002 (Open) Inadvertent Start of Auxiliary Feedwater Pump

On October 9, 1990, the motor driven AFW pumps were inadvertently started while performing a main steam hydrostatic test on Unit 2 simultaneously with an unrelated maintenance on the Unit 2 "B" train safeguards relay. This event is discussed in detail in inspection report 266/90019; 301/90019. Inadequacies in the test procedure contributed to the operator's failing to verify that the AFW auto initiation logic was not satisfied prior to unblocking the signal.

The LER failed to identify the cause of the inadvertent start other than stating that the starting logic had been satisfied. No consideration was given towards addressing either operator error or procedural deficiencies. The report commits to a root cause evaluation to determine additional corrective actions. The licensee indicated that they will issue a supplemental report. This item remains open pending the inspector's review and acceptance of the additional corrective actions.

301/90-003 (Closed) Degradation of Steam Generator Tubes

This report documents the results of the most recent steam generator tube bundle inspection performed on Unit 2 and the subsequent tube plugging information. Details are contained in paragraph 8.b.

*301/90-004 (Closed) Actuation of the Containment Fan Coolers Service Water Valve

This report describes the inadvertent opening of the containment fan coolers service water valve when a technician accidentally bumped the actuation relay. Details are contained in paragraph 3.e.

c. LER Followup (92700)

The LERs denoted by asterisk above were selected for additional followup. The inspector verified that appropriate corrective action was taken or responsibility was assigned and that continued operation of the facility was conducted in accordance with Technical Specifications and did not constitute an unreviewed safety question as defined in 10 CFR 50.59. Report accuracy, compliance with current reporting requirements and applicability to other site systems and components were also reviewed.

d. Missed Review of Temporary Changes to Procedures (40500)

During a review of the plant's Monthly Open Item Status Report, the inspector found that on May 1, 1990, Non-Conformance Report N-90-179 identified several temporary procedure changes which were not reviewed and approved by the Manager's Supervisory Staff within 14 days of issuance. This is a violation of procedure PBNP 2.1.1, "Classification, Review and Approval of Procedures" and Technical Specification 15.6.8.3.B. Discussions with plant staff indicate that these temporary changes were made to procedures RESP 3.1, "Primary System Tests"; RESP 4.1, "Initial Criticality and ARO Physics Test"; and RESP 4.2, "Reference Bank Worth". This violation is being cited along with the other example of failure to follow procedures in paragraph 3.f (301/90022-01).

All other activities were conducted in a satisfactory manner during this inspection period.

10. Outstanding Items (92701)

Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. An unresolved item disclosed during the inspection is discussed in paragraph 8.a.

11. Management Meetings (30702)

A Meeting was held onsite between senior NRC Region III management and Wisconsin Electric senior management on November 19, 1990 to discuss the recently completed Systematic Assessment of Licensee Performance.

12. Exit Interview (71707)

A verbal summary of preliminary findings was provided to the licensee representatives denoted in Section 1 on December 5, 1990, at the conclusion of the inspection. No written inspection material was provided to the licensee during the inspection.

The likely informational content of the inspection report with regard to documents or processes reviewed during the inspection was also discussed. The licensee did not identify any documents or processes as proprietary.