U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-266/89030(DRP); 50-301/89030(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 Unit 1 and 2

Inspection At: Two Rivers, Wisconsin

Dates: October 16 through November 30, 1989

Inspectors: C. L. Vanderniet
J. Gadzala

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

12-11-89
Date

Inspection Summary

Inspection from October 16 through November 30, 1989 (Report Nos. 50-266/89030(DRP); No. 50-301/89030(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; safety assessment/quality verification; and management meetings.

Results: During this inspection period, Unit 1 operated at full power, Unit 2 completed work on refueling outage #15 and returned to full power operation on November 28. Other items addressed in this inspection report include: Inadvertent protective system actuations (Paragraph 3.e); Inadequate battery bus design (Paragraph 3.f); Containment closeout and sump inspection (Paragraph 3.g); Steam generator tube plug repairs (Paragraph 8.b); Modification of CRDM and IRPI cables (Paragraph 8.c); Reuse of flawed RCCA (Paragraph 8.d); Inadequate design of RHR relief valve (Paragraph 8.e); Failure to meet commitments to NRC (Paragraph 9.c); and Management meetings (Paragraph 12). Two unresolved issues were identified regarding potential overexposure of a health physics technician (Paragraph 4.a) and an inadvertent reactor trip on "B" S/G feedflow/steamflow mismatch (Paragraph 3.e). Two violations were identified regarding inadequate firewatch (Paragraph 3.h) and inadvertent auxiliary feedwater isolation (Paragraph 3.i).

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DETAILS

1. Persons Contacted (30703)

   *J. J. Zach, Plant Manager
   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 & Controls
   R. J. Bruno, Superintendent, Technical Services
   T. L. Fredrichs, Superintendent, Chemistry
   D. F. Johnson, Superintendent, Health Physics
   R. C. Zyduck, Superintendent, Training
   *J. E. Knorr, Regulatory Engineer
   *F. A. Flentje, Administrative Specialist
   *E. J. Lipke, Nuclear Plant Eng. & Reg. Section, General Superintendent
   *C. W. Krause, Senior Project Engineer, Licensing

   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)

   The licensee has demonstrated an increased awareness and concern over the items in the NRC's Open Item List. This improved attitude was demonstrated in part by the allocation of additional assets to track open items and pursue the various corrective actions required. Consequently, many more items than usual were able to be closed during this period.

   a. (Open) Violation (No. 266/88009-02; No. 301/88009-02): Failure to Follow Equipment Isolation Procedures.

      On April 25, 1988, an instrument and control supervisor submitted a request for equipment isolation to the DSS without specifying the time and date of the planned work activity. The associated tagout was accomplished well before the start of the work and inadvertently rendered the automatic containment ventilation isolation system inoperable during a period when it was required to be in operation.

      The licensee evaluated this event and initiated corrective actions. PBNP 4.13, 'Equipment Isolation Procedure', was revised to include a precaution directing that the effects of a tagout on other plant systems be considered. A commitment to change the Technical Specification (TS) was made by the licensee wherein the operability requirements of the containment purge supply and ventilation system would be clarified in line with the Westinghouse Standardized Technical Specifications. Although the completion date for corrective actions was November 1, 1988, the TS change has yet to be submitted. This item remains open pending determination of TS revision.

b. (Closed) Open Item (No. 266/88023-01): Use of Emergency Diesel Generator Cross Tie Breaker.

Each of the two units at Point Beach has two 480 VAC emergency safeguards busses (B03 & B04) which can receive power from the diesel generators through 4160/480 step down transformers. Busses B03 and B04 have a common, normally open, tie breaker. A problem identified in TS Section 15.3.7 is that no Limiting Condition for Operation (LCO) exists for the case when the bus tie breaker between B03 and B04 is closed. Shutting the tie breaker relaxes the single failure criterion in that two independent trains are connected together.

The licensee submitted TS Change Request 132 on September 22, 1989. This request specifies that a bus is inoperable when the tie breaker is shut and proposes establishment of an 8 hour LCO under such conditions. This item is closed.

c. (Closed) Open Item (No. 266/88023-03; No. 301/88022-03): Containment Ventilation System Modification.

The design and accident air flow rates for both units were found to be below the values stated in the FSAR. Changes were made to the ventilation system including replacement of damper counterweights with lighter ones to increase system air flow. The modification was performed on Unit 1 during the April 1989 outage and completed on Unit 2 during the October 1989 outage. The inspector reviewed the post modification test results in MSSM 89-10 and discussed the modification with the licensee. No concerns were identified. This item is closed.

d. (Closed) Open Item (No. 266/87005-03): Modification of Instrument Bus Inverters.

The licensee initiated Modification Requests 84-227 and 84-228 to replace six instrument bus invertors with ones containing internal static transfer switches and to install static transfer switches in the other six instrument bus inverters. These switches will allow the automatic transfer of an instrument bus power supply to the alternate source upon loss of the normal supply. This will permit isolation of the affected inverter and manual transfer of the instrument bus supply to the backup swing inverter without loss of instrument bus power. Bid requests were submitted the week of September 18, 1989, with a required delivery date of April 15, 1990.

Installation is planned during the 1990 refueling outages for both units. This item is closed.

e. (Closed) Open Item (No. 266/87013-05; No. 301/87012-05): Modification of Test Equipment to Reduce Connection Errors.

On April 4, 1987, a licensee engineer incorrectly connected a test instrument to the rod control cabinet while performing physics testing. As part of the corrective actions, the licensee was to consider the possibility of modifying the test equipment to make connection errors less likely. After completing their evaluation, the licensee generated modification requests 87-192 and 87-193 to acquire four new digital data acquisition units to be permanently mounted in the control room and the rod drive rooms. The licensee is currently writing the specifications with expected installation dates of Fall 1990 for Unit 2 and Spring 1991 for Unit 1. In the interim, instructions were added to the reactor engineering procedures to be followed when connecting test equipment. This item is closed.

f. (Closed) Open Item (No. 266/87017-01): Improper Lineup of Boric Acid Transfer Pump.

On September 10. 1987, an operator improperly lined up the boric acid transfer pump during a routine procedure to obtain a chemistry sample. The operator failed to open the pump suction valve which resulted in damage to the pump from overheating. The licensee evaluated the incident and decided to change the setpoint of the pump cubicle high temperature alarm to 215 deg F. This setpoint change is intended to provide a warning of a possible system misalignment during pump operation. Additional instrumentation to warn of improper operation was considered but decided not to be economically feasible. The inspector reviewed the licensee's documentation of this event and was satisfied. This item is closed.

g. (Closed) Unresolved Item (No. 266/87022-01; No. 301/87023-01): Improper Environmental Qualification of Solenoid Valve Extension Wires.

An audit of the installation of longer pigtails on various solenoid valves revealed that the wire used to extend the pigtails was not properly certified as being environmentally qualified (EQ). Eleven of these valves are required to be EQ. The licensee has since replaced this wire with a type meeting appropriate EQ requirements or sleeved it with approved Raychem sleeving. The inspector reviewed the records of the wire replacement and sleeving and discussed this issue with the licensee. No concerns were raised. This item is closed.

h. (Closed) Open Item (No. 266/88006-01; No. 301/88006-01): Leakage Rate Correction Factors for Different Pressures.

An inspection under Temporary Instruction TI 2515/84 noted that the licensee did not correct the measured leakage value to account for testing at pressures other than normal service pressure. This correction is required to normalize measured leakage data for trending purposes.

The licensee has revised procedures TS-30 and TS-31, 'High & Low Head Safety Injection Check Valve Leakage Test', to require normalizing measured leakage rate based on test differential pressure. The inspector reviewed the procedure revision and has no further concerns. This item is closed.

i. (Closed) Open Item (No. 266/88006-02; No. 301/88006-02): Safety Analysis of Feed Regulating Valves (FRVs) Failure to Close.

The licensee identified the possibility that during a steam line break accident in containment, the heater drain pumps and condensate pumps could continue to provide water to the faulted steam generator if the FRVs failed to isolate as designed. The reactor vendor (Westinghouse) is preparing a modified safety analysis of this accident. Tracking of this issue has been consolidated with LER 266/88-008 and therefore this item is administratively closed.

j. (Closed) Unresolved Item (No. 266/89015-04; No. 301/89014-04): Power Operated Relief Valve (PORV) Nitrogen Supply Modification.

Testing of the Unit 1 PORVs actuation using the backup nitrogen supply demonstrated that the valves could not open in the required two seconds. The pressure from the nitrogen regulator to the valves, was determined to be insufficient to overcome flow restrictions due to the length of the line from the regulators to the valves. A modification to shorten the line corrected the problem on Unit 2 and allowed one of the PORVs on Unit 1 to operate properly but the other one remains slightly out of specification. The licensee continues to evaluate the operation of the slower operating PORV and plans further adjustments. Tracking of this issue has been consolidated with LER 266/89-005 and therefore this item is administratively closed.

3. Plant Operations (71707) (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.

Plant and corporate management were observed making frequent tours of the control room and through the plant, most evident in his tours of containment was the Superintendent - Health Physics.

b. Facility Tours (71707)

Tours of Unit 2 Containment, Turbine Building, and Primary Auxiliary Building (PAB), 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. Overall, plant cleanliness has remained adequate.

c. Unit 1 Operational Status (71707)

The unit continued to operate at full power during this period.

d. Unit 2 Operational Status (71707)

The unit completed refueling outage #15 during this inspection period and was started up November 24. The unit was placed on the grid November 25 and achieved full power November 28. Major work completed during the last part of the outage included completion of the replacement of the B station battery (D06); inservice inspections of the reactor vessel and core barrel; completion of Reactor Coolant Pump seal package work; steam generator tube plug repair, sleeving and additional tube plugging; replacement of the core barrel and refueling of the core; testing of station batteries D105 and D106; and removal and replacement of the Unit 2 equipment hatch.

e. Inadvertent Protective System Actuations (93702)

On October 27, the licensee notified the NRC via the ENS regarding an inadvertent Emergency Safeguards Feature (ESF) actuation of the A train of ESF on Unit 2 while shut down. The actuation occurred on a containment high pressure trip signal generated while testing a modification of the condensate pump trip circuitry.

The cause was attributed to an inadequacy in the test procedure. Containment and control room ventilation auto isolated, train A of auxiliary feedwater and service water started, and G01 diesel generator started as designed. Safety injection pumps received a start signal but did not start due to their control switches being locked out. The licensee's investigation of this event showed that the safeguards test switch position was improperly controlled by the test procedure. The procedure was rewritten to correct this deficiency. This event is an example of the type of procedural deficiencies identified in Inspection Report No. 50-266/89027; No. 50-301/89026 for which the licensee was asked to provide a written response (No. 266/89027-03; No. 301/89026-03).

On November 3, the licensee notified the NRC via the ENS regarding an inadvertent reactor trip signal on Unit 2 while in cold shutdown. The trip was actuated by a B steam generator low feedwater flow (steam flow/feedwater flow mismatch) signal which was caused by an instrument bus wire shorting to ground. The licensee suspected that the cables suppling power from two of the Unit 2 inverters (2Y04 and 2Y03) to their respective instrument racks had been crossed, resulting in each inverter supplying the other's rack. To rectify this problem, the licensee issued two Maintenance Work Requests (MWRs 894659 and 892660) which required disconnecting, meggering, and reconnecting the correct wires to the proper instrument racks. After disconnecting both racks from the inverters, one of the wires was meggered and reconnected. The instrument bus now thought to be supplying the reconnected wire was then energized. The technician proceeded to megger the other supposedly dead wire and in the process, allowed the wire, which actually was live, to contact the grounded frame of the instrument rack. This resulted in an electrical transient which actuated the low feedwater flow trip. Since the scram breakers were already open, no protective action occurred. The initial cause appears to have been mislabeled wires however, the licensee is continuing to evaluate the event.

This item will remain unresolved pending completion of the licensee's investigation and subsequent NRC review (No. 266/89030-01; No. 301/89030-01).

On November 15, the licensee notified the NRC via the ENS regarding an inadvertent reactor trip signal and ESF actuation on Unit 2 while in cold shutdown. The initiating signals were caused by a low-low level in both steam generators. The steam generators had been drained as a precondition for routine maintenance. Test signals were inserted into the steam generator level circuitry during draining to prevent the system from sensing the low level condition and initiating an unnecessary protective action. The test signal generator was plugged into an outlet powered by the control room lighting panel. As part of an unrelated surveillance (ORT-3), the feeder breaker for the control room lighting panel was opened to test the emergency lighting. This cut power to the steam generator level test signal generator and allowed the level circuitry to sense actual level. Since actual level was below the trip setpoints, protective action was initiated. The steam supply valves to the turbine driven auxiliary feedwater pump opened, but with no steam present, the pump did not operate. Since the scram breakers were already open, no protective action resulted from the reactor trip signal. The licensee attributes responsibility for this action to inadequate precautions in the ORT-3 procedure, which is another example of procedural inadequacy. However, the sagacity of powering the test signal generator from a non vital source is also subject to reevaluation.

f. Inadequate Battery Bus Design (93702)

On November 7, the licensee made an ENS report to the NRC declaring three of four station batteries technically inoperable: Battery D106 due to a battery charge in progress and batteries D05 and D06 due to a design deficiency. The design placed non safeguards loads on the safeguards DC busses with insufficient fault current interrupt capability. The circuit breakers supplying these loads only have thermal overload protection which would not trip the breaker on a short duration current fault or on excessively high current faults. Since the non safeguards loads powered by D05 and D06 may not have adequate electrical separation and their supply cables run through common raceways, a high current fault on one load could conceivably discharge both the D05 and D06 batteries. This condition required entry into a three hour Limiting Condition for Operation (LCO). The licensee requested discretionary enforcement of this LCO and was granted an extension for 24 hours.

D106 was shortly returned to service by finishing its post capacity test battery charge and reconnecting it to the bus. Battery D06 was made operable by transferring all its non safeguards loads to battery D05. Those circuit breakers supplying non safeguards DC loads from D05, which are required for continued plant operation, were then replaced with the proper breakers. The remaining non safeguards loads were secured with their breakers left open. This restored battery D05 to operation. After the licensee procures additional new breakers, the secured non safeguards loads will be equipped with the new breakers.

The design deficiency was by the licensee during an audit of a Safety System Functional Inspection (SSFI) of the main DC systems. A detailed explanation is contained in Wisconsin Electric letter VPNPD-89-141 of November 10, 1989. The inspector discussed this event with the licensee including their corrective action plans, verified the electrical alignment that transferred non safeguards loads to battery D05, and observed the circuit breaker replacement and subsequent realignment of loads to restore the batteries to operability prior to expiration of the LCO. The inspector was satisfied with the final electrical arrangement.

g. Containment Closeout (71707)

The inspector performed a detailed walkthrough of the Unit 2 containment immediately prior to reactor startup to evaluate the general condition of the area and to verify proper storage of all equipment. All material brought into the containment for use during the outage (e.g., tools, ladders, hoses, radiation area postings, etc.) were noted to be removed. Most portable equipment and material stored in the containment was found to be adequately secured. A few items were found laying adrift and were brought to the licensee's attention for correction. The containment sump was verified to be unobstructed.

h. Inadequate Firewatch (71707)

During a tour of the Unit 2 containment on November 16, the inspector noted that licensee personnel were removing the circular stair case. The work involved a fair amount of grinding as evidenced by the shower of sparks which attracted the inspector's attention. He reviewed the posted Ignition Control Permit (PBNP 3.4.1), which specified the need for a properly briefed firewatch and portable fire extinguisher. Upon questioning the firewatch, the inspector discovered that no fire extinguisher had been provided and that the firewatch did not know the location of the nearest installed fire extinguisher. The firewatch also did not appear certain of his requirements to have a fire extinguisher. This is a violation of the PBNP 3.4.1 requirements to have a portable fire extinguisher with the firewatch during performance of hot work (No. 301/89030-03).

This incident is similar to a violation of PBNP 3.4.1 noted in a previous inspection wherein a firewatch was not stationed during welding on a diesel generator stack support (No. 266/89027-01; No. 301/89026-01). That violation was not cited because the criteria specified in Section V.A of the Enforcement Policy were satisfied.

i. Inadvertent Auxiliary Feedwater (AFW) Isolation (93702)

On November 20, the licensee discovered that Unit 1 AFW flow transmitters 1F1 4036 and 1F1 4037 were isolated. These flow transmitters provide AFW flow indication in the control room. A review of the event found that the transmitters were isolated for three days.

The cause of the event appears to be a combination of procedural inadequacy and personnel error. On November 17, operators were lining up systems to start procedure RP-6A (Revision 2), 'Steam Generator Crevice Flush (Vacuum Mode)' on Unit 2. Attachment C of this procedure directs that the AFW flow transmitters on the appropriate unit be isolated. It further states that the flow transmitters are located on the 26' elevation of the Primary

Auxiliary Building (PAB). In actuality, the AFW flow transmitters are located as follows:

Unit 1 - 26' elevation
Unit 2 - 8' elevation

When the operator went to isolate the Unit 2 AFW flow transmitters, he commenced his search on the 26' elevation per the procedure. Unable to find them, he continued his search on the 26' elevation of the PAB until he found the Unit 1 AFW flow transmitters and isolated them.

Unit 1 operation with the AFW flow transmitters isolated is a violation of Technical Specifications (TS) Table 15.3.5-5 (No. 266/89030-04). DCS 3.1.3, (Revision 1) 'Technical Specifications Interpretation, AFW Flow Rate', broadens the TS requirement to include all channels of AFW flow as necessary for continued operation.

The purpose of isolating the transmitters during RP-6A was a precautionary measure to protect them from pressure fluctuations. The licensee determined that no damage occurred to the Unit 2 flow transmitters which were consequently left in service during the steam generator crevice flush.

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 noted that during the Unit 2 refueling outage, waste barrels at the access points to the Unit 2 containment would occasionally overflow with discarded protective clothing (plastic booties). Since the barrels were also kept outside the controlled step off pad area, the potentially contaminated booties would overflow onto the floor outside the contamination control barrier. This was brought to the licensee's attention for correction.

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

a. Potential Overexposure of a Health Physics Technician (71707)

On November 15, the licensee notified the NRC via the ENS regarding a potential overexposure to the hand of a Health Physics Technician.

On April 3, 1989, while cleaning the lower refueling cavity on Unit 1 prior to the unit's refueling, a technician found a small fragment (2-3 mm) thought to be a fuel particle. He picked it up with a piece of tape held in his hand, measured it with a radiac and noting the high reading, dropped it in a plastic bag. The technician notified the hot particle program coordinator of his finding but the matter of his holding the particle in his hand somehow became sidetracked.

During a briefing on cleaning the Unit 2 refueling cavity held November 9, technicians were reminded to pick up hot particles with tape attached to the end of a stick, and if they inadvertently held one directly, to promptly obtain a dose estimate. This briefing reminded the overexposed technician that no dose estimate had ever been done on his hand from the April 3 incident. He notified the supervisor-Health Physics, who promptly initiated a dose assessment. The initial estimate on November 15 was reported as 75 Rem to the fingers. This was revised on November 17 to 375 Rem. A reenactment of the event allowed a redetermination of the dose to approximately 20 Rem. The fragment itself, which remains stored onsite, has been analyzed and determined to have a source content of 40 mCi. This incident will be reviewed in detail in a future inspection and remains unresolved pending completion of its evaluation (No. 266/89030-02).

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 and reviewed:

- Unit 2 "B" Steam Generator (S/G) Code Safety Testing

  One of the Unit 2 "B" S/G code safety valves was removed from the main steam line, tested and reinstalled. All work associated with this work item appears to have been completed satisfactorily and no problems were identified.

- Battery D06 Replacement

  This work is identical to the replacement of the "A" station battery which was replaced during the last Unit 1 refueling outage. Installation of the new battery racks and battery cells was completed as was the testing of the new battery. The testing of the new battery was completed as part of Special Maintenance Procedure (SMP) 1015. Testing required the connection of the new D06 battery to a vendor supplied resistor bank and operating the battery at a design load profile for one hour. The initial and second attempts to conduct the test failed due to failure of the resistor bank. A new resistor bank was obtained from the vendor and a third test was run on October 26, 1989. The test consisted of loading the battery to 1237 amps for the first minute, 849 amps for the next 58 minutes and 875 amps for the final minute. The final battery voltage was 106.7 volts which was above the 105 volt acceptance criteria. With the successful testing of the battery completed the temporary battery was disconnected and the "[illegible]" station battery (D06) was returned to service.

- Removal of the Unit 2 Core Barrel

  In preparation for the performance of inservice inspections of the reactor vessel and core barrel, the core barrel was removed from the reactor vessel and placed on a support stand in the refueling cavity. While setting the core barrel down it became misaligned with the support stand and one of the barrel guide keys wedged askew one of the support stand keyways. The licensee noticed the core barrel start to cock as it was being lowered on its support stand and suspended further motion. Though the barrel remained on its stand, the majority of its weight remained supported by the containment polar crane. Attempts to free the wedged key were successful and the barrel was properly aligned and fully lowered onto the support stand. Analysis by the licensee determined that no damage had been caused to the core barrel or the support stand as a result of the misalignment or the actions used to free the wedged key. After the completion of the inservice inspection, the core barrel was replaced in the reactor vessel without further incident.

These items were completed in a professional manner with no discrepancies identified.

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:

- SMP 1037 (Revision 0) D106 Battery Performance Test

  The temperature compensated test discharge rate was 773 amps. Battery voltage dropped to the completion point of 105 volts after discharging for 64 minutes. This corresponds to a 107% capacity factor. Step 3.2.3 of the procedure requires use of a calibrated thermometer to measure cell temperatures. The installed D106 thermometer was used instead.

  The inspector also reviewed the results of the D105 battery test (SMP 1036, Revision 0). Discharge rate was 777 amps and lasted 62 minutes. Battery capacity is 103%. Step 3.8.8 of the procedure contains a pen and ink change which is not listed on its attached procedure change form. This discrepancy was discussed with the licensee.

- ORT 3 (Revision 22) Safety Injection Actuation with Loss of Engineered Safeguards AC Unit 2

  This procedure involved extensive coordination by the licensee to ensure all necessary data was obtained. Though there were in excess of 20 people in the control room involved in the test, the background noise was subdued and there were no extraneous distractions. The duty shift supervisor monitored the overall test process, leaving the step by step coordination to an auxiliary shift supervisor. This allowed the duty shift supervisor to continue monitoring the remainder of the plant and not become inundated in the myriad details of the test procedure. The inspector noted one of the less experienced operators underexcite a diesel generator during paralleling operations leading to a slight motorizing of the generator. With the exception of the November 15 inadvertent protective system actuation discussed in Paragraph 3.e, the remainder of the test went well.

- IT-295B (Revision 0) — Overspeed Test, Turbine Driven Auxiliary Feedwater Pump

- IT-1015 (Revision 3) — 10 Year Hydrostatic Test of the Reactor Coolant and Safety Injection System

- RMP-31 (Revision 2) — Measurement of Air Flow Rates for Containment Accident Fans

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.

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.

The licensee made a one hour ENS notification to the NRC on October 27 regarding a degradation of a security door. Immediate compensatory measures were implemented and an investigation initiated. This item has been referred to regional security personnel for evaluation.

The licensee also made a one hour ENS notification to the NRC on October 27 regarding an unauthorized entry into the facility through a vehicle gate. Immediate compensatory measures were implemented and an investigation initiated. This item has also been referred to regional security personnel for evaluation.

8. Engineering and Technical Support (71707) (37828)

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. Installation and Testing of Modifications (37828)

The inspector observed onsite activities and hardware associated with the installation of selected plant modifications to ascertain that modification activities are in conformance with requirements. This inspection included verification of the following items:

- Verification by direct observation that work is being performed by qualified workers and in accordance with approved procedures.
- Verification that the installation conforms to the as-built drawings.
- Confirmation that the equipment and material being used is correct.
- Determination whether the modified equipment was properly prepared for preoperational testing.
- Verification that preoperational testing was conducted using properly reviewed procedures and the test results appropriately evaluated against established criteria.
- Verification that test performance records received an independent QA audit.

Selected portions of the following modification were reviewed:

- Mod 89-156 'Emergency Diesel Generator Stack Support'

No unacceptable conditions were identified.

b. Steam Generator Tube Plug Repairs (71707)

In response to NRC and vendor concern regarding certain Inconel 600 steam generator tube plugs that may be susceptible to Primary Water Stress Corrosion Cracking (PWSCC), the licensee made repairs to 188 Unit 2 steam generator hot leg tube plugs. This repair required the insertion of an addition plug into the existing susceptible plug and was performed robotically by the vendor. The remaining susceptible plugs in Unit 1 steam generators hot and cold legs and the Unit 2 steam generators cold legs will be completed during upcoming outages. The inspector will continue to monitor the progress of the repairs.

c. Modification of the Control Rod Drive Mechanism (CRDM) and Individual Rod Position Indication (IRPI) Cables (71707)

The licensee has experienced degradation in the portions of CRDM and IRPI cables which lay across the vessel head. This degradation is the result of aging of the cables and insulation breakdown caused by the continual heat load on the cables during reactor operation. To correct this problem the licensee installed hermetically sealed, stainless steel shrouded, jumpers from each CRDM and IRPI to terminal plates attached to the outside circumference of the upper head area. The licensee also replaced several feet of each CRDM and

IRPI cable. The terminal plates feature quick disconnect plugs that facilitate the connection of the newly installed cables. This modification is intended to improve the reliability of the IRPIs and CRDMs and reduce the effects of cable degradation. This will also reduce the time required to strip the cables from the CRDMs and IRPIs which is necessary prior to head removal.

d. <u>Reuse of Flawed Control Rod Cluster (71707)</u>

After completion of Unit 2 refueling activities the licensee discovered that control rod R71 had been mistakenly returned to use in the core. One of the rodlets in this cluster has a small two inch longitudinal crack near its tip which was discovered in 1983. The crack was evaluated by the vendor, and after this evaluation the licensee decided not to reuse the control rod in 1983 based upon a vendor recommendation. Because the control rod had been reloaded into the core during this outage (#15), the licensee contacted the vendor again regarding the reuse of the rod and requested further evaluation. This evaluation was completed and the results communicated to the licensee on November 20, 1989. The vendor stated that the use of this control rod for one more fuel cycle would be acceptable. The licensee has issued a justification for continued operation based on the vendors most recent evaluation and satisfactorily performed hot and cold rod drop testing on this control rod. The inspectors will continue to monitor the use of this control rod throughout the fuel cycle to ensure that it continues to operate in an expected manner.

e. <u>Inadequate Design of Residual Heat Removal (RHR) Relief Valve</u>

In the early 1970's the licensee experienced problems associated with the starting of a Reactor Coolant Pump (RCP) while on RHR recirc. The starting of the RCP resulted in a momentary spike in Reactor Coolant System (RCS) pressure resulting in the overpressurization of the RHR system. To prevent this from occurring in the future, the licensee had relief valves installed on the RHR suction line from the RCS hot leg. The relief valves (1RH-861C and 2RH-861C) were installed in 1972 by Modification M-46.

While performing a routine inspection of the RHR system during the Unit 2 refueling outage (Fall 1989), licensee personnel noticed that the flange on the suction side of the relief valve was rated at 300# instead of 600# as required for Class 601 piping. After further review by the licensee, it was determined that a 300# rated flange made of F316 steel could be use in this application. The licensee discovered however, that the flange in question was made of F304 steel and therefore was not acceptable for use in a class 601 system. The further review also identified a failure by the licensee to perform adequate radiography on the flange as required by the FSAR.

In light of these findings the licensee initiated a Modification Request (MR) 89-179 to replace the flange with a 300# rated flange of F316 steel. This work has been completed and the proper radiography was also done.

Based on the findings in Unit 2 the licensee inspected the same relief valve in Unit 1 and discovered that the same conditions existed. The licensee also discovered that the piping on the discharge side of the relief valve was not seismically qualified (a condition that does not exist on the Unit 2 relief valve.) The lack of seismically qualified discharge piping presented the potential for a failure of the relief valve line and a possible non-isolable rupture of the three inch suction line. To immediately remove the possibility of the line rupture, the licensee initiated MR 89-191 to cutaway the relief valve discharge line.

MR 89-191 will also include the replacement of the suction flange, however, this action will be deferred until the Spring 1990 Unit 1 refueling outage. The licensee has performed a 50.59 evaluation of the Unit 1 relief valve and has determined that the existing relief valve suction side flange is within acceptable limits to permit continued operation of Unit 1. The licensee also evaluated the area inside containment that would be effected by any release of steam/water if the relief valve lifted and has determined that no damage or degradation will occur.

The resident inspectors will continue to monitor the modification of the RHR relief valve and will perform a review of the 50.59, MR 89-179 and MR 89-191.

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. 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:

*LER 266/88-001:	Single Failure Potential in 4160 Volt Safeguards Switchgear.
*LER 266/88-003:	Surveillance of 4160 V Undervoltage Relay Not in Accordance With Technical Specifications.
LER 266/88-003-01:	Overpressure Mitigation System Testing Not in Accordance With Technical Specifications.
LER 266/88-003-02:	4160 Volt Undervoltage Relay Not in Accordance With Technical Specifications.
*LER 266/301/88-006:	Suspected Inoperability of Containment Isolation Valve.
*LER 266/88-007:	Safety Injection Block Switch Design Deficiency.
LER 266/88-010-01:	Electrical System Misalignment.
LER 301/88-001:	Reactor Trip Due to Malfunction of Instrument Bus Power Supply Mechanical Interlock.
LER 301/88-002:	Source Range Nuclear Instrument Reactor Trip.
LER 266/89-002:	Containment Isolation Valve Leakage In Excess of Technical Specification Limits.
LER 266/89-003:	Blowdown Sample Isolation Valve Failure To Close On High Radiation Signal.
LER 266/89-005:	Low Temperature Overpressure Protection System Nitrogen Operation Design Inadequacy.
*LER 301/89-001:	Automatic Turbine Runback Caused By Instrument Bus Perturbation.
LER 301/89-003:	Safety Injection Accumulator Level Detector Instrument Failure.
LER 301/89-004:	Transformer Sudden Pressure Signal Results in Reactor Trip.
LER 301/89-005:	Intermediate Range High Flux Trip Signal.
*LER 301/89-006:	Degradation of Steam Generator Tubes.

b. 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.

c. Failure to Meet Commitments to NRC (71707)

During the review of LERs and a previous violation, the inspector noted that several commitments made by the licensee to the NRC had not been completed even though the commitment date had past. This issue was discussed with licensee corporate and plant personnel and it was determined that the problem appears to be one of communicating the completion of, or reevaluation of, commitments from the licensee to the NRC. The following are descriptions of the commitments which were missed;

- LER 266/88010-01 "Electrical System Misalignment"

  The LER's corrective action 3.c stated that administrative procedure PBNP 4.13, "Equipment Isolation Procedure," and its associated forms is being reviewed to consider inclusion of separate approvals for the tag sheet preparation (including verification of operability requirements) and installation authorization. The appropriate changes were to be made by September 1, 1989. This action does not appear to have been fully completed in that the associated forms were not reviewed.

- LER 301/88002-00 "Source Range Nuclear Instrumentation Reactor Trip"

  The LER's corrective action included the revision of procedure OP-3B, "Reactor Shutdown," by April 1989, to clarify the expected responses of the instrument and associated annunciators and also to clarify the applicable Technical Specification operability limits. Though the procedure was revised prior to April 1989, the intended revisions mentioned in the LER did not occur.

The licensee stated that they were in the process of reviewing their commitment tracking system and were intent on improving communication in this area. Due to the level of interest generated by these findings, the inspectors feel that the situation will be corrected.

d. Off Site Review Committee Meeting (40500)

The inspector observed session 42 of the Off Site Review Committee on November 30. A quorum of the committee members were present as well as the Chairman and the President of Wisconsin Electric and the Vice President of the Nuclear Department. Information exchange was considered open and candid and the inspector had no concerns.

e. Meetings with Corporate Staff and Tour of Corporate Headquarters

The inspector met with licensee corporate personnel and were given a tour of the corporate headquarters on November 29, 1989. This was done in an effort to gain a better understanding of the licensee's corporate structure and facilities and to aid in the effort to improve communication between the corporate staff and the NRC.

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

11. Outstanding Items (92702)

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. Unresolved items disclosed during the inspection are discussed in Paragraphs 3.e and 4.a.

12. Management Meetings (30702)

A meeting was held between NRC Region III management and plant management on October 30, 1989, to discuss items of interest and foster improved communications between the licensee and the NRC.

A meeting was held between NRC Region III and plant personnel on November 20, 1989, to discuss the electrical distribution issues leading to the need for discretionary enforcement discussed in Paragraph 3.f.

13. Exit Interview (30703)

A verbal summary of preliminary findings was provided to the licensee representatives denoted in Section 1 on December 1, 1989, 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.