U.S. NUCLEAR REGULATORY COMMISSION

REGION III

REPORT NO. 50-266/301-95008

FACILITY

Point Beach Nuclear Plant Units 1 and 2

License No. DPR-24; DPR-27

LICENSEE

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

DATES

June 17, 1995, Through August 11, 1995

INSPECTORS

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APPROVED BY

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9/14/95
Date

AREAS INSPECTED

A routine, unannounced inspection of operations, engineering, maintenance, and plant support and a special security inspection were performed. Safety assessment and quality verification activities were routinely evaluated. Follow-up inspection was performed for non-routine events. Special inspections of the licensee's independent spent fuel storage installation (ISFSI) continued during this period.

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RESULTS

Assessment of Performance

Performance within the area of OPERATIONS was very good this inspection period. The operators were challenged twice by off normal events. In both instances the shift supervision and crew responded well and had strong communications with the engineering staff (see Section 1.1).

Overall performance within the area of MAINTENANCE was very good this inspection period. However, the inspectors did note instances of declining material condition that may warrant management attention (see Section 2.0).

Overall performance within the area of ENGINEERING was good this inspection period for routine site engineering activities. In particular, very good communications and teamwork were evident between operations and engineering during the repairs to the Unit 1 turbine generator excitor and the Unit 1 main feed pump (see Section 3.0). However, several weaknesses were noted in the licensee's implementation of a program to perform 10 CFR 72.48 safety evaluations of design changes to the ISFSI (see Section 3.1). Nonconformance reports were not properly reviewed for unreviewed safety questions in accordance with licensee procedures. In addition, some engineering design changes did not receive a full safety evaluation as required by 10 CFR 72.48 or did not receive an adequate second engineering review as required by licensee procedures. Since these concerns were identified by the inspectors there has been significant management involvement to ensure that all VSC-24 design changes received adequate safety evaluations, if required, prior to loading a VSC-24.

Performance within the area of PLANT SUPPORT was acceptable this inspection period. Improvements were noted in housekeeping in contaminated areas. A security inspection showed performance was good and the required level of security protection was being provided to the site (Section 4.2). Security program design and proposed implementation practices of the Independent Spent Fuel Storage Installation were determined to be excellent. However, lack of procedure guidance, weak management support and lack of attention to detail by a security officer resulted in an isolated failure to control personnel access to the protected area. A failure to implement a compensatory measure in a timely manner occurred when alarm station operators failed to recognize the condition of an alarm zone. In addition, previous corrective action for a similar failure to implement a compensatory measure in a timely manner demonstrated that your previous actions were not sufficient to prevent recurrence.

Summary of Open Items

Violations: One identified in Section 4.1.
Unresolved Items: One identified in Section 3.1
Inspector Follow-up Items: not identified in this report
Non-cited Violations: not identified in this report

INSPECTION DETAILS

1.0 OPERATIONS

NRC Inspection Procedure 71707 was used in the performance of an inspection of ongoing plant operations. The findings showed that overall performance was very good during both normal and off-normal operations. Good Communications with engineering during recovery from events were also noted.

1.1 Performance of Operations at Power.

- During this inspection period the operators were challenged twice by off normal events. The inspectors' assessment of both instances was that the shift supervision and crew responded well and had strong communications with the engineering staff.

  - On July 14, 1995, Unit 1 tripped from 100% power due to a loss of excitation to the turbine generator (see Section 3.3). The inspectors observed the operating crew's recovery from the trip. Good command and control were displayed by the duty shift supervisor (DSS) and the duty operations supervisor (DOS). Recovery was normal, all equipment responded as required.

  - On July 31, 1995, the DSS initiated a rapid power reduction of Unit 1 in accordance with AOP-17A. During the afternoon of July 31, the Unit 1 control operator received indications of a high temperature on the outboard motor bearing of main feed pump 1P-28B. The DSS notified engineering who monitored the temperature throughout the day. The motor driven lube oil pump was placed on-line and initially the bearing temperature decreased. However, at approximately 5:30 p.m., the bearing temperature began to increase again. At that time the DSS directed the control operator to start the power rampdown to avoid a reactor trip should 1P-28B trip on high bearing temperature. The unit was reduced to 55 percent power and 1P-28B was secured for repairs. The inspectors noted strong command and control by the crew who prepared early in the afternoon when the bearing temperature problem was noted, in anticipation of the need to conduct a rapid power reduction. This resulted in the rampdown being performed efficiently and with no problems.

- The inspectors randomly observed shift turnovers, surveillance activities, operator rounds and other routine control room activities. No concerns were identified this inspection period.

1.2 Verification of G-04 Emergency Diesel Generator (EDG) readiness

The inspector independently verified, in accordance with inspection procedure 71707, that the G-04 EDG was at a good level of readiness to perform its safety function. The G04 EDG is an engineered safety feature (ESF) and was in good material condition with less oil leakage than was noted during previous inspection periods. The system lineup checklist CL-11A, "G-04 Diesel Generator Checklist" matched the G04's as-built configuration and the checklist matched the G04 drawings except for some minor discrepancies. The inspector informed licensee personnel of the minor drawing discrepancies and they identified that the G04 EDG starting air system flow diagram (UE&C 6704-E-222004) discrepancies also applied to the G03 EDG starting air system flow diagram (UE&C 6704-E-222003). Licensee personnel determined that the root cause of the discrepancies was that a portion of an engineering change request (ECR) that would have changed both flow diagrams had not been implemented. An Instrumentation and Calibration engineer determined that this was an isolated occurrence, assured the inspector that walkdowns by training and operations personnel ensured that other drawings did not have discrepancies, and stated that when all four of the EDGs became operable about October, 1995, he would perform an as-built walkdown verification of all the EDG drawings. Licensee personnel initiated ECR 95-288 to correct the drawing discrepancies.

2.0 MAINTENANCE

NRC Inspection Procedures 62703 and 61726 were used to perform an inspection of maintenance and testing activities. Material condition of the plant remains acceptable; however, several items need attention. These include, but are not limited to, significant boric acid build up on the boric acid transfer pumps and insulation on CV-110C which still has not been replaced even though work on the valves was completed several month ago. The inspectors feel that additional management attention is warranted to restore the overall material condition of the plant.

The inspectors observed various portions of the overhaul of emergency diesel generator G02 and repair of main feed pump 1P-28B. Also observed were the monthly surveillance of 4160 kV and 480 kV degraded grid voltage. Overall performance of these maintenance and surveillance activities was very good.

3.0 ENGINEERING

NRC Inspection Procedures 37001, 37551 and 60846 were used to perform an onsite inspection of the engineering functions. Overall, engineering was considered good. In particular, good communications and teamwork was evident between operations and engineering during the repairs to the Unit 1 turbine generator excitor and a modification to the to 1P-28B, main feed pump following motor bearing failure. However, deficiencies were noted regarding the licensee's implementation of a program to

perform safety evaluations of ISFSI changes in accordance with 10 CFR 72.48.

3.1 Independent Spent Fuel Storage Installation (ISFSI)

The inspectors observed various portions of the licensee's planning, preparation, and analysis for loading spent fuel into a ventilated storage cask (VSC-24). Overall inspector assessment of the these activities is that the licensee is proceeding in a cautious and deliberate manner. However, the inspectors did have concerns with the licensee's program for performing safety evaluations in accordance with 10 CFR 72.48.

ISFSI Radiation Protection Program

The inspectors reviewed the licensee's evaluation of the radiation protection (RP) program for operation of the ISFSI and which concluded that no modifications to the existing program were necessary. The inspectors found no problems with this evaluation. The inspectors also reviewed several RP procedures developed to support dry cask activities, and found them to be well written, to incorporate limits specified in the Certificate of Compliance (C of C) and the Safety Analysis Report (SAR), and to incorporate good health physics and As Low As Reasonably Achievable (ALARA) practices. Overall, the inspectors determined that ISFSI operations would not decrease RP program effectiveness.

The inspectors reviewed the dose rate calculations relating to ISFSI operation. Postulated annual dose to the nearest members of the public for farming and hypothetical residence were calculated and determined to be 0.01 mrem and 0.96 mrem respectively. These calculations were based on radiation levels from an ISFSI containing 48 storage casks. These radiation levels were well below the requirements stated in the C of C and the SAR. Environmental monitoring for long term surveillance of the facility included weekly radiation and contamination surveys performed by RP personnel and thermoluminescent dosimeter (TLD) monitoring provided by the state of Wisconsin and by the licensee. State TLDs were already in place at the ISFSI at the time of this inspection. The licensee will add their TLDs at the pre-determined ISFSI location prior to fuel loading. The inspectors walked down the pad and verified locations of TLDs already in place.

The inspectors reviewed the licensee's training program enhancements developed to address ISFSI operation. The phase one and phase two training for RP personnel included a system overview, specific training for RP interaction with loading and unloading, and an overview of RP procedures. The inspectors assessed the initial training efforts in the area of radiation protection to be effective, and noted that specific task oriented training would be provided prior to actual fuel loading during phase three training (the dry run).

Multi-Assembly Sealed Basket Transfer Cask (MTC) and MTC Lifting Yoke Load Test

The inspectors performed a technical review of 94033-LTP, "Load Test Procedure" using ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More"; WEP-04.109.002.1, "System Component Weight Calculations for the WEPCo VSC-24"; PB-588, "Point Beach Nuclear Plant MSB Transfer Cask (MTC)"; and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" as references. The inspector verified that the load test calculation values were accurate prior to performance of the load test.

The inspector observed the load test of the MTC and the MTC lifting yoke. The inspector verified that all test instrumentation was within its current calibration cycle and that calibration of instrumentation and hydraulic test cylinders was properly certified. The inspector verified that a test pressure equivalent to at least 300 percent of the maximum service load was obtained and held for more than ten minutes. During load testing of the MTC doors, dial indicators were placed on each rail at opposite ends to measure rail movement during testing. The dial indicators showed downward deflection of 0.002" at maximum load and returned to zero after the load was removed. The inspector reviewed the NDE qualification records for the Hi Tech Manufacturing NDE inspector and certifications for the LPT penetrant, cleaner/remover, and developer. The inspector observed performance of the post load test NDE inspections of the MTC lifting yoke and visually inspected the MTC. The inspector reviewed all test and NDE data for accuracy and completeness.

All testing and NDE examinations observed were conducted in accordance with approved procedures and the inspectors had no concerns.

Procedure Review and Observations of Dry Runs

The inspectors observed portions of the dry runs to load and unload a VSC-24 cask. Specific tasks observed were:

- Attaching, lifting and moving a VSC with the transporter
- Initial fit-up of the shield lid and structural lid to the MSB
- Welding of the shield and structural lids to the mock-up MSB
- Cutting the shield weld on the mock-up MSB
- Qualification of a high temperature liquid penetrant examination procedure.

One concern was noted by the inspectors during the dry runs. The roadway has been modified in two locations to allow the transporter turn more easily. The modification consisted removing the blacktop pavement and replacing it with gravel. However, it is not clear which areas have been analyzed to support the transporter and which areas need to be avoided. The inspectors discussed this issue with the licensee. The licensee is currently reviewing what actions need to be taken to ensure the transporter remains on the analyzed portion of the road. In

addition, the licensee has written a condition report to address their concerns with road degradation during transporter moves.

The inspectors performed a cursory review of the licensee's analysis to reflood the MSB in the event that a cask needed to be unloaded. No concerns were identified but a more in depth evaluation is continuing by the inspectors.

During the dry runs observed, the licensee did a good job of following procedures and documenting necessary changes using the temporary change process. This will ensure that all changes identified during the dry runs will be incorporated into the procedures and used during the actual load process. The inspectors did not identify any concerns with the dry runs observed and will continue to monitor dry runs as they take place.

Safety Evaluations in accordance with 10 CFR 72.48

The inspectors randomly reviewed licensee Engineering Change Requests (ECRs) and Nonconformance Reports (NRs), to ascertain if they had been properly screened for an unreviewed safety question per NP 10.3.1, "Authorization of Changes, Tests and Experiments (10 CFR 50.59 and 72.48 Reviews)" and if necessary, a safety evaluation (SE) was performed in accordance with 10 CFR 72.48. The inspectors had concerns that the licensee did not properly screen ECRs and NRs in accordance with NP 10.3.1, and that in some cases, a safety evaluation in accordance with 10 CFR 72.48, was not performed as required. Specific inspector concerns are as follows:

- Several of the NRs made one-time, dimensional changes to the MSB and received the approval of the vendor, Seirra Nuclear; however, none were screened in accordance with Step 3.1.6, of NP 10.3.1 to evaluate if any of the nonconformances affected licensing bases documents, and required an evaluation in accordance with 10 CFR 72.48, for an unreviewed safety question. Examples are NR 95104-004, -005 and -008.

- ECRs were screened by the licensee in accordance with NP 10.3.1. The licensee determined that full evaluations per 10 CFR 72.48, were not required because they did not effect licensing basis documents. However, in some cases, the ECR did change components described in the Safety Analysis Report (SAR) without a thorough evaluation of the change in accordance with Step 3.3, of NP 10.3.1. Examples are ECR 95-0065 which modified the barrel to bottom plate fit-up and weld of the MSB. The screening did not contain enough information to determine if an unreviewed safety question existed from the modification. The same is true for the screening of ECR 95-0042, which modified the MSB vent hole size.

- Many of the ECRs reference SNC letter 95-133 for the basis on why a complete SE did not need to be performed. However, that letter was a draft document. When the final basis was sent via SNC letter 95-155, a thorough review was not performed to ensure no

bases were changed. All ECRs received a final review after SNC letter 95-155 was issued but none referenced it. This is an example of an inadequate review in accordance with Step 3.4.1, of NP 10.3.1. This is evidenced in ECR 95-0175 which stated that a 150 percent load test of the MTC trunnions and doors was acceptable and referenced SNC letter 95-133. However, SNC letter 95-133, did not mention ECR 95-0175. In addition, SNC letter 95-155, stated that the licensing intent for the MTC was to perform a 300 percent load test and meet the requirements of ANSI N14.6 for lifting critical loads. The letter went on to say that since the licensee is using a single-failure proof crane and does not consider consequences of a load drop, the MTC lifting components should be tested to
300 percent per Section 7.3.1 of ANSI N14.6.

Based on the above findings, the licensee did not have a strong program in place to properly review and analyze potential safety issues in accordance with 10 CFR 72.48.

Since these concerns were identified by the inspectors there has been significant licensee management involvement to ensure that all VSC-24 design changes received adequate safety evaluations, if required, prior to loading a VSC-24. The licensee is reviewing all NRs and ECRs in accordance with NP 10.3.1. In addition, the licensee has stated they will review all changes made to the SAR since its issue, to ensure no unreviewed safety questions exist, prior to placing the VSC-24 system in service.

Once the licensee's review and evaluations are complete the inspectors will follow up to see what, if any, significant changes were made to the original evaluations and if any unreviewed safety questions were identified. In addition, the inspectors will assess the corrective actions taken to by the licensee to ensure adequate screenings and SE are performed in the future. The inspector follow-up will be tracked as an unresolved item (URI 266/301-95008-01, (DRP)).

3.2 <u>Containment Accident Fan Coolers Fail to Achieve Required Service Water Flow Rate and High Lake Water Temperature Observed</u>

During the inspection period, containment accident fan coolers in both Units 1 and 2 obtained only 1170 gpm service water flow while being tested per TS-33 and TS-34. The minimum acceptance criteria at the time was 1200 gpm. The service water flow model was reanalyzed and a new minimum flow rate of 1070 gpm will be used during TS-33 and TS-34.

Previously, the design basis service water temperature was 75°F. In late June, the lake water inlet temperature reached 74°F. Recently completed calculations N-94-066 and N-94-067 allowed a lake water inlet temperature of 77°F with two service water pumps per train operable, if two service water outboard valves are open and both spent fuel pool heat exchangers are isolated during Post-LOCA Recirculation phase. Two service water pumps are required to be operable per TS 15.3.3.D and the

a LOCA so that the spent fuel pool heat exchangers are isolated. JCO 95-05-01 was written and included in the DCS handbook on June 27 to require two service water overboard valves to be open when the lake water inlet temperature reaches 70°F. Calculations have also been completed which allow a lake water inlet temperature of 83°F when all three service water pumps per train are operable, the spent fuel pool heat exchanger is lined up to Train A, and two service water outboard valves are open. The inspectors are continuing to review of all calculations and administrative controls regarding high lake water inlet temperature. This issue is being tracked by IFI 50-266/301/93012-01(DRS) and remains open at this time.

3.3 Loss of Unit 1 Turbine Generator Excitation

On July 14, 1995, a turbine trip/reactor trip occurred on Unit 1 due to electrical generator problems related to hot and humid weather combined with low lake temperatures. The lake provides the intake for the service water system which in turn supplies the heat sink for the turbine generator excitor air coolers. The trip occurred when condensation from the coolers was blown around inside the excitor house resulting in several blown fuses and diodes on the excitor. This was followed by a loss of excitation and a unit lockout. The subsequent turbine trip then initiated a reactor trip.

As noted in Section 1.1, the operations crew responded well to the trip. However, the inspectors were concerned with the events that led to condensation in the excitor house. Specifically, the service water temperature to the excitor air cooler was not maintained in accordance with vendor recommendations. In the past, the temperature was properly controlled; however, within the last few years, the throttle valves were left in the fully open position to allow maximum cooling. This was done primarily in an effort to prolong the life of the excitor as cooler temperatures slow degradation of electrical equipment.

Since the reactor trip occurred, the licensee has formed the "Main Turbine Generator Reliability Team" to improve the reliability of the generators and support equipment for both units. The team will not only review the events that led to the recent trip but other previous problems such as the EHC oil leak that ultimately led to a Unit 2 reactor trip (see IR 266/301-95003).

The inspectors believe this to be adequate follow-up to this event and have no other concerns with this reactor trip.

3.4 Licensee Action on Previously Identified Items

(Closed) I.E. Bulletin 79-14 (266/79014-BB; 301/79014-BB): Seismic Analysis for As-Built Safety Related Piping Systems.

Results: The licensee's I. E. Bulletin (IEB) 79-14 re-evaluation program was comprehensive and well organized. Although some discrepancies were identified by the NRC, these were very minor or were

discrepancies were identified by the NRC, these were very minor or were determined to be isolated cases. The overall effort was considered to be very good.

Background: Because multiple errors were identified by the NRC in existing pipe stress analyses, the licensee committed to a complete re-evaluation of the I. E. Bulletin 79-14 program, in their letter dated September 5, 1990. Since the identified errors were attributed, in part, to the accelerated schedule imposed by the bulletin, the licensee proposed a multi-year effort to complete this re-evaluation. Currently, the re-evaluation program has completed 92 percent of the subsystem walkdowns and 75 percent of the subsystem evaluations with almost all of the remaining scope outside of the containment structures. Depending on the extent of the discrepancies identified in the original analyses, subsystems were either reconciled using simple calculational approaches or pipe stress analyses were completely rerun. Approximately 65 percent of the subsystems required the more extensive re-analysis and of the approximately 1600 pipe supports evaluated to date, about 8 percent required modifications to resolve identified problems.

Details: A regional inspector reviewed a limited sample of reconciliation, re-analysis, and pipe support modification packages to verify the accuracy and adequacy of the re-evaluation program. Although no problems were identified in the majority of the reviewed calculations, some minor discrepancies were found by the NRC. In one case, the finite element analysis for a support included a non-existent structural member and incorrectly restrained another member. The re-analysis with the corrected model indicated that these discrepancies did not cause the support to be overstressed. In another case, the stress intensification factor for a socket-welded fitting was incorrectly reduced. Again, the re-analysis demonstrated that this discrepancy did not result in an overstressed condition. Both of these examples indicated some weakness in the verification and checking activities associated with the calculations. However, the lack of consequence, the isolated nature, and the licensee's aggressive response to the issues reduced the significance of the discrepancies. Several calculations reviewed by the NRC did not clearly explain the bases for the analytical methods. Although these methods were eventually justified through additional discussions with the originating engineer, the lack of clear documented bases made it difficult for the NRC inspector and presumably the licensee's verifiers to validate the analyses.

Engineering: The licensee's recent discovery of an original construction deficiency with channel embed installations was a very positive example of identification and resolution of a technical issue. In trying to increase the design capacity of the existing channel embeds, the licensee discovered that the embedment bar configuration was different from the installation drawings. Instead of "L" shaped welded bars, the embedded channels were held-in with welded studs. In order to verify the as-installed configuration, the licensee chipped out several unused channel embeds in the plant and discovered that some embeds used welded studs and some used "L" bars. However, evidently due to

clearance problems, the "L" bars had been bent off to one side and were only embedded two inches, significantly reducing the structural capacity. The licensee's corrective action included the inspection of all channel embeds using an ultrasonic examination technique to identify discrepant embeds. Several embeds were identified that subsequently had to be modified to return the design margin to within Code requirements. The difficulty of identification and the comprehensive resolution was considered very good performance by the licensee.

Based on these reviews, this item is closed.

4.0 PLANT SUPPORT

NRC Inspection Procedures 71750, 81001, 81070, 81064 and 83750 were used to perform an inspection of Plant Support Activities. Overall, performance in this area remained adequate.

4.1 Radiological Protection

Radiological protection performance showed improvements since the last inspection period. Several previously contaminated areas have been decontaminated. Continued emphasis in this area is encouraged to ensure previously noted declines do not recur.

4.2 Special Security Inspection

The purpose of the inspection was to conduct a pre-operational security inspection of the Independent Spent Fuel Storage Installation (ISFSI) and to review two licensee identified and reported security events: (1) a failure to implement a compensatory measure for an alarm zone that occurred on July 13, 1995 and (2) an incident involving unauthorized personnel access to the protected area that occurred on June 18, 1995. The conclusions and findings of the inspection are discussed below.

Independent Spent Fuel Storage Installation

Inspection results concluded the security program can support the Independent Spent Fuel Storage Installation (ISFSI) and provide the level of protection required by the NRC approved ISFSI security plan. Inspection activities included: equipment, procedures, training and personnel and vehicle control. Inspection activities identified six minor security issues that are required to be completed prior to operation of the ISFSI. Those issues are: (1) senior management approval of security procedures; (2) development and implementation of a security "walk down" procedure immediately prior to security implementation; (3) completion of grading to assure IDS performances; (4) resolving a minor lighting problem effecting assessment; (5) complete some minor security training activities and (6) document required LLEA coordination activities. The licensee's site security supervisor stated those issues noted above would be completed prior to storing fuel at the facility. Licensee activities relating to the security of the ISFSI were excellent.

Compensatory Measure

On July 13, 1995, the licensee identified compensatory measures had not been implemented for a period of 23 minutes for a protected area intrusion alarm zone that was not functioning. The failure to implement compensatory measures was in violation of the licensee's security plan. When identified, initial corrective action was implemented and the event was reported to the NRC, as required. A similar event occurred on March 23, 1994. Corrective actions for that event did not prevent recurrence. The actual event, which was short in duration, and fortuitously resulted in minimum impact on plant safety. (VIO 50-266/301-95008-02).

On July 13, 1995, at 1:26 p.m., a protected area alarm zone went into alarm and a response was conducted. The alarm was acknowledged and response was verified by both alarm station (CAS/SAS) operators, but the alarm zone was not reset by the CAS operator as required, nor did the SAS operator recognize this fact. In this condition, the alarm zone would not detect intrusion. At 1:49 p.m., the same day, a security supervisor entered CAS and while reviewing CAS operations recognized the alarm zone had not been reset. Further questioning of the CAS and SAS operators by the supervisor determined compensatory measures had not been implemented. The alarm zone was immediately reset and tested, and licensee security management was notified. A search of the protected area was conducted to ensure no authorized breach occurred. Nothing unusual was found during the search.

Licensee review showed both alarm station operators were alert and fully knowledgeable of alarm station activities. The licensee concluded procedural guidance was adequate and both operators had been adequately trained. The procedure required communication between alarm stations to provide high assurance alarm zones are reset. Licensee review could not verify if communication had occurred. In addition, software to the security computer that had been installed in late 1994 to remind operators to reset accessed alarm zones had been deactivated by the CAS operator. When this software was installed, licensee security management made the decision not to require compliance for this function because the other alarm station was required to monitor the actions of the CAS.

The following initial corrective action was implemented: (1) both alarm station operators were discharged from alarm station duties and (2) all alarm station operators and security supervisors were briefed on the event. Additional actions are being evaluated by the licensee.

In addition, our review identified that on March 23, 1994, alarm station operators failed to implement compensatory measures for a failed protected area alarm zone. This finding resulted in a non-cited violation being documented in our letter, dated September 14, 1994. It appeared corrective actions for that event did not prevent recurrence. We also identified and addressed a decline in the general performance of alarm station operators as a Inspection Followup Item in Inspection

Reports No. 50-266/301-94014, dated August 4, 1994. Alarm station operators did not work as a team to address alarm related activities.

Access Control - Personnel

On June 20, 1995, the licensee discovered that a suspended licensee employee had gained access to the Point Beach protected area on June 18, 1995, by utilizing his security key card which had not been revoked upon his suspension four days prior. The individual had access occurred for 51 minutes. The employee's access had been verbally suspended pending his formal resignation from Wisconsin Electric. The supervisor had determined that the suspended individual had no need for unescorted site access over a weekend. The employee entered the protected area to return company material and to pickup his personal effects.

The licensee attributed the cause of the failure to weak communication by the site security supervisor in not assuring the badge was removed on June 16, 1995, and a security officer's lack of a questioning attitude to assure that the badge had been removed from active status when advised on June 18, 1995. A contributing factor was the lack of procedural guidance providing instructions for suspension or revocation of unescorted access.

The inspector reviewed the event and licensee corrective action both of which were documented in Licensee Event Report 95-S01 submitted by Point Beach Nuclear Plant under Docket No. 50-266 on July 10, 1995. Corrective actions were considered appropriate to preclude recurrence of similar events. The failure to remove the badge was not willful and a similar event had not occurred.

This was a violation of Section 2.4, Paragraph 2.0 of the approved security plan which required individuals who enter the protected area to be authorized access based on need. The employee's badge should have been taken out of the system when his supervisor concluded the individual's need for plant access was not necessary. This licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII of the NRC Enforcement Policy.

5.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (SAQV)

The inspectors attended several Management Supervisory Staff Meetings. Management review of safety issues was very good this inspection period. Particularly good was the team approach to recovery from the Unit 1 reactor trip.

6.0 PERSONS CONTACTED AND MANAGEMENT MEETINGS

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

At the conclusion of the inspection on August 14, 1995, the inspectors met with licensee representatives (denoted by *) and summarized the scope and findings of the inspection activities. The licensee did not identify any of the documents or processes reviewed by the inspectors as proprietary.

G. J. Maxfield, Plant Manager
A. J. Cayia, Production Manager
*F. A. Flentje, Administrative Specialist
*W. B. Fromm, Sr. Project Engineer - Plant Engineering
C. M. Gray, Duty Shift Superintendent
*T. Guay, Health Physics Supervisor
L. D. Halverson, Site Services Manager
F. P. Hennessy, Manager - Chemistry
W. J. Herrman, Sr. Project Engineer - Construction Engineering
N. L. Hoefert, Manager - Production Planning
T. Jessesky, Quality Assurance
J. A. Palmer, Manager - Maintenance
S. A. Patulski, Nuclear Engineering Manager
J. C. Reisenbuechler, Manager - Operations
*D. D. Schoon, Regulatory Services Manager
*J. G. Schweitzer, Maintenance Manager
R. D. Seizert, Training Manager
G. R. Sherwood, Manager - Instrument & Controls
T. G. Staskal, Sr. Project Engineer - Performance Engineering
*M. F. Baumann, Manager - Nuclear Fuels Services
*P. D. Bronk, Project Manager - ISFSI
*K. R. Aundensun, Project Manager - ISFSI