U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-266/90005(DRP); 50-301/90005(DRP)

Docket Nos. 50-266; 50-301 License 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

Inspection Dates March 1 through April 30, 1990

Inspectors: C. L. Vanderniet
J. Gadzala

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

5-18-90
Date

Inspection Summary

Inspection from March 1 through April 30, 1990 (Reports No. 50-266/90005(DRP); No. 50-301/90005(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 in Tavg coastdown until March 31 when it was shutdown for refueling. This was the second consecutive continuous run for this unit. Unit 2 operated at full power with only requested load following power reductions. Issues addressed in this inspection report include: Boric Acid System Walkdown (Paragraph 3.c); Diesel Generator Fuel Oil System Inoperability (Paragraph 3.g); Fuel Handling Incident (Paragraph 3.h); Disregard of a Posted High Radiation Area Boundary (Paragraph 4.a); Emergency Preparedness Exercise (Paragraph 6.a); Electrical Distribution System Deficiencies (Paragraph 8.a); Unit 1 Reactor Vessel Outlet Nozzle Code Rejectable Indications (Paragraph 8.b); Neutron Source Replacement (Paragraph 8.c); Design Basis Reconstitution (Paragraph 9.a); Temporary Waivers of Compliance (Paragraph 9.c); and Fitness for Duty (Paragraph 11). One non-cited violation was identified and reviewed during this inspection period:

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A technician performing surveillance work for which he was not formally qualified (Paragraph 5.b). A new issue which remains unresolved: Inadvertent Auxiliary Feedwater (AFW) Initiation (Paragraph 3.f).

The utility exercised good control over work activities to keep the Unit 1 outage close to schedule. A visual aid, the 'Critical Path Football,' was used to keep work groups aware of who was performing the controlling tasks. Operations of Unit 2 continued in a safe and professional manner during this inspection period. Wisconsin Electric appears to be making significant progress in addressing long standing NRC concerns regarding the adequacy of procedures and control of procedure changes.

DETAILS

1. Persons Contacted (30703) [(30702)]

*J. J. Zach, Plant Manager
E. J. Lipke, General Superintendent, NPE&RS
R. A. Newton, General Superintendent, NSEAS
T. J. Koehler, General Superintendent, Maintenance
*G. J. Maxfield, General Superintendent, Operations
J. C. Reisenbuechler, Superintendent, Operations
W. J. Herrman, Superintendent, Maintenance
N. L. Hoefert, Superintendent, Instrument and Controls
R. J. Bruno, Superintendent, Technical Services
T. L. Fredrichs, Superintendent, Chemistry
J. J. Bevelacqua, Superintendent, Health Physics
R. C. Zyduck, Superintendent, Training
R. E. Heiden, Superintendent, Nuclear Quality Assurance
S. A. Schellin, Superintendent, Reactor Engineering
A. L. Reimer, Superintendent, Nuclear Plant Engineering
D. R. Stevens, Nuclear Specialist
H. J. Gleason, Emergency Planning Coordinator
R. D. Seizert, Regulatory Engineer
*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) Unresolved Item (266/89020-03; 301/89019-03): Piping Exposed to Thermal Expansion Beyond Capability of Piping Support.

The licensee identified a design error in a piping support on the primary side return line from the Residual Heat Removal (RHR) coolers. The maximum temperature used in the original design only considered post loss of coolant accident (LOCA) conditions of 128 degrees F. It did not consider the routine cooldown operations where RHR return temperatures can reach 340 degrees F. An operability evaluation was performed by the utility using the number of cooldowns the plant has experienced since initial startup and the resultant stresses imposed on the supports. Based on the new value for expected stresses, the system was determined to be operable.

Wisconsin Electric reanalyzed the piping supports and determined that 33 were in need of redesign. The principal change was the reworking of support stanchion H-9. Additionally, several piping shims were installed and several spring cans modified with stronger springs. The work was completed on Unit 1 during the current refueling outage under Modification 90-052. It is scheduled to be completed on Unit 2 during the next refueling outage. The inspector reviewed the licensee's analysis, observed the modification to the H-9 support stanchion, and had no further concerns. This item is closed.

b. (Closed) Unresolved Item (266/89024-01; 301/89023-01): Shutoff Head of Fire Pumps Greater Than Fire System Design Pressure.

During a review of a fire system modification package, plant engineers discovered that the 170 psig shutoff head of the fire pumps, is greater than the fire system design pressure of 125 psig. The utility subsequently performed an evaluation of the pressure capability of the entire fire protection system. They verified all components to have at least a 175 psig design rating. Procedure PBNP 3.2.5, 'PBNP Pressure Test Program,' was then revised to reclassify the working pressure of the fire water system components to 175 psig. The hydrostatic test pressure was changed to be 50 psig above the working pressure (225 psig). The inspector discussed the corrective actions with the licensee, reviewed their analysis and the revised test procedures, and had no further concerns. This item is closed.

c. (Open) Unresolved Item (266/89024-02; 301/89023-02): Incorrect FSAR LOCA Analysis Calculations.

During preparation of an instrument line modification package, plant engineers noted that Final Safety Analysis Report (FSAR) LOCA analysis calculations were incorrect. While the FSAR states that one charging pump can make up the loss from a 3/8 inch line break, calculations show that 2.1 pumps would actually be required (each unit has three). The erroneous FSAR statement was used as a basis to exempt 3/8 inch and smaller tubing attached to the Reactor Coolant System (RCS) pressure boundary from being controlled as Quality Assurance (QA) scope.

The licensee performed an operability evaluation of the RCS pressure boundary and determined that the system was operable. Their justification was based on a verification that all 3/8 inch and smaller lines attached to the RCS were built to the same codes and standards as all other RCS piping. As a result, Appendix B of the QA Policy Manual was revised to exempt from QA requirements, only those portions of 3/8 inch and smaller RCS piping which are beyond the first normally shut isolation valve. The basis for only including one valve is that in the unlikely event of both a downstream line break and concurrent valve failure, the resultant RCS inventory loss will be slow enough to allow the operator to respond without activating the emergency core cooling system. A change to the FSAR is being drafted to correct the LOCA analysis calculations for small diameter tubing breaks.

The licensee's reevaluation of the LOCA analysis originally used optimal values for water inventory in the pressurizer. The inspector questioned these values and was informed that Wisconsin Electric engineers became aware of this deficiency and were reworking their calculations using worst case initial conditions. Information from the revised calculations indicate that with a leak from a 3/8 inch line break, time available until pressurizer heaters start uncovering is 5 minutes with one charging pump and 30 minutes with two charging pumps. Further questioning by the inspector revealed that further analysis is required to determine what procedural changes or additional training is needed to prepare the operators to respond to this type event. This item remains open pending completion of the licensee's evaluation and subsequent review by the inspector.

d. (Closed) Unresolved Item (266/89027-02; 301/89026-02): Rod Control Operating Procedures at End of Life (EOL).

Reactor operators were inconsistent in their manner of maintaining control rod position and delta flux at EOL, primarily due to procedural ambiguities regarding this aspect of operation. Acknowledging this deficiency, the plant revised data sheet ROD 1.2 to indicate that the delta flux target values are not applicable during EOL Tavg coastdown operations. Procedure REI 11.0, 'EOL Tavg Coastdown,' was revised to specifically state that all rods be maintained fully out during Tavg coastdown. The inspector reviewed the procedure changes and had no further concerns. This item is closed.

e. (Closed) Unresolved Item (266/89030-01; 301/89030-01): Inadvertent Reactor Trip on B Steam Generator Feedflow/Steamflow Mismatch.

This item's tracking has been transferred to LER 301/89008-00 and its closure discussed in Paragraph 9.d.

f. (Open) Unresolved Item (266/89030-02): Potential Overexposure of a Health Physics Technician.

A licensee Health Physics Technician received a potential overexposure to his left hand on April 3, 1989, while handling a small fuel fragment. This incident was initially identified in Inspection Report No. 266/89030 and discussed in more detail in Inspection Report No. 266/89031. The licensee has since completed their investigation of this incident, a summary of which is presented in Wisconsin Electric Memorandum NEM-90-122 of January 31, 1990.

The final analysis of the fuel fragment determined a gamma activity of 54.2 mCi and a total activity of 142.5 mCi. A micrometer measurement of the fragment revealed its size as 0.419 cm x 0.226 cm x 0.178 cm. The total exposure time was evaluated to be 5.9 seconds.

The evaluation concludes that the Health Physics Technician received a maximum localized extremity dose of 12.4 rem as a result of his exposure to the fuel fragment. This dose is calculated to one square centimeter of tissue at a depth of 7 milligrams per square centimeter and is less than the 18.75 rem quarterly limit for exposure to an extremity as specified in 10 CFR 20.101. This technician received an additional extremity dose of 1.06 rem during the same quarter as this event, bringing his total extremity exposure for that quarter to 13.46 rem. This remains below the allowable quarterly limit.

The NRC performed an independent dose assessment with the aid of Brookhaven National Laboratory (BNL). The maximum extremity dose was determined by BNL to be 13 rem.

The utility formed an event investigation committee to identify any programmatic problems, the associated root causes, and to propose corrective actions. This team provided a list of recommendations which the licensee's memo concurred in. Among the recommendations were implementation of hot particle documentation forms, increasing the number of hot particle retrieval tools available at the worksite, revision of the health physics contractor training module, and reinforcement of the training module through hands on field application aspects.

The inspector reviewed the licensee's analysis of this incident and the implementation of the selected corrective actions. The corrective actions have been implemented and new procedures were in use prior to the start of the current refueling outage.

A remaining concern regards the plant's initial handling of the event. Specifically, communications between technicians and their supervisors, and the initial dose assessment, may have been inadequate. These concerns were investigated by NRC regional specialists the week of April 23rd and this item remains open pending completion of their evaluation. Their findings will be presented in Inspection Reports No. 266/90008 and No. 301/90008.

g. (Closed) Unresolved Item (266/89032-02; 301/89032-02): Inadequate QA Program Implementation - Monthly Open Item Status Report.

The issues addressed by this item have been reclassified as a violation of NRC requirements and incorporated into Violation 266/89033-02; 301/89033-02, '10 CFR 50 App B, Criteria XVI - Corrective Actions Deficiency.' This item is therefore closed.

h. (Closed) Unresolved Item (266/89032-03; 301/89032-03): Information Notice Review Program.

During a routine review of the licensee's response to NRC Information Notices, it appeared that one of the notices had not received adequate review. This raised concerns over the licensee's review program and prompted further inspection in this area.

During the further inspection, conducted in the Milwaukee offices, the inspector evaluated the licensee's program for the review of NRC Information Notices. All notices are channeled and tracked through one office and a status of the notices is maintained on a computer listing. Notices, once received, are assigned an Administrative Punch List (APL) number (for tracking purposes) and are distributed to responsible staff members for evaluation. An individual is assigned to followup on the progress of the evaluations and ensure that the closure of the notices is in a timely manner. The inspector reviewed several notices in the system and it appeared that adequate reviews were being completed. One problem identified by the inspector was the lack of a specific program for handling notices that become delinquent. This issue was discussed with the licensee and will be reevaluated when the review program is reinspected. This item is closed.

i. (Closed) Violation (266/89033-01; 301/89033-01): Failure to provide sufficient independence and separation for the DC distribution system per 10 CFR 50 App A, Criteria 17.

On November 7, 1989, the licensee declared their station batteries technically inoperable in part due to identification of a common mode failure concern which, in a postulated accident scenario, could disable two trains of the DC power supply. The licensee requested and obtained enforcement discretion from the NRC to allow continued reactor operation while the problem was corrected. The DC electrical distribution system was realigned as a near term solution, and new breakers were purchased and installed as a permanent correction of the design deficiency.

An aggravating factor in this issue is that deficiencies in the electrical distribution system design were identified much earlier by the licensee, but a lack of timeliness in the tracking and correction of open items significantly delayed determination of the significance of these deficiencies. The corrective actions deficiencies are cited in Violation 266/89033-02; 301/89033-02, '10 CFR 50 App B, Criteria XVI - Corrective Actions Deficiency.'

However, the hardware issues discussed above have been resolved and this item is closed.

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.

The plant installed a new shutdown status board on the Unit 1 side of the control room for use during this refueling outage. The board is prominently posted and clearly marked with the current status of major plant parameters such as reactor coolant system, low temperature overpressure protection, residual heat removal, safety injection, containment integrity, and others. This board provides the operator with means of rapidly assessing and keeping abreast of changing plant status during outages.

b. Facility Tours (71707)

Tours of the Turbine Building, Service Water Building, Primary Auxiliary Building and Unit 1 Containment 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 few signs of leakage and that equipment appears to be in good operating condition. Plant cleanliness remains good although there appears to be a downward trend (in part due to the current outage work).

c. Engineered Safeguards Features (ESF) System Walkdown (71710)

The inspector performed a detailed walkdown of portions of the Boric Acid system in order to independently verify operability. The Boric Acid system walkdowns included verification of the following items:

- Inspection of system equipment conditions.
- Confirmation that the system check-off-list (COL) and operating procedures are consistent with plant drawings.
- Verification that system valves, breakers, and switches are properly aligned.
- Verification that instrumentation is properly valved in and operable.
- Verification that valves required to be locked have appropriate locking devices.

- Verification that control room switches, indications, and controls are satisfactory.

- Verification that surveillance test procedures properly implement the Technical Specifications surveillance requirements.

Numerous minor deficiencies were identified with this system. Examples include an inadequately secured nitrogen bottle, valve labels detached from their respective valves, a support stanchion which had broken loose from the floor, a locking chain which was detached from its valve, and several conduit boxes which were open with exposed leads protruding. Additionally, much of the area was covered with solidified boric acid from a 345 gallon spill which occurred March 10 when a recirculation pump seal failed. The pump was promptly repaired.

The inspector presented these deficiencies to the licensee and corrective measures were initiated for each. The cause for such an accumulation of deficiencies was also discussed with the licensee and the majority attributed to maintenance work currently being performed on this system. The inspector will review the corrective actions upon their completion and had no further concerns.

d. <u>Unit 1 Operational Status (93702)</u>

The unit entered this period at 92% power and continued Tavg coast down until March 31, when it was shut down for refueling outage 17. The unit completed this fuel cycle with a run of 318 continuous days. The unit completed the previous fuel cycle with a continuous run of 316 days. The last time this unit was off line for other than refueling was November 21, 1987. During this outage, the entire core was offloaded to perform an inspection of the reactor vessel nozzles. Major work performed included the containment integrated leak rate test, steam generator tube plug repairs, reactor coolant pump seal maintenance and motor work, thimble tube eddy current testing, hydro test of the RHR loop suction, and pressurizer power operated relief valve modification.

The utility exercised good control over work activities to keep the outage close to schedule. A visual aid, the 'Critical Path Football,' was used to keep work groups aware of which of the activities currently in progress controlled the outage schedule. The controlling group was given possession of the 'football' as a continuing reminder that their activity lay on the outage critical path. The projected startup date is May 17.

e. <u>Unit 2 Operational Status (93702)</u>

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

f. Inadvertent Auxiliary Feedwater (AFW) Initiation (93702)

On April 5, the licensee informed the NRC via the Emergency Notification System (ENS) of an inadvertent initiation of the AFW system. The plant was preparing to reroute a cable in the Unit 2 turbine driven AFW pump undervoltage start circuit to correct a train separation deficiency identified by an NRC electrical team inspection. During testing associated with this modification, an inadvertent actuation signal was sent to the A electric AFW pump. The actuation was caused by current flow through a circuit test light into a sneak path which developed when the negative DC source was disconnected. This sneak path does not exist or otherwise affect the circuit when the circuit is in its normal configuration.

The push button circuit test light exists only to check if its light bulb is operable. With the negative DC source disconnected, however, pushing the circuit test light sends current through the sneak path, which energizes the AFW pump start solenoid. After disconnecting the negative DC source in the course of performing the modification, the operator was surprised to find a voltage on the test circuit without the light being lit. He pushed the button to test the light bulb and unknowingly started the AFW pump. The cause of this incident appears to have been inadequate review of the circuit diagram in preparation for the modification. This item remains unresolved pending completion of the licensee's analysis and subsequent review by the NRC (301/90005-01).

Injection of AFW did occur but did not produce any perturbations in steam generator water level. The Unit 2 operator took appropriate corrective action to secure the pump and restore the system once he determined AFW operation was not necessary. The inspector reviewed the operator's actions and discussed this incident with shift personnel shortly after its occurrence.

g. Diesel Generator Fuel Oil System Inoperability (93702)

On April 9, the licensee determined that a section of the Emergency Diesel Generator (EDG) fuel oil piping was not seismically qualified. This condition resulted in the EDG fuel oil system and consequently, both EDGs, becoming technically inoperable. These systems are required by Technical Specification 15.3.7 and their inoperability forced Unit 2 into a 3 hour Limiting Condition for Operation (Unit 1 was shutdown for refueling). The plant informed the NRC of this event via the ENS. Wisconsin Electric requested and was granted a Temporary Waiver of Compliance from the NRC for 7 days, during which time the fuel oil system was modified to comply with seismic requirements.

Technical Specification 15.3.7 requires that 11,000 gallons of fuel oil be available. The section of piping declared inoperable is located in the fuel oil pump house and connects the 12,000 gallon emergency fuel tank to the EDG day tanks. During review of an EDG fuel oil system seismic concern identified by an NRC electrical team inspection, the piping in the pump house, which contains no horizontal restraints, was evaluated by the licensee as not being able to perform its support function under the loading conditions imposed by an operating basis seismic event. A failure of this fuel line would prevent the flow of fuel oil from the emergency fuel tank to the EDG. Continued operation in this condition is not permitted by Technical Specifications.

The NRC granted a Temporary Waiver of Compliance from this Technical Specification requirement based on the licensee's compensatory measures and to prevent an unnecessary transient on Unit 2. About 3 hours after declaring the fuel oil system inoperable, Wisconsin Electric acquired two tanker trucks on site containing approximately 7,000 gallons of fuel oil each. The tankers were located on site outside the protected area and their fuel oil analyzed. A verification test was performed to ascertain the ability to pump fuel form the tankers to the EDG day tanks. The outlet valves from the non seismic above ground fuel oil storage tanks were shut and placed under administrative control of the duty shift supervisor. This would prevent flooding the pumphouse with oil in the event a seismic event carries away a section of this piping inside the pumphouse. In the event of an incident requiring the use of the emergency power system, each EDG has adequate fuel in its day tank and base tank to run in excess of 4 hours. The standby tanker truck arrangement was sufficient to allow the plant to begin fuel transfer operations from the tanker to the day tank within that 4 hour period. The tankers remained available for use until the EDG fuel oil system modifications were completed and the system declared to operable on April 16.

Of note is that the diesel driven fire pump also obtains its fuel supply from the underground emergency fuel tank. However, the fire pump has its own tank containing an 8 hour supply of fuel and its fuel usage is minimal compared to the EDGs. The total fuel oil loads of the EDGs and the fire pump are well within the supply capability of the tanker truck arrangement.

The NRC reviewed the licensee's compensatory actions and the seismic qualification modification performed on the fuel oil transfer system. No further concerns were raised on this specific issue, although additional concerns regarding the emergency power system are discussed in Inspection Reports No. 266/90201; 301/90201.

h. Fuel Handling Incident (93702)

On April 21, the site notified the NRC via the ENS about a fuel handling incident. While lifting fuel assembly U-21 from the spent fuel storage rack, the corner fuel pin apparently caught the lip of the rack and was severely bent. An examination performed by an underwater camera revealed that the fuel pin tore through the grid clip assembly and was distorted into a severe 'S' shape. The end cap was sheared off and the fuel pellet retainer spring was visible.

The fuel pellets were exposed to the fuel pool water. The inspector responded to the site and monitored the licensee's examination of the damage. Repeated air samples taken in the fuel pool vicinity revealed no detectable gaseous or iodine activity. Assembly U-21 has completed one fuel cycle in the core.

A log review for this assembly revealed that when it was initially placed in its present location in the fuel pool (SE-46), the operator noticed a slight drop in load (approximately 70 lbs) just prior to the fuel assembly being placed on the bottom. Since such load drops occasionally occur, the operator only made a note in his log and continued with the procedure in progress. Several days later, when this assembly was being lifted for a routine examination, the load increased from the normally expected 1050 lbs and rapidly approached 1200 lbs. The operator immediately ceased pulling but not before the load jumped to approximately 1550 lbs. Pulling was actually stopped by the overload cutout at 1450 lbs. The assembly was lowered back into the rack, examined from a distance with no damage noted, and a second attempt made at pulling it. This attempt was terminated when load reached 1200 lbs. The assembly was then held at this position for detailed observation with an underwater camera.

The licensee speculated that the grid clip assembly caught on the fuel pool rack and was torn when the fuel assembly was initially lowered into the rack. This allowed the corner fuel pin to spring loose and catch the lip of the fuel pool rack when the fuel assembly was subsequently raised. After the licensee examined the damaged assembly with underwater cameras, it was lowered back onto the bottom of its rack. No further movement is planned for this assembly until the licensee has evaluated their options. The utility has arranged for the fuel vendor to produce a reduced enrichment assembly to replace the damaged one. The reduced enrichment assembly will match the reactivity of the damaged assembly and is intended to complete the remaining three fuel cycles originally scheduled for the damaged fuel assembly. The inspector will continue to follow the licensee's actions for this incident.

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. Disregard of a Posted High Radiation Area Boundary (71707)

On April 18, the primary auxiliary building (PAB) operator and a trainee walked through a posted high radiation area while proceeding on their routine rounds. The posted high radiation area encompasses a southern catwalk on the 66 foot elevation of the PAB and exists as such only during fuel movement. When questioned about his actions, the operator stated that he thought his standing radiation work permit allowed entrance into this high radiation area. A check of the individuals' dosimetry revealed no abnormal exposure received. Their dosimeters read 25 and 20 mRem respectively, which is typical of the doses received for this watch station. This incident is discussed in detail in Inspection Reports No. 266/90008; 301/90008.

All other 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:

- Gas Turbine Generator G05 Annual Inspection and Repair of Fuel Pressure Switches

- Service Water Solenoid Valve (AF 4014A) Refurbishment

- Service Water Pump P-38B Bearing Replacement

  On March 20, service water pump P-38B wa' removed from service due to excessive upper motor bearing noise. Inspection of the bearings found no excessive signs of wear, however, small pieces of metal were found in the bearing oil reservoir. It has been speculated that these metal pieces got in to the upper bearing housing while the motor was being rebuilt by Westinghouse in early 1989. The licensee cleaned the bearing housing and replaced the upper motor bearings.

- Unit 1 Pulse to Analog (P/A) Converter Replacement

  During an earlier failure of the Unit 1 electro-hydraulic control (EHC) system, alarms associated with control rod position were actuated. Rod motion occurred after the EHC failure due to a small change in the turbine load, however, the motion was not enough to cause the alarms received. Investigation into the alarm actuation traced the problem to a faulty power supply in the P/A converter. This power supply was replaced and the Unit 1 P/A converter was returned to service after being satisfactorily tested.

b. Surveillance (61726)

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:

- ICP 13.4 (Revision 6) Spec 200 Internals; Subcooling, RV Level, Cont, H2 & High RG RMS

  The technician performing this surveillance, though knowledgeable in this type of calibration methodology, was not certified as qualified for this specific surveillance in accordance with the JPM training program described in Section 8.1 of the Point Beach Training Manual. This is in violation of Procedure TRNG 3.0 (Revision 4) which requires that a technician be qualified prior to performing a task without supervision. The inspector brought this to the licensee's attention who assigned a qualified individual to supervise the unqualified technician complete the assigned task. The licensee also reviewed the work already performed and determined that, based on the current qualification status of the technician, they were confident with the accuracy of the test data he obtained. The licensee also stated their intention to increase vigilance in this area. Since this appears to be an isolated incident, the violation is not being cited because the criteria specified in Section V.A of the Enforcement Policy were satisfied (NCV 301/90005-02). The inspector had no further concerns.

- ICP 4.24 (Revision 15) Calibration Procedure, Nuclear Instrumentation Source Range Channels

- WMTP 12.18 (Revision 1) Containment Integrated Leak Rate Test

  The test commenced at 0200 April 6 and ran for 24 hours due to sufficiently wide data scatter on some of the instruments to preclude a shorter test duration. A 4 hour verification phase followed yielding satisfactory results.

- IT 1190 (Revision 0) 10 Year Hydrostatic Test of the 1P1A RCP Component Cooling Water System

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

6. Emergency Preparedness (71707)

An inspection of emergency preparedness (EP) 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.

a. EP Exercise (71707)

The annual EP exercise was conducted on March 14. The exercise results were satisfactory and showed noticeable improvement over the last exercise. Details of the exercise are contained in Inspection Reports No. 266/90006 and No. 301/90006.

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

7. Security (71707)

The inspectors, by direct observation and interview, verified that portions of the physical security plan were being implemented in accordance with the station security plan. The inspectors also continued to monitor compensatory measures that have been enacted by the licensee.

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. Electrical Distribution System Functional Inspection (71707)

An Electrical Distribution System Functional Inspection (EDSFI) Team was onsite from March 26 through April 6 to evaluate concerns recently raised about electrical distribution train separation and safety parameters. The EDSFI Team discovered several significant deficiencies in the Point Beach emergency power system including improper safety train separation, inadequate emergency diesel generator (EDG) loading calculations, and seismic qualification uncertainties in the EDG fuel oil system. Several of the items were of immediate concern requiring the licensee to take the action discussed in Paragraphs 3.f and 3.g of this report. Additionally, the licensee removed the auto start control power fuses on the running component cooling water pump after finding that both auto start control cable trains were run together.

The NRC's immediate concerns were discussed with the licensee during a conference call on April 11. Wisconsin Electric provided a briefing for the NRC on their operability determinations and status of corrective actions relating to these issues, during the formal exit meeting with the EDSFI Team on April 17. The NRC continues to closely follow developments in this area, details of which are contained in Inspection Reports No. 266/90201 and No. 301/90201.

b. Unit 1 Reactor Vessel Code Rejectable Indications (71707)

Ultrasonic examination of the Unit 1 reactor vessel outlet nozzle to shell welds during the current refueling outage revealed flaws which were sized in excess of ASME Code Section XI allowable limits. These same flaws were originally identified in 1984 and again in 1987. The current examination was intended to reevaluate these flaws with refined flaw sizing techniques using focused beam transducers. Failure of the transducers during the examination prevented this reevaluation and no replacement transducers were readily available.

Wisconsin Electric plotted each of the flaws in accordance with the Westinghouse 'Flaw Evaluation Handbook Point Beach Units 1 and 2 Reactor Vessels' (WCAP-11477). Based upon the plots, the licensee determined that all the flaws will be acceptable by fracture mechanics as provided in IWB-3600 of ASME Code Section XI. The documentation was transmitted to the NRC Office of Nuclear Reactor Regulation in Wisconsin Electric letter PBL 90-0128 dated April 30, 1990. The inspector discussed this issue with the licensee and had no additional concerns.

c. Replacement of Secondary Neutron Source (71707)

In the process of defueling the core, the licensee attempted to remove a secondary neutron source from a depleted fuel assembly. After several tries, it was determined that the source was stuck in the fuel assembly. The licensee decided that rather than risk damaging the source or the fuel assembly by continuing removal attempts, they would abandon the source in the spent fuel assembly. The licensee will correct the problem by installing a replacement secondary source into the core. This replacement source has been in the spent fuel pool since the mid 1970's, however, the licensee has determined that it meets the criteria necessary for installation.

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

9. Safety Assessment/Quality Verification (71707) (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. Design Basis Reconstitution (71707)

The inspector met with the licensee to discuss their program for the reconstitution of design basis documentation. This is a multi year program established by the licensee to construct complete design basis documents for each of the facility's safety-related features, components, and systems. The licensee discussed the methods to be used to determine the design basis and the structure of the organization that will be performing the reviews. The resulting design documents will enhance the licensee's ability to perform detailed design reviews and safety assessments and should be considered a significant improvement over current documentation.

b. Manager's Supervisory Staff Meeting (40500)

The inspector observed the March 20, 1990 session of the Manager's Supervisory Staff Meeting. The main topic of discussion regarded the determination of the proper level of Quality Assurance coverage that should be given to various nonsafety-related components.

c. Temporary Waivers of Compliance (71707)

The resident inspector met with the Plant Manager to discuss the recent temporary waiver of compliance memorandum. Points of the memorandum discussed included Regional waiver of compliance, NRR waiver of compliance, licensee's request, and termination of waiver.

d. 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 and closed:

- 266/88003-01 Overpressure Mitigating System Testing Not in Accordance With Technical Specifications

  This report describes a procedural inadequacy that results in not fully testing the channels required for low temperature overpressure protection (LTOP). In 1988, the licensee discovered that their procedures did not call for testing of the LTOP system automatic control relay. To correct this deficiency, the licensee revised their procedures to include a surveillance callup task sheet within their automated surveillance database to direct testing of the automatic control relay. With this addition, all the relays within the circuit are now being tested.

- 266/88003-02 4160 Volt Undervoltage Relay Not in Accordance With Technical Specifications

  This report describes a procedural inadequacy that results in not fully testing the undervoltage channels for initiation of the turbine driven auxiliary feedwater (AFW) pumps. This issue was identified in the original LER and its correction is addressed in this supplement. The relay contacts which provide a partial logic signal to actuate the matrix output, and the matrix output relays which direct opening of the AFW pump turbine steam supply valve, were not being tested under existent procedures. A maintenance work request was written to completely test the channels pending a revision to the procedures. The procedures governing the relay testing have since been revised to include the heretofore neglected relays. The following procedures were affected: RMP 56, revised 6/20/89; RMP 65, revised 7/07/88; RMP 73, revised 10/19/89; RMP 74, revised 10/19/89; RMP 75, revised 5/02/89; RMP 76, revised 10/19/89; IT-08, revised 10/21/88; IT-09, revised 1/25/89; and ORT 3, revised 8/17/89.

- 266/88010-01 Electrical System Misalignment

  This report describes an incident involving the inadvertent red tagging out of service of an emergency diesel generator while two safeguards busses were tied together, resulting in the busses being considered inoperable. This error was discovered and corrected within 1 hour by restoring the diesel to service. To prevent reoccurrence, a Technical Specification (TS) amendment request (132) was submitted to the NRC on September 22, 1989. This TS change provides descriptions of allowable electrical system configurations during diesel maintenance. The site's daily outage planning meeting has been formalized to enhance the communication between work groups. Procedural controls for safeguards work were reviewed and maintenance call up flow charts were developed for use in a new plant procedure PBNP 3.1.9, 'CHAMPS PM/ST Callups.' PBNP 4.13, 'Equipment Isolation Procedure', was rewritten to require separate approvals for tag sheet preparation and installation authorization. This procedure was also changed to require a formal announcement to control operators prior to tagging out equipment.

- 266/89002-00 Containment Isolation Valve in Excess of Technical Specification Limits.

  On April 12, 1989, during a Type "C" containment leak rate test, Valve 1-370, charging system check valve, was discovered to have leakage in excess of limits cited in Technical Specification 15.4.4.II.B.

  The licensee issued a Maintenance Work Request (MWR) to inspect and if necessary repair the valve. Upon opening the valve it was determined that the valve disc had been rubbing on the valve body due to a bent dowel pin and, therefore was not seating properly and allowing the valve to leak. The licensee straightened the pin, verified proper valve seating, and retested the valve satisfactorily.

- 266/89003-00 Blowdown Sample Isolation Valve Failure to Close on High Radiation Signal.

  On June 1, 1989, during a periodic test of the operation of 1MS-2083 (steam generator blowdown sample isolation valve), the valve failed to close when an artificial high radiation signal was activated. The valve also failed to close when the switch in the control room was activated. Troubleshooting indicated that the failure of the valve to close was due to a faulty solenoid valve controlling air to the valve air operator.

The licensee issued an MWR to replace the solenoid and retest the isolation valve. The solenoid was replaced and the isolation valve tested satisfactorily.

- 301/88001-00 Reactor Trip Due to Malfunction of Instrument Bus Power Supply Mechanical Interlock

This report describes the events that occurred during the transfer of the red instrument bus power supplies which resulted in the tripping of the reactor. The title of the LER implies that the trip occurred due to malfunction of the mechanical interlock, however, this is not the case. The trip was caused by the combined effects of inadequate procedural controls and personnel error. The licensee has since corrected the procedural inadequacies by revising Operating Instruction 37, 'Shifting of Instrument Supply Bus Feeders.' Training has also been provided to the operators covering this event and operator aids have been posted by the power supply transfer switches to preclude the recurrence of this event. The manufacturer of the switching equipment was also consulted to verify no problem with the mechanical interlock installation existed, and none were identified.

- 301/88002-00 Source Range Nuclear Instrumentation Reactor Trip.

This report describes the events leading to a reactor trip during a power decrease to a hot shutdown condition for hot rod drop testing prior to entry into a refueling outage. As reactor power decayed, the source range instruments automatically energized at 1.5×10^{-10} amps on the intermediate range instruments as expected. However, instead of both instruments following the continuing decrease in reactor power, the N32 source range instrument began an upward trend. This was not noticed by the operator and as a result when the N32 instrument reached the high flux trip setpoint of 1×10^{5} CPS, a reactor trip occurred due to the satisfying of the 1/2 logic needed. After the trip, overcooling of the RCS to approximately 516 degrees F occurred due to unanticipated though proper operation of the feedwater regulation valves.

The licensee corrective actions for this event included; revising OP-3B, "Reactor Shutdown", Revision 13; additional training on the effects of Tavg range resistors for operations personnel; and the issuing of a nonconformance report (NCR) to address a missing wire in the source range instrument.

Training of the operations personnel has been completed and the NCR for the missing wire has been closed out. OP-3B, however, was not revised as stated in the LER corrective actions. The licensee determined that it was not necessary to clarify the expected responses of the N32 instrument and associated annunciators in that procedure. This was a management decision that has been done with no formal documentation of the change to the LER's corrective actions. Because no further corrective actions related to this LER are now pending, this LER is considered to be closed.

- 301/89003-01 Safety Injection Accumulator Level Detector Instrument Failure

Further review of this LER was requested by regional management in a memorandum dated February 26, 1990, regarding possible degradation of the environmental qualification of the accumulator level instruments. This review was conducted and no environmental qualification degradation of the level instruments was identified. A memorandum documenting this review and the inspector's conclusions was sent to regional management.

- 301/89005-00 Intermediate Range High Flux Trip Signal

This report describes an unexpected Unit 2 reactor trip signal received from the intermediate range nuclear instrumentation during a refueling shutdown on October 6, 1989. Investigation of the incident revealed a loose connection in a power supply cable for the instrumentation. This loose connection is believed to have caused a momentary signal spike, which caused the trip signal. The connection was tightened and the other Unit 2 nuclear instruments checked. No other loose connections were found. During the current Unit 1 outage, connection tightness was checked on Unit 1 nuclear instrumentation. No loose connections were found. The inspector observed portions of the cable verification and was satisfied.

301/89008-00 Instrument Bus Ground Resulting in Spurious Safeguards Actuation

This report describes a false Unit 2 reactor trip signal which occurred when investigating a wiring discrepancy in the reactor protection system instrument racks on November 3, 1989, while the reactor was shut down for refueling. An original wiring error was considered the root cause of this event. The wiring was actually correct but the cables were incorrectly labeled. This labeling error resulted in power continuing to be supplied to an electrical cabinet thought to be deenergized for work to correct the supposed wiring error. While working in the cabinet, a technician received a mild electrical shock from one of the exposed leads. Suspecting a residual capacitive charge in the circuitry, he brushed the energized lead against the grounded cabinet frame. This created a sufficient voltage spike to generate the trip signal.

The plant stopped further work and reevaluated the actual electrical configuration. The wires were then relabeled under modification 89-041 to indicate their correct identities. System drawings were also verified for accuracy. The technicians responsible for the work and the engineers overseeing it were counselled that grounding low voltage conductors to test for current is an unacceptable work practice.

e. 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.

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

10. Followup of Information Notices (92701)

The effectiveness of the licensee's program for handling Information Notices (IN) was evaluated on a sampling basis. Select INs were examined to verify that the licensee performed reviews for applicability, that they received appropriate distribution at the site and corporate levels, and that scheduling or performance of any necessary corrective actions was conducted. The following INs were examined:

a. (Closed) IN 88-46: Licensee Report of Defective Refurbished Circuit Breakers, Supplements 1, 2, 3, and 4.

The licensee performed a review of all documentation regarding the procurement of electrical equipment called into question by the notice and did not find any material supplied from the vendors listed. The licensee also inspected components in the warehouse and did not identify any that appeared to have been refurbished. These efforts appear to be adequate and this Information Notice is closed.

b. (Closed) IN 88-55: Potential Problems Caused by Single Failure of an Engineered Safety Feature Swing Bus.

The licensee does not utilize an electrical configuration with an engineered safety feature swing bus and therefore, is not susceptible to this type of a single failure. Based on the design of the facility it appears that this Information Notice does not apply to the licensee and is therefore closed.

c. (Closed) IN 88-67: PWR Auxiliary Feedwater Pump Turbine Overspeed Trip Failure.

The licensee disassembled the overspeed trips on both steam-driven Auxiliary Feedwater (AFW) pumps and inspected the internal parts of the tripping mechanism. No problems were identified during the inspections and the steam-driven AFW pumps overspeed trips were tested and performed satisfactorily. The licensee is evaluating increasing the frequency of the inspection of the AFW pump overspeed trips. These actions appear to be adequate and this IN is closed.

11. Temporary Instructions (TI)

(Closed) TI 2515/104 Fitness for Duty

Using the TI for guidance, the implementation of the fitness for duty program (10 CFR 26) was evaluated. The inspector also reviewed the following associated documentation to verify the licensee's conformance with the TI:

The training videotape for general employees, supervisory personnel, and personnel required to perform escort duties was observed. The topics outlined in the TI were generally addressed in the training videotape.

Training records of selected personnel were reviewed to verify attendance at the appropriate training sessions. No deficiencies were noted.

The inspection determined that the licensee has met the requirements of the fitness for duty rule. This TI is closed.

12. 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 3.f.

13. Violations for Which a "Notice of Violation" Will Not Be Issued

The NRC uses the Notice of Violation (NOV) as a standard method for formalizing the existence of a violation of a legally binding requirement.

However, for isolated Severity Level V violations, a NOV normally will not be issued regardless of who identifies the violation provided that the licensee has initiated appropriate corrective action before the inspection ends. A violation of a regulatory requirement identified during the inspection for which a NOV will not be issued is discussed in Paragraph 5.b.

14. Management Meetings (30702)

A meeting was held between NRC Region III management and plant management on April 3, 1990, to discuss items of interest and foster improved communications between the licensee and the NRC. Items of discussion included the recent electrical team inspection, schedule for increasing the nuclear power department staff levels, and licensee open item resolution.

A meeting was held between NRC Region III and Wisconsin Electric senior management on April 27, 1990, to discuss items of mutual interest. The principal topic was the recent electrical team inspection and the utility's corrective actions to the findings thereof.

15. Exit Interview (30703)

A verbal summary of preliminary findings was provided to the licensee representatives denoted in Section 1 on May 1, 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.