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TABLE OF CONTENTS

EXECUTIVE SUMMARY

Point Beach Nuclear Plant
NRC Inspection Report 50-266/97025(DRS); 50-301/97025(DRS)

This inspection included a review of the licensee's implementation of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The report covers a two-day site visit, a four-day in-office inspection, and a one-week on-site inspection by regional and Office of Nuclear Reactor Regulation inspectors, and a contractor from the Idaho National Engineering & Environmental Laboratory.

The team concluded that the maintenance rule had been properly implemented at Point Beach.

Operations

- Operator knowledge was consistent with their responsibility for implementation of the maintenance rule. There was no indication that the maintenance rule detracted from the operators' ability to safely operate the plant.

Maintenance

- In general, scoping of structures, systems, and components (SSCs) was considered adequate. Rescoping efforts in response to the May 1997, self-assessment findings appropriately placed additional SSCs in the maintenance rule program scope; however, two SSCs were still inappropriately excluded.

- The approach to establishing the safety significance ranking for SSCs within the maintenance rule scope was adequate. The expert panel's safety determinations effectively compensated for the limitations of the probabilistic safety assessment applications. A weakness in the determination process was the use of a probabilistic safety assessment model that did not reflect plant configuration modifications and contained old plant-specific data.

- The procedure for performing periodic assessments met the requirements of the rule and the intent of the Nuclear Management Resource Council implementing guidance. The yearly assessment, issued March 19, 1997, was acceptable.

- The process to balance availability and reliability appeared adequate. Adjustments had been made to preventive maintenance tasks on some systems as a result of these evaluations.

- Processes for assessing plant risk resulting from taking equipment out of service during at-power and shutdown conditions were adequate. Plans for an on-line risk monitor, additional probabilistic safety assessment model insights in the matrix, and a shutdown probabilistic safety assessment model will strengthen the licensee's assessment processes.

- In general, performance criteria were appropriately established to measure system performance. Performance criteria for one system did not adequately measure system performance. Established goals were generally conservative; however, two (a)(1) systems had inappropriate goals. A weakness existed in the process for establishing and revising performance criteria in that new or revised criteria were not formally reviewed by an individual responsible for the maintenance rule program. A number of performance criteria documentation discrepancies were also identified.

- With the exception of inadvertently omitted performance monitoring criteria, the structure monitoring program was well-organized and comprehensive. Overall, with the exception of baseplate-to-structure hanger support gaps, the structures inspected were in good condition. The licensee noted that the hanger baseplate-to-structure inspection had not yet been performed, but was forthcoming.

- The industry operating experience review program was well-organized and properly linked to the maintenance rule program. System engineers were clearly using industry experience information and generally understood the need to incorporate it into the maintenance rule program. However, there was little documentation showing consideration of industry operating experience in maintenance rule activities.

- With the exception of the inappropriate goal for the residual heat removal system and the untimely identification of exceeding the component cooling water criteria, the maintenance rule was properly implemented for the systems the team examined. Classifying a number of systems (a)(1) due to performance concerns, although no performance criteria had been exceeded, was considered a strength.

- In general, the material condition of the systems examined was acceptable. However, the maintenance department and the system engineer were not effectively monitoring battery electrolyte levels.

Quality Assurance

- During implementation of the maintenance rule program, three audits and a Nuclear Energy Institute evaluation provided good observations and findings in several aspects of maintenance rule implementation. The most recent audit (May 1997) of the maintenance rule program was extremely thorough and was considered a strength. Corrective actions sampled by the team were appropriately implemented.

Engineering

- System engineers were experienced and knowledgeable about their systems. System engineers were generally familiar with the maintenance rule program, although some system engineers were not knowledgeable of certain aspects of the program.

Report Details

Summary of Plant Status

Units 1 and 2 were both shut down prior to the inspection and remained in that condition during the inspection.

Introduction

This inspection included a review of the licensee's implementation of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The report covers a two-day site visit, a four-day in-office inspection, and a one-week on-site inspection by regional and Office of Nuclear Reactor Regulation inspectors, and a contractor from the Idaho National Engineering & Environmental Laboratory.

I. Operations

O4 Operator Knowledge and Performance

O4.1 Operator Knowledge of Maintenance Rule

a. Inspection Scope (62706)

The team interviewed one Duty Shift Superintendent and three Control Operators to determine if they understood the general requirements of the maintenance rule and their particular duties and responsibilities for its implementation.

b. Observations and Findings

Operations personnel had a general knowledge of the maintenance rule and their role in its implementation. The operators stated that their duties included recording equipment out of service times and the use of the protected system matrix to evaluate emergent or fill-in maintenance activities. The operators noted that more attention had been focused on accurate log keeping for out of service times due to maintenance rule implementation. Operations personnel were also involved with the initial development of system performance criteria and continued to be involved in concurrence on revised criteria. The operators indicated that the maintenance rule increased plant personnel's awareness of (a)(1) systems. In addition, the operators viewed the maintenance rule as beneficial in focusing management attention on (a)(1) systems to address equipment issues. With the possible exception of the administrative resources devoted to the development of performance criteria, operations personnel did not identify any negative impact upon their duties due to the maintenance rule.

c. Conclusions

Operator knowledge was consistent with their responsibility for implementation of the maintenance rule. There was no indication that the maintenance rule detracted from the operators' ability to safely operate the plant.

II. Maintenance

M1 Conduct of Maintenance (62706)

M1.1 Scope of Structures, Systems, and Components Included Within the Rule

a. Inspection Scope

The team reviewed scoping documentation to determine if the appropriate structures, systems and components (SSCs) were included within the maintenance rule program in accordance with 10 CFR 50.65(b). The team used NRC Inspection Procedure 62706, Nuclear Management Resource Council 93-01, Regulatory Guide 1.160, the Point Beach Final Safety Analysis Report, Emergency Operating Procedures, and other information as references. The team selected an independent sample of SSCs that could have been included within the scope of the rule, but were not. The team used this sample to evaluate scoping decisions.

b. Observations and Findings

The licensee's maintenance rule program and scoping were described in the Nuclear Power Business Unit Administrative Manual AM 3-4 Procedure, "Implementation of the Maintenance Rule at PBNP," Revision 2, dated May 30, 1997, and Procedure Manual NP 7.7.4, "Scope and Risk Significant Determination for the Maintenance rule," Revision 3, dated October 15, 1997. The procedures described the methodology used to identify (scope) SSCs for inclusion in the maintenance rule program. The processes and responsibilities associated with the program and scoping were clear and well-defined. The methodology used was consistent with the guidance of Nuclear Management Resource Council 93-01.

The team also reviewed the licensee's computer database (MRLIN) for those systems that were included or excluded from the maintenance rule. This database identified systems which were high safety significance, had safety related functions, had non-safety related components which were used to mitigate an accident or transient, and those systems which had non-safety related components used in the emergency operating procedures, or which could cause a trip or safety system actuation.

The team's review of scoping procedures, maintenance rule applicable structural monitoring procedures, and the MRLIN database, revealed numerous documentation errors. Examples were:

- A comparison of the scoping information contained in Appendix A of the Scoping and Risk Procedure (NP 7.7.4), and the maintenance rule database revealed many discrepancies between the two. For example, the Diesel Starting Air and Computer Room Heating, Ventilation, and Air Conditioning systems were considered high safety significance on one list but not on the other. The Reactor Coolant, 125VDC Electrical, and Safety Injection systems were considered to have non-safety components, used in emergency operating procedures or which could cause a trip or safety system actuation, on one list but not the other.

- A comparison of the list of systems within the scope of the maintenance rule (Procedure NP 7.7.4 Appendix A) and the licensee's structure monitoring program instruction Procedure NDE-751 "Structural Surveillance of Containments and General Structures," Revision 5, dated October 31, 1997, revealed that although the Unit 1 and Unit 2 containment facades were considered within the scope of the rule and were being monitored by the structures program, these structures were not contained in the list of in-scope systems (nor in MRLIN).

The team reviewed SSCs which were not within the scope of the maintenance rule to verify the appropriateness of their exclusion. The team noted that the Facade Freeze Protection System which supported refueling water storage tank water level instrumentation used in emergency operating procedures during a safety injection event was noted included in the scope of the program. The also found that the Switchyard Control Building which housed electrical distribution equipment, was within the first inter-tie with the off-site distribution system, and provided support for offsite power sources, was not within the scope of the program. On review, the team noted that these systems had a potential impact on the plant or the safety-related systems with which they interacted. The licensee's Scoping Basis and Justification for these items did not adequately describe why they were not within the scope of the rule. The team discussed the need to provide acceptable documentation for the basis for maintenance rule scoping decisions with licensee personnel. Failure to include the Facade Freeze Protection system and the Switchyard Control Building with the scope of the maintenance rule program was considered a violation of 10 CFR 50.65(b) (VIO 50-266/97025-01(DRS): 50-301/97025-01(DRS)).

c. Conclusions

In general, scoping of SSCs was considered adequate. Rescoping efforts in response to May 1997, self-assessment findings appropriately placed additional SSCs in the maintenance rule program scope; however, two SSCs were still inappropriately excluded.

The team noted a general weakness in the area of scoping documentation. The licensee did not thoroughly document why some SSCs were excluded from the maintenance rule scope, and the maintenance rule scoping data base and scoping procedures had numerous discrepancies.

M1.2 Safety (Risk) Determination, Risk Ranking, and Expert Panel

a. Inspection Scope

Paragraph (a)(1) of the maintenance rule required that goals be commensurate with safety. Implementation of the rule using the guidance contained in Nuclear Management Resource Council 93-01 also required that safety be taken into account when setting performance criteria and monitoring under (a)(2) of the rule. This safety consideration would then be used to determine if the SSCs should be monitored at the plant, system, or train level. The team reviewed the methods that the licensee established for making these required safety determinations. The team also reviewed the safety determinations that were made for the systems that were examined in detail during this inspection.

b.1 Observations and Findings on the Expert Panel

The licensee separated the responsibilities of maintenance rule implementation between two groups. An expert panel determined high or low safety significance of SSCs; expert panel composition included personnel from maintenance engineering, system engineering, probabilistic safety assessment group, scheduling, and operations. The panel used their experience in conjunction with the probabilistic safety assessment to assess SSC safety significance. The maintenance rule committee's responsibilities included review and approval of scoping decisions, goal-setting action plans, performance criteria selection, and the reclassification of SSCs from (a)(2) to (a)(1) and (a)(1) to (a)(2). Meeting minutes described both groups' activities. Expert panel members had received some form of probabilistic safety assessment training and demonstrated an understanding in the use of probabilistic safety assessment. However, most of the expert panel and maintenance rule committee members had not received formal training in probabilistic safety assessment and some were not familiar with Nuclear Management Resource Council 93-01. In the past, the expert panel and the maintenance rule committee met on an infrequent basis, generally whenever maintenance rule issues were discussed or evaluated. Further, expert panel and maintenance rule committee composition was not always made up of a consistent group and some of those in attendance at meetings were members for only short periods of time. The expert panel and the maintenance rule committee members appeared knowledgeable concerning the requirements of the maintenance rule and understood their responsibilities.

The licensee recently updated procedure AM 3-4, "Implementation of the Maintenance Rule at PBNP," Revision 3, and established a charter that delineated the responsibilities of a standing maintenance rule overview expert panel. The charter, which had not been formally approved at the time of the exit, required as a minimum, quarterly meetings. Responsibilities included approving scope changes, reviewing performance criteria, and review of action plans.

On November 20, 1997, the licensee convened an expert panel meeting to discuss whether the facade freeze protection system should be within the scope of the

maintenance rule. This issue had been raised by a quality assurance audit and by a maintenance rule self-assessment team. The inspector observed that the meeting was conducted well with a good exchange of information among the panel members, the maintenance rule coordinator, and the system engineer. The panel decided that the facade freeze protection was within the scope of the maintenance rule. The panel also decided that the rest of the system should be evaluated to determine whether other SSCs should be included in scope.

c.1 Conclusions on the Expert Panel

The expert panel consisted of qualified, experienced personnel. The members used their experience in conjunction with the probabilistic safety assessment to properly assess SSC risk significance. The previous expert panel and the maintenance rule committee composition were not consistently staffed by the same personnel. However, the recent development of the charter for the reestablished maintenance rule overview expert panel was expected to provide a more consistent panel that would attend future maintenance rule meetings.

b.2 Observations and Findings on Risk Determinations

b.2.1 Analytical Risk Determining Methodology

The licensee's process for establishing the risk significance of SSCs within the scope of the maintenance rule was documented in Point Beach procedure NP 7.7.4, "Scope and Risk Significant Determination for the Maintenance Rule," Revision 3, October 15, 1997. This document was reviewed and found to adequately describe the process of determining risk significance.

The licensee used plant-specific probabilistic safety assessment studies to rank SSCs with regard to their risk significance. These studies included the Point Beach Nuclear Plant Individual Plant Examination and an updated probabilistic safety assessment (known as the PSA93 model). The PSA93 model was a small event tree and large fault tree model; the NUPRA computer code was used to develop and quantify the model.

For the risk ranking process, the licensee used a 1.0E-11 truncation level. This truncation level was almost seven orders of magnitude less than the overall core damage frequency of 9.38E-5 per reactor year. The truncation level used for the risk significance determination process was considered reasonable.

The general quality and level of detail of the plant-specific probabilistic safety assessment was not reviewed. However, the PSA93 model did not reflect either plant modifications since 1993 or updated operating experience data since September 1990. Not having updated the probabilistic safety assessment model was considered by the team to be a weakness in maintenance rule implementation. At the time of the inspection, the licensee had demonstrated that an updated probabilistic safety assessment model had been developed and results had been generated, but the results had not been reviewed or verified.

For SSCs modeled in the licensee's probabilistic risk assessment, three importance measures were evaluated (core damage frequency contribution, risk achievement worth, and risk reduction worth), as recommended in Nuclear Management Resource Council 93-01. If a basic event's importance measure met one or more of the criteria, then the SSC associated with that basic event was judged to be maintenance rule risk significant. Because the Nuclear Management Resource Council 93-01 guidance specified that an SSC would probably be considered risk significant if any of the three importance measure criteria were met, the approach used by the licensee was determined by the team to be adequate. The expert panel then made the final determinations with respect to risk significance. One SSC, the Waste Gas system, had been indicated as high safety significance but had been downgraded because the expert panel had determined that the assumptions in the probabilistic safety assessment were not valid.

b.2.2 Adequacy of Expert Panel Evaluations

For SSCs not modeled in the individual plant examination, the expert panel determined the risk significance by a consensus vote. The expert panel evaluated risk significance for systems based on two functions: preventing core damage and maintaining containment integrity. The evaluations of the expert panel were determined by the team members to be adequate.

c.2 Conclusions on Risk Determinations

The team concluded that the licensee's approach to establishing the risk ranking for SSCs within the maintenance rule scope was adequate. The expert panel's risk determinations effectively compensated for the limitations of the probability safety assessment applications. A weakness in the determination process was the use of a probabilistic safety assessment model that did not reflect plant configuration modifications. Another weakness noted by the team was the use of older plant-specific data in the probabilistic safety assessment model.

M1.3 (a)(3) Periodic Evaluations

a. Inspection Scope

Paragraph (a)(3) of the maintenance rule requires that performance and condition monitoring activities, associated goals, and preventive maintenance activities be evaluated, taking into account where practical, industry wide operating experience. This evaluation was required to be performed at least one time during each refueling cycle, not to exceed 24 months between evaluations. The team reviewed both the procedural guidelines for these evaluations and the completed yearly evaluation.

b. Observations and Findings

The licensee's maintenance rule program document provided adequate guidance for preparing periodic assessments which met the requirements of 10 CFR 50.65(a)(3) and the intent of Nuclear Management Resource Council 93-01. The licensee's procedure

NP 7.7.7 "Guideline for Maintenance Rule Periodic Report," Revision 2, required that periodic assessments be performed on a yearly basis. The procedure required that the maintenance rule assessment of activities would include comparing actual performance of systems to the performance criteria and goals, evaluating the effectiveness of corrective actions and reviewing the establishment of goals for (a)(1) systems. The team reviewed the assessment issued on March 19, 1997, which included evaluations of both units. The report provided a summary of what systems were in scope, which were considered high safety significance, a review of initial performance criteria and goals set for the systems and disposition of the systems to (a)(1) or (a)(2) monitoring. The report stated that no systems were moved from (a)(1) monitoring to (a)(2) monitoring because most of the corrective action plans were still being implemented and that there was insufficient information to determine if corrective actions had been effective. The periodic assessment was considered acceptable.

c. Conclusions

The procedure for performing periodic assessments met the requirements of the maintenance rule and the intent of the Nuclear Management Resource Council implementing guidance. The yearly assessment was considered acceptable.

M1.4 (a)(3) Balancing Reliability and Unavailability

a. Inspection Scope

Paragraph (a)(3) of the maintenance rule requires that adjustments be made where necessary to assure that the objective of preventing failures through the performance of preventive maintenance was appropriately balanced against the objective of minimizing unavailability due to monitoring or preventive maintenance. The team reviewed the plans to ensure this evaluation was performed as required by the maintenance rule.

b. Observations and Findings

Balancing reliability and availability consisted of monitoring SSC function performance against the established performance criteria. If the performance criteria were met, then the criteria were considered balanced. The licensee's procedure NP 7.7.7 provided guidelines to balance reliability and availability. The procedure required that each system be reviewed during the periodic assessment to determine if the balance was being maintained. This analysis was performed during the yearly periodic assessment. In reviewing system performance, the licensee had started to balance reliability and availability by making adjustments to preventive maintenance activities for the service water pumps, the 480 Volt breakers and the electro-hydraulic control system.

c. Conclusions

The team concluded that the process to balance availability and reliability appeared adequate.

M1.5 (a)(3) On-line Maintenance Risk Assessments

a. Inspection Scope

Paragraph (a)(3) of the maintenance rule states that the total impact on plant safety should be taken into account before taking equipment out of service for monitoring or preventive maintenance. The team reviewed the licensee's procedures and discussed the process with a Probabilistic Safety Assessment Engineer, Scheduling Technician, Plant Outage Coordinator, and a Shift Outage Coordinator.

b. Observations and Findings

The licensee's process for determining plant safety when equipment was taken out of service was documented in several procedures: NP 10.1.1, "TS Equipment OOS/Voluntary Entry into an LCO," Revision 7; NP 10.2.1, "Outage Planning, Scheduling, and Management," Revision 9; NP 10.2.2, "Non-Outage Work Planning," Revision 1, and NP 10.3.6, "Outage Safety Review and Safety Assessment," Revision 3. These procedures covered plant at-power and shutdown risk evaluations.

The primary risk evaluations for taking equipment out of service were technical specification driven. The licensee had used a "Cross Train Voluntary LCO & Testing Restrictions Matrix," (PBF-9133a, Revision 1) for evaluating plant risk when taking equipment out of service. The licensee had developed the matrix primarily from a technical specification basis, but insights from the probabilistic safety assessment model provided some input to the development of the matrix. Risk evaluations have been performed by the licensee for various configurations contained in the matrix to verify qualitative assumptions in the matrix. The licensee's procedures indicated that the relative risk of a plant configuration could be obtained from the Nuclear Safety Assessment Group in Milwaukee when deemed necessary.

A review of recent control room log books and danger tag index log books did not identify any unacceptable plant configurations. Different plant configurations were discussed with outage coordinators and configurations that were not indicated on the matrix were avoided. If the configuration could not be avoided, the outage coordinators indicated that the Nuclear Safety Assessment Group was contacted to evaluate the plant configuration.

The licensee indicated that there were plans in the upcoming year for an on-line risk monitor and an expanded matrix. The expanded matrix would include further probabilistic safety assessment model insights and plant configuration evaluations.

The licensee used the "PBNP Shutdown Safety Assessment and Fire Condition Checklist" (PBF-9700 Revision 1) for shutdown evaluations. The checklist was based primarily on deterministic evaluations. The licensee's process was based on standard industry approach with guidance from the Institute for Nuclear Power Operations, Nuclear Management Resource Council, and Electric Power Research Institute. During the inspection, the licensee indicated that a shutdown probabilistic safety assessment

model was planned to be developed in the upcoming year. Insights from the developed shutdown model would be used in the licensee's shutdown safety assessment procedures.

c. Conclusions

The team concluded that the licensee's processes for assessing plant risk resulting from taking equipment out of service during at power and shutdown conditions were adequate. Licensee plans for an on-line risk monitor, additional probabilistic safety assessment model insights in the matrix, and a shutdown probabilistic safety assessment model will strengthen the licensee's processes for assessing plant risk resulting from taking equipment out of service during at-power and shutdown conditions.

M1.6 (a)(1) Goal Setting and Monitoring and (a)(2) Preventive Maintenance

a. Inspection Scope

The team reviewed program documents in order to evaluate the process established to set goals and monitor under (a)(1) and to verify that preventive maintenance was effective under (a)(2) of the maintenance rule. The team also discussed the program with appropriate plant personnel and reviewed the following systems in depth:

(a)(1) systems	(a)(2) systems
Rod Drive Control	Circulating Water
Emergency Diesel Generators	Component Cooling Water
Instrument Air	125 Volt DC
Turbine and Auxiliaries	
Service Water	
Residual Heat Removal	

The team reviewed each of these systems to verify that goals or performance criteria were established commensurate with safety, that industry wide operating experience was taken into consideration, where practical, that appropriate monitoring and trending were being performed, and that corrective actions were taken when an SSC failed to meet its goal or performance criteria or experienced a maintenance preventible functional failure.

The team reviewed the Point Beach structure monitoring program and documentation to determine if structures at Point Beach were being monitored in accordance with the maintenance rule and in accordance with industry and NRC guidance. A review of the performance criteria and monitoring established for structures within scope was performed. Structures evaluated by the team included buildings, enclosures, storage tanks, earthen structures, and passive components and materials housed in these structures. In addition, the team assessed how structures within scope were monitored for degradation.

b. Observations and Findings

Performance criteria were established in accordance with procedure NP 7.7.5, "Determining, Monitoring and Evaluation Performance Criteria for the Maintenance Rule," Revision 3, dated October 15, 1997. Procedure NP 7.7.5 specified concurrence in revised performance criteria by the system engineer, an operations representative, a maintenance representative, and, for high safety significance systems, a probabilistic safety assessment representative. The team identified a weakness in that there was no concurrence required by an individual responsible for the maintenance rule program. Consequently, inconsistencies among performance criteria for systems could arise. In addition, the potential existed for performance criteria to be formally approved even though the criteria did not meet the intent of the maintenance rule program. During the inspection, the Maintenance rule Coordinator stated that a performance criteria document had to be returned to a system engineer for revision, even though the document had all of the necessary concurrences, because the revised criteria would not meet the intent of the program. Although necessary oversight was occurring in practice, the team considered the failure to specify the oversight as part of the formal process a weakness.

b.1 Reliability and Unavailability Performance Criteria

In general, performance criteria for both availability and reliability were appropriately established to gauge system performance. One exception was identified with respect to the 120 volt AC electrical system as outlined below. Performance criteria were, in general, either monitored over two years or two cycles.

120 Volt AC Electrical System: The 120 Volt AC electrical system was a low safety significance system with a reliability performance criterion of 1 functional failure per unit over two years. The 120 Volt AC system included emergency lighting. A functional failure was defined as two adjacent emergency lights failing an eight hour test. This performance criterion was adopted in October 1997. The team noted that, theoretically, a 50 percent failure rate could be permitted by the functional failure definition. In addition, the team questioned whether failures of two adjacent emergency lights to pass an eight-hour test would be recorded as a functional failure if the tests for the lights were performed several months apart. In response to concerns raised by the team, the licensee revised the performance criteria to also specify that the emergency light failure rate did not exceed five percent. The five percent failure rate was consistent with the two to three percent failure rate typically experienced during battery tests. The failure to have performance measures which limited emergency lighting failures to an acceptable value was a violation example of 10 CFR 50.65 (VIO 50-266/97025-02(DRS); 50-301/97025-02(DRS)).

For the reliability and unavailability performance criteria, the licensee had tied the performance criteria to the PSA93 database. The performance criteria were estimated based on data from January 1, 1985 to September 5, 1990, presented in the Probabilistic Safety Assessment. The probabilistic safety assessment data used was presented in the following tables:

Table 4.2-6 Summary of component failure rates.
Table 4.2.7 Summary of maintenance unavailabilities.

Based on the established performance criteria, failure rates were estimated and a sensitivity analysis had been performed. However, during the inspection the licensee identified an error in the estimation of the failure rates for the sensitivity analysis. The licensee stated that the sensitivity analysis will be redone, thus the sensitivity analysis is an Inspection Follow-up Item (IFI 50-266/97025-03(DRS); 50-301/97025-03(DRS)).

b.2 Plant Level Performance Criteria for Low Safety Significance Normally Operating SSCs

Plant level performance criteria were established for low safety significance normally operating SSCs using the guidelines contained in Nuclear Management Resource Council 93-01. Plant level performance criteria were established for unplanned reactor trips (one per year per unit), capacity factors (87.7 percent for Unit 1 and 88.1 percent for Unit 2), and unplanned safety system actuations (one per year for both units combined). The team did not identify any concerns with respect to the plant level criteria.

b.3 Performance Criteria Bases Documentation

Many of the performance criteria, as written, were easily misinterpreted. The documentation maintained on the computer database was not always in agreement with the approved criteria. Examples were as follows:

- The performance criterion documented on the computer database for the radiation monitoring system indicated that the unavailability criterion was 18,670 hours total unavailability for all 107 channels which represented 2 percent unavailability. However, the approved criterion was 5 percent unavailability. Although the criterion of 18,670 hours total unavailability had been significantly exceeded in 1997, the licensee failed to recognize the documentation error because the system engineer tracking the data was relying upon a different portion of the database which correctly used the 5 percent figure.

- The documented performance criteria for the residual heat removal system stated that unavailability time for the pumps when needed would be counted. The unavailability criteria implied that only pump unavailabilities would be counted against the residual heat removal system criteria and other unavailabilities, such a heat exchangers, would not be counted.

- A number of performance criteria had qualifiers which tended to confuse rather than clarify what the performance criteria applied to. In discussions with the plant staff, the team learned that the qualifiers did not apply and that the performance criterion was intended to correctly reflect train failures. For example, the reliability performance criterion for spent fuel pool cooling was two functional failures per 24 months "based on 2 each redundant equipment." Although the system engineer couldn't explain the phrase "based on 2 each

redundant equipment," the engineer stated that train failures would be counted as functional failures. In another example, the unavailability criterion for the nuclear instrumentation system had the qualifier "based on bistable replace" which implied that only unavailabilities associated with bistable replacements would be counted against the criteria. Again, discussions with the maintenance rule coordinator indicated that all unavailabilities when the system was required would be appropriately counted. The team identified questionable qualifiers for performance criteria associated with the following systems:

345 kV Electrical
Component Cooling Water
120/208 Vac Electrical
Nuclear Instrumentation
Reactor Protection
Spent Fuel Pool Cooling

- One of the reliability performance criteria for the fuel handling system was one functional failure per "usage of RCCA change basket." As written, the criterion implied that it was acceptable for the rod control cluster assembly change basket to fail every time it was used. The maintenance rule coordinator clarified that criteria applied to every cycle that the change basket was used.

- Discrepancies existed between approved performance criteria documentation and the computer database documentation. In most case, the criteria were properly reflected in part of the database information, but not consistently throughout. However, the inspector identified that a condition monitoring criterion for the incore flux mapping system was completely omitted from the computer database information. Discrepancies between approved criteria and the computer database documentation were identified for the following systems:

  Primary Containment Integrity
  Engineered Safety Features
  Incore Flux Mapping
  Main Steam
  Fuel Oil
  345 kV Electrical
  Reactor Coolant
  Reactor Protection

As previously discussed in section M1.1, many documentation discrepancies were also identified in the scoping records. The licensee's May 1997, audit also identified problems with accurate documentation. To ensure that recurrence of documentation discrepancies is addressed and corrected, this is an Inspection Follow-up Item (IFI 50-266/97025-04(DRS); 50-301/97025-04(DRS)).

b.4 Goals Established for (a)(1) SSCs

In general, the established goals for systems classified as (a)(1) were the same as the performance criteria. In some cases, specific goals in addition to existing performance criteria were adopted. For example, the diesel generator system had an additional goal of no failures in two years for disposition to (a)(2) status. Similar supplemental goals had been established for the 480 Volt electrical, instrument air, and rod drive control systems. Goals for (a)(1) systems were conservatively established to ensure eventual disposition to (a)(2) classification was appropriate; however the team identified inappropriate goals for the reactor coolant and residual heat removal systems as discussed below:

Reactor Coolant System: The reactor coolant system was a high safety significance system which was classified as (a)(1) and monitored against unavailability and reliability goals as follows:

Unavailability:	1000 hours total/unit/train/24 months for power operated relief valves when block valve closed at power
Failed Starts or Reliability:	1 functional failure/train/24 months at power based on power operated relief valves and blocks
	0 functional failures/24 months based on reactor coolant system leakage limit
	1 functional failure/24 months/train based on low temperature over-pressure protection and blocks when <355 degrees Fahrenheit until vented - none simultaneously
Other:	0 functional failures/24 months for pressurizer safeties overpressure protection function

The above goals were derived from the (a)(2) performance criteria and were adopted in October 1997.

Although the reactor coolant system included the reactor vessel level indication system, failures and unavailabilities of reactor vessel level indication would not be captured by the above performance criteria. Also, the criteria for the low temperature overpressure protection and associated blocks, although monitored over 24 months, were only applicable for several weeks a year. The team calculated that if a unit were in conditions requiring low temperature over-pressure protection for only three weeks a year, the permitted failure rate would be over four functional failures per train per year

rather than the 0.5 failures per train per year implied by the performance criteria. Given the risk significance of low temperature over-pressure protection function, the team considered the failure rate permitted by the goal to be unacceptably high. The failure to have goals which gauged reactor vessel level indication system performance and limited low temperature over-pressure protection failures to an acceptable value was a violation of 10 CFR 50.65 (VIO 50-266/97025-05a(DRS); 50-301/97025-05a(DRS)).

Residual Heat Removal System: The unavailability goal for the residual heat removal system adopted in October 1997, was 40 hours per train per two years total unavailable time for pumps when needed. Because the unavailability goal was limited to the pumps, unavailabilities for other reasons were not counted. For example, the two unavailabilities due to cracks in a component cooling water line from a residual heat removal heat exchanger were not counted even though both events caused one train of residual heat removal to be out of service (see Section M2.1). The two unavailabilities totaled 45 hours and, by themselves, would have caused the 40-hour goal for the residual heat removal system to be exceeded. Consequently, the goal adopted in October 1997, was inadequate in that not all unavailabilities were being captured. The failure to have a goal which captured all unavailabilities leading to train unavailability was a violation of 10 CFR 50.65 (VIO 50-266/97025-05b(DRS); 50-301/97025-05b(DRS)).

b.5 Structures and Structure Monitoring

The licensee's program for inspection of structures under the maintenance rule was documented in Wisconsin Electric Nondestructive Examination Procedures Manual Procedure NDE-751 "Structural Surveillance of Containments and General Structures," Revision 5, dated October 31, 1997. Review of the program and discussion with the responsible engineers determined that while the program was adequately established and had performance criteria for monitoring structures under (a)(2) of the rule in accordance with the guidance provided in Nuclear Management Resource Council 93-01, and Regulatory Guide 1.160, the licensee had not provided criteria for moving structures from (a)(2) to (a)(1) of the rule. This was not consistent with the current guidance provided by Regulatory Guide 1.160. The Maintenance Rule Coordinator explained that the criteria had been inadvertently omitted from the most recent revision of the procedure and it would be reissued with the criteria included. Verification that the structure monitoring procedure is reissued with these criteria included is an Inspection Follow-up Item (IFI 50-266/97025-06(DRS); 50-301/97025-06(DRS)).

The team determined that the licensee had completed the historical review of structures in accordance with Nuclear Management Resource Council 93-01. The team noted that PBNP had proactively started a structural monitoring program with documentation in 1989, prior to the implementation of the maintenance rule. The historical review of structures found all site structures inspected to date to be acceptable, and resulted in the licensee placing all the applicable structures in category (a)(2).

At the time of this inspection, the licensee had completed most of the required baseline inspections for both Units 1 and 2. The licensee noted that the hanger baseplate-to-

structure inspection had not yet been performed, but was forthcoming. The team performed an inspection of the material condition of the exterior and interior of the circulating water/service water pumphouse, emergency diesel generator building and the gas turbine building. The turbine block, the upper level of the residual heat removal heat exchanger room, the auxiliary feedwater pump room walls, and selected safety related hanger baseplate-to-wall structure supports were also inspected.

A comparison of Procedure NDE-751, "Structural Surveillance of Containments and General Structures," Revision 5 dated October 31, 1997, and the PBNP procedure NDE-754, "Visual Examination of (VT-3) Nuclear Power Plant Components," Revision 5, dated October 31, 1997, revealed that section 9.1.1.h of procedure NDE-754 stated the jurisdictional boundaries between pipe support and building structure and that baseplate, bolts and nuts shall be considered building structure and as such will not considered within the inservice inspection boundary but rather will be monitored under the building structure program. However, this area was not contained in the licensee's structure monitoring program instruction, Procedure NDE-751, dated October 31, 1997. An auxiliary feedwater system concrete structure-to-hanger baseplate gap greater than the maximum allowable was noted by the team in the auxiliary feedwater pump room.

c. Conclusions

In general, performance criteria were appropriately established to gauge system performance. Performance criteria for one system did not adequately measure system performance. Established goals were generally conservative; however, two (a)(1) systems had inappropriate goals. A weakness existed in the process for establishing and revising performance criteria in that new or revised criteria were not formally reviewed by an individual responsible for the maintenance rule program. A number of performance criteria documentation discrepancies were also identified.

With the exception of the inadvertently omitted performance monitoring criteria, the structure monitoring program was well-organized and comprehensive. Overall, with the exception of baseplate-to-structure hanger support gaps the structures inspected were in good condition. The licensee noted that the hanger baseplate-to-structure inspection had not yet been performed, but was forthcoming.

M1.7 Use of Industry-wide Operating Experience

a. Inspection Scope

Paragraph (a)(1) of the maintenance rule states that goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. Paragraph (a)(3) of the maintenance rule states that performance and condition monitoring activities and associated goals and preventive maintenance activities shall be evaluated at least every refueling cycle. The evaluation shall be conducted taking into account industry operating experience. The team reviewed the program to integrate industry operating experience into the maintenance rule monitoring program. Ten System Engineers, the Maintenance Rule Coordinator and the Point

Beach Industry Operating Department Coordinator were interviewed to determine the extent of knowledge of industry operating experience information as applicable to maintenance rule processes.

b. Observations and Findings on Use of Industry-wide Operating Experience

The team reviewed NP 5.3.2, Revision 4, "Industry Operating Experience Review Program," and noted that it was a detailed procedure for accumulating, evaluating, and acting on industry operating experience. The team also noted that program was properly linked to the maintenance rule program in section 6.6 of the implementation procedure AM 3-4. Discussions with system engineers indicated that they actively participated in industry users' groups, communicated frequently with system engineers at other plants, and routinely received industry operating experience information related to their systems from the Point Beach Industry Operating Department Coordinator. However, from existing documentation it was not evident that information from industry operating experience was incorporated into goal development for (a)(1) systems or the 1996 Point Beach Periodic Assessment. The Maintenance Rule Coordinator stated that the need to increase the awareness of system engineers for documentation of industry operating experience in the maintenance rule program was recently identified.

c. Conclusions for Use of Industry wide Operating Experience

The industry operating experience review program was well-organized and properly linked to the maintenance rule program. System engineers were clearly using industry experience information and generally understood the need to incorporate it into the maintenance rule program. However, there was little documentation showing consideration of industry operating experience in maintenance rule activities.

M2 Maintenance and Material Condition of Facilities and Equipment (61706, 71707)

M2.1 General System Review

a. Inspection Scope

The team conducted a detailed examination of nine systems from a maintenance rule perspective to assess the effectiveness of the program when it was applied to individual systems.

b.1 Observations and Findings for the Instrument Air System

The instrument air system was considered a high safety significance system with performance criteria to monitor unavailability at the train level and reliability of the compressor and the rest of the system components. The instrument air system was placed in (a)(1) on May 20, 1997, as a result of two maintenance preventable functional failures that occurred in a one-year period. In May 1996 and February 1997, discharge valve capscrews backed off, causing the valve to lose its pressure-retaining capabilities. Although the function of instrument air was not lost, the event was caused by improper

maintenance because the capscrews were not properly tightened. Undertorquing and subsequent loosening of the valves prevented the valves from sealing against the back leakage. This caused the interstage pressure to rise and the interstage pressure relief to lift. Proper torque values were obtained from the manufacturer.

In November 1997, the performance criteria of the instrument air system were revised. The revised criteria added unavailability which was applied retroactively to include total unavailability of 720 hours per train. The licensee determined that the criterion was exceeded with the unavailability totaling 836 for one train and 826 hours for the other trains.

The team determined that proper goals and corrective actions were established for this system. The instrument air system would remain in (a)(1) status until no more equipment failures occurred due to improper torquing before March 1998. The goals established appeared appropriate to return the system to (a)(2).

b.2 Observations and Findings for the Diesel Generator System

The diesel generator system was considered a high safety significance, standby system with performance criteria to monitor reliability and unavailability. The diesel generator system was being monitored under (a)(1) due to diesel generators G01 and G02 not meeting unavailability criteria prior to the maintenance rule becoming effective in July 1996. Each one of the four diesel generators was limited to 216 hours of unavailability under the old performance criteria. A number of problems were causing all four diesel generators (including G03 and G04) to be unavailable. These problems included maintenance or modification of the starting air, the governors, generator bearing insulation, the fuel tank level switches, and the control relays. The G01 and the G02 diesel generators were experiencing problems with control and relay failures to the speed input circuits, and the time delay relays contained in the engine start logic. The primary fault with the control system was the inability to reliably measure engine speed and the inability to maintain accurate timing settings on the control systems time delay relays. The speed sensing relays were replaced in 1997.

Corrective and preventive maintenance, and modifications caused the diesel generators to be unavailable for an extensive period of time in 1997. According to the data provided by the licensee, G01, G02, G03 and G04 were unavailable 1,618, 2,802, 1,504, and 461 hours, respectively (data does not include the month of November). The licensee stated that the extensive unavailability was necessary to correct multiple problems with the diesel generators. It should be noted that, although Point Beach was a two-unit site, there were two redundant trains and consequently, only two of the four diesel generators of opposite trains were required to be operable. In November 1997, the unavailability performance criterion was changed from individual diesel generator unavailability to total unavailability of 316 hours per train. This change in the performance criteria would allow an individual emergency diesel generator to be unavailable approximately 4 percent of the time.

The diesel generators would remain in (a)(1) status for 24 months following the overhaul and inspections that have been completed or scheduled in 1997. The goals established appeared appropriate to return the system to (a)(2).

b.3 Observations and Findings for the 125 Volt DC System

The 125 volt DC system was considered a high safety significance system with performance criteria to monitor reliability and unavailability. The DC system was being monitored under (a)(2) of the maintenance rule. System performance was good; no maintenance preventable functional failures or unavailability concerns were identified.

b.4 Observations and Findings for the Rod Drive Control System

The rod drive control system was a low safety significance system with performance criteria to monitor reliability. The system was classified as (a)(1) on June 26, 1996, based primarily on rod drop problems identified during the historical review. Although the problems were identified with respect to Unit 1, both units were conservatively classified as (a)(1). The established goals were the same as existing performance criteria which were adequate for monitoring performance. The corrective actions which were initiated included monitoring control rod drive mechanism shroud cooling temperatures to ensure effective cooling, taking inductance and cold resistance measurements for control rod drive coils during outages, and replacing control rod drive stationary coils as appropriate. No functional failures had been identified since the corrective actions were initiated. Although the initial plan was to monitor the system performance for two cycles before returning the system to (a)(2) classification, the system engineer was considering extending the monitoring period to provide greater assurance that the corrective actions were effective.

b.5 Observations and Findings for the Circulating Water System

The circulating water system was a low safety significance system with performance criteria to monitor reliability. Although two load reduction events, one for each unit, due to circulating water had appropriately been identified, the system reliability performance criteria had not been exceeded.

b.6 Observations and Findings for the Component Cooling Water System

The component cooling water system was a high safety significance system with performance criteria to monitor reliability and unavailability. During October 1997, the reliability performance criterion was revised to 1 functional failure per unit over a two-year period. Prior to October, the reliability performance criterion was one maintenance preventable functional failure per unit over a two-cycle period.

On October 9, 1996, the Unit 2 component cooling water surge tank was observed trending down, indicating a small leak in the component cooling water system. On October 10, 1996, the six gallons per hour leak was discovered to originate from a one-inch pipe on the component cooling water system. When the leak was discovered,

water was observed spraying from a crack in the pipe adjacent to a fitting. Condition Report 96-1109, dated October 10, 1996, described the crack as a 50 percent circumferential crack which had propagated due to cyclic fatigue. Similar piping was radiographed and evaluated for potential fatigue failures. The team concurred with the condition report's conclusion that the failure was not a maintenance preventable functional failure. This event was also documented by Licensee Event Report 96-002, dated November 8, 1996.

Similarly, on July 29, 1997, the Unit 2 component cooling water surge tank was again observed trending down indicating another leak in the system. A 0.38 gallon per minute leak (documented by Condition Report 97-2310, dated August 8, 1997) was discovered originating from a partial circumferential crack adjacent to another fitting in same section of piping from which leakage had occurred in October 1996. The team noted that the leakage rate was over three times greater than that observed from the October 1996 leak. Root Cause Evaluation 97-053, dated October 23, 1997, documented the cause of the July 1997 failure as also being cyclic fatigue. The root cause evaluation also noted the section of pipe with the July 1997, leak was not one of the locations radiographed after the October 1996, leak event though it was adjacent to piping which had been replaced as a result of the October 1996, leak. This event was also documented by Licensee Event Report 97-004, dated August 28, 1997.

During this inspection, the system engineer indicated that there had been no failures which were being tracked against the component cooling water reliability criteria even though he was aware of the above two events. The engineer considered the leakage experienced during both events to be within the plant makeup capabilities for the system. Even if the pipe had ruptured, the leakage would have been within the makeup capability due to the relatively small pipe size. However, the engineer failed to consider the system design basis as a closed system and that potential makeup sources were non-safety related. As such, credit could not be taken for the non-safety related sources for design basis events such as a seismic event. Consequently, the team considered both events to be functional failures. In response to the issues identified by the team, the licensee classified the component cooling water system as (a)(1) on November 20, 1997.

The team recognized that at the time the failures occurred, the performance criteria for the system which had been in place at the time were not exceeded. However, the number of failures exceeded the performance criteria signed by the system engineer on October 20, 1997. Although system was still under going a historical review due to the revised performance criteria, the significance of the two events would have suggested immediate recognition that the reliability performance criteria had already been exceeded.

In addition to the two leakage events, the team noted that condition reports 97-3219 (dated October 6, 1997) and 97-3335 (dated October 15, 1997) documented transfer of water from the Unit 2 component cooling water system to the Unit 1 system. The inter-system leakage was estimated to be 6-1/2 gallons per day. The system engineer stated that operation of the two systems cross-connected was prohibited because doing so

would place the system in an unanalyzed condition. The team questioned why the inter-system leakage was not considered a functional failure and requested an explanation from the licensee. At the time of the inspection, the licensee was still reviewing the issue. This issue is considered an unresolved item pending review of the licensee's functional failure determination and associated justification (URI 50-266/97025-07 (DRS); 50-301/97025-07(DRS)).

b.7 Observations and Findings for the Service Water System

The service water system was considered a high safety significance system and had performance criteria to monitor both availability and reliability. Unavailability was measured by Limiting Condition for Operation time, tracked whenever either unit was above cold shutdown conditions. The service water system contained six common pumps for both units; pump unavailability was tracked on an individual pump basis to avoid 'masking' (e.g., high unavailability) the poor performance of one train by averaging or summing it with other trains. Other availability criteria were applicable for the entire system. Reliability performance criteria were established for: pumps and pump motors, balance of system components, and through-wall pipe leaks.

The service water system was initially classified as (a)(1) on November 19, 1996, due to three pinhole piping leaks over a four-month period. Although the leaks did not impact system operability, and therefore were not classified as functional failures, the (a)(1) classification was conservatively made to focus attention on service water piping integrity issues. An end-of-year maintenance rule review of the system by the system engineer identified a number of past functional failures which also warranted the (a)(1) classification. The functional failures included: service water pump motor bearing failures, inoperability of component cooling water heat exchanger manual outlet throttle valves, failure of the diesel generator heat exchanger outlet air-operated valve to open, and excessive leakage for the spent fuel pool outlet motor-operated valves. Although these were examples of inadequate problem and event evaluation, they did not affect the licensee's classification of the system and its functions.

Monitoring against goals began January 1, 1997. Performance goals were recently reviewed and updated to better conform with probabilistic safety assessment assumptions for unavailability. The licensee stated that the service water system would remain an (a)(1) system until goals had been met for at least one rolling period. The team reviewed the corrective action for these failures, the goals, and monitoring under the (a)(1) status, and concluded that the corrective action, goals, and monitoring were appropriate.

b.8 Observations and Findings for the Residual Heat Removal System

The residual heat removal system was considered a high safety significance system with performance criteria to monitor both availability and reliability. The unavailability criterion was established based on Institute for Nuclear Power Operations goals, on values assumed in the probabilistic safety assessment, or, if higher values were used, by showing that they did not involve a significant collective increase in baseline core

damage frequency. Unavailability was based solely on the total unavailability time for the pumps when needed (which included testing and oil changes). Reliability was based on motor-operated valves, pumps, and other failures not considered in the probabilistic safety assessment.

The team compared periods of unavailability between the component cooling water and residual heat removal system. The review revealed that there were two times that a Unit 2 residual heat removal heat exchanger was unavailable due to cracks in the component cooling water piping. The component cooling water system engineer stated that the unavailability would be counted against residual heat removal system because most of the component cooling water system was available and it was only the one residual heat removal train that was affected. However, the residual heat removal unavailability criterion referred only to the pumps. Consequently, unavailabilities due to heat exchanger problems were not tracked even though that one train was inoperable. A potential violation regarding not correctly recording unavailability time due to the incorrect performance criteria is discussed in Section M1.6.

Per an expert panel review on November 3, 1997, the residual heat removal systems for Units 1 and 2 were placed into (a)(1) monitoring status due to a high safety significance functional failure which occurred on August 4, 1997 (condition report 97-2388). Due to excessive pump seal leakage, the Unit 2 residual heat removal system exceeded its maintenance rule availability criteria on that date and although the Unit 1 residual heat removal system had not exceeded any performance criteria, it also was conservatively placed into (a)(1) status until a root cause evaluation could be completed to determine whether the failure was a generic issue. Per condition report 97-3760, when the root cause evaluation (RCE 97-058) was complete, the licensee planned to reconvene the expert panel to review whether (a)(1) status was still appropriate. If the system remained (a)(1), the licensee planned to develop an action plan for improved system performance and would establish system goals at that time.

The residual heat removal system engineer had lead responsibility for trending, determining maintenance preventable functional failures, and incorporating residual heat removal system industry wide operating experience. The team considered that the residual heat removal MRLIN quarterly updates were particularly well documented with expert panel members listed for traceability purposes. The team reviewed the goals and monitoring under the (a)(1) status, and concluded that with exception of the restricted unavailability criteria noted above, the corrective action, goals and monitoring were appropriate.

b.9 Observations and Findings for the Turbine Generator and Turbine Related Systems

The turbine system was considered a low safety significance system with performance criteria to monitor reliability. The turbine-related, electro-hydraulic control, and turbine and feedwater pump lube oil and seal oil systems were classified as (a)(1). Although no one individual system had exceeded performance criteria, a cumulative concern for all systems associated with 1993 - 1996 events, which had resulted in challenges to safety

systems (as a result of turbine trips causing reactor trips), was the justification for the (a)(1) scoping.

The team reviewed the maintenance history, condition reports, tag-out and unit logs, and the system scoping document. The turbine and turbine lube oil systems were walked down, and the team considered that the material condition was good. The two system engineers were interviewed, and the team concluded that they were aggressive about improving the performance and reliability of the turbine and turbine-related systems. The licensee's efforts indicated that they were diligent in their efforts to improve turbine availability and reliability. Of particular note was the development of a Comprehensive Equipment Maintenance Matrix to identify proper equipment maintenance intervals for major turbine generator equipment and support systems. This action plan item resulted in a historical tabulation of the maintenance done on both units and a tabulation of all Point Beach applicable Availability Improvement Bulletins, Customer Advisory Letters, and Westinghouse Operation and Maintenance Memos along with the current status of Point Beach implementation for each.

The team considered that the goals were reasonable and achievable, industry wide operating experience was taken into consideration where practical, and that appropriate monitoring and trending were being performed,

c. Conclusions for General System Review

With the exception of the residual heat removal performance criteria problems and the lack of timely identification of exceeding the revised component cooling water criteria, the maintenance rule was properly implemented for the systems the team examined. Classifying a number of systems (a)(1) due to performance concerns, although no performance criteria had been exceeded, was considered a strength.

M2.2 Material Condition

a. Inspection Scope

In the course of verifying the implementation of the maintenance rule, the team performed walkdowns using Inspection Procedure 71707, Plant Operations, to examine the material condition of the systems listed in Section M1.6.

b. Observations and Findings

With minor exceptions, the systems were free of corrosion, oil leaks, water leaks, trash, and based upon external condition, appeared to be well maintained.

The team noted, however, that the safety-related swing battery electrolyte levels were at the low level mark for about 20-25 percent of the battery cells. This battery was used to supply safety related loads when one of the other safety related station batteries was taken out of service. This situation was discussed with the system engineer who committed to have maintenance fill the battery cells with water. However, this problem

was an indication that the maintenance department and the system engineer were not carefully monitoring the electrolyte levels of the swing battery.

c. Conclusions

In general, the material condition of the systems examined was acceptable. However, the maintenance department and the system engineer were not effectively monitoring battery electrolyte levels.

M7 Quality Assurance in Maintenance Activities (40500)

M7.1 Licensee Self-Assessments of the Maintenance Rule Program

a. Inspection Scope

The team reviewed the following documents related to self assessments and audits conducted to evaluate implementation of the maintenance rule.

- Nuclear Energy Institute letter, Rains - Jilek, Maintenance Rule Site Assist Visit, dated October 20, 1995
- Quality Assurance Audit Report, A-P-95-12, "Maintenance Rule Program," dated November 30, 1995
- Point Beach Nuclear Plant Self Assessment, "PBNP's Implementation of the NRC's Maintenance Rule," dated June 25, 1996
- Quality Assurance Audit Report, A-P-97-14, "PBNP Implementation of the NRC Maintenance Rule (10CFR50.65)," dated June 5, 1997

The team also held discussions with members of the licensee's staff concerning activities associated with the above audits and Nuclear Energy Institute evaluation, including how findings and observations were received and handled through closure.

b. Observations and Findings

The team reviewed the audits and Nuclear Energy Institute evaluation reports listed above during the in-office inspection. Based on this review, the team concluded that the licensee was aware of problems with their implementation of the maintenance rule. A number of the issues identified in the audits were also noted during this maintenance rule inspection. These audits and evaluation highlighted the licensee's need to improve the maintenance rule program. Recent audits and related reports were thorough and considered by the team to be a programmatic strength.

During this inspection, the team sampled several audit findings and observations to ensure that identified items were appropriately handled and dispositioned. No major omissions were identified. The majority of the findings and observations had been

addressed. However, the team noted that several of the audits findings or observations were late to closure. For example, the May 1997 quality assurance audit identified 46 findings and observations. One of the 14 open audit findings or observations had exceeded its due date. Twenty of the 32 closed audit findings or observations exceeded their due dates upon closure. Based on similar items identified both by the licensee and the NRC team and the number of audit items remaining open, this was identified as an Inspection Follow-up Item (IFI 50-266/97025-08(DRS); 50-301/97025-08(DRS)).

c. Conclusions

Audits and a Nuclear Energy Institute evaluation provided good observations and findings of areas associated with maintenance rule implementation. The most recent audit (May 1997) of the maintenance rule program was extremely thorough and was considered a programmatic strength. Corrective actions sampled by the team were appropriately implemented.

III. Engineering

E4. Engineering Staff Knowledge and Performance (62706)

E4.1 Engineer's Knowledge of the Maintenance Rule

a. Inspection Scope (62706)

The team interviewed engineers and managers to assess their understanding of probabilistic risk assessment, the maintenance rule, and associated responsibilities.

b. Observations and Findings

The team interviewed the system engineers assigned responsibility for SSCs selected, and walked down systems with them. The system engineers were experienced and knowledgeable about their systems. Maintenance rule training and probabilistic safety assessment familiarization in risk assessment were provided to the system engineers. System engineer responsibilities for the maintenance rule included the maintenance preventable functional failure decision process and the preparation of (a)(1) corrective action plans.

Most system engineers were appropriately conservative in making functional failure determinations for their system and recommending (a)(1) classification when warranted. One exception was identified when a system engineer did not accurately consider the design basis for his system for functional failure determinations (see section M2.1).

Although the system engineers were generally familiar with the maintenance rule, one system engineer did not recognize that failure to properly position a valve, when returning a system to service from a maintenance activity, could be considered a

maintenance preventable functional failure. In addition, one system engineer did not realize that out of service time was counted against an SSC in situations involving operator error.

c. Conclusions

System engineers were experienced and knowledgeable about their systems. System engineers were generally familiar with the maintenance rule program, although some system engineers were not knowledgeable of certain aspects of the program.

V. Management Meetings

X1 Exit Meeting Summary

The team discussed the progress of the inspection with licensee representatives on a daily basis and presented the inspection results to members of licensee management at the conclusion of the inspection on November 21, 1997. The licensee acknowledged the findings presented. The team asked the licensee whether any materials examined during the inspection should be considered proprietary; none was identified.

In the two weeks following the November 21, 1997 exit meeting, the licensee staff provided additional information in response to questions that the team had asked prior to the exit. On receipt of this information, the team reconsidered and revised two findings. A telephone exit with licensee staff and management was held on December 15, 1997. The licensee acknowledged the revised findings.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Boldt, Special Asst. to Site Vice President
K. Craft, System Engineer
D. Duenkel, System Engineer
F. Flentje, Regulatory Specialist
S. Goukas, Probabilistic Safety Assessment Engineer
H. Hanneman, Continuous Safety and Performance Group
N. Hoefert, Continuous Safety & Performance Improvement Manager
P. Huffman, Senior Project Engineer - Site Engineering
C. Jilek, Maintenance Rule Coordinator
D. Johnson, Regulatory Services & Licensing Manager
J. Knorr, Regulation and Compliance Manager
D. Kunkel, Plant Outage Coordinator
P. Kurtz, Senior Engineer
P. Lightbody, Scheduling Technician
M. Millen, Nuclear Safety Analysis Group Head
M. Reddemann, Quality Assurance Manager
B. Sasman, Sr. Project Engineer, Civil/Seismic Engineering
J. Schroeder, System Engineer
J. Schweitzer, Site Engineering Manager
P. Snyder, System Engineer
J. Sopata, System Engineer
T. Williams, Shift Outage Coordinator
E. Ziller, FIN (fix it now) Team

NRC

F. Brown, Senior Resident Inspector
P. Louden, Resident Inspector
A. McMurtray, Senior Resident Inspector

LIST OF INSPECTION PROCEDURES USED

IP 62706:	Maintenance Rule
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 71707:	Plant Operations

LIST OF ITEMS OPENED

50-266/97025-01(DRS); 50-301/97025-01(DRS)	VIO	Scoping
50-266/97025-02a(DRS); 50-301/97025-02a(DRS)	VIO	120 VAC system performance criteria
50-266/97025-03(DRS); 50-301/97025-03(DRS)	IFI	Flawed Sensitivity Analysis
50-266/97025-04(DRS); 50-301/97025-04(DRS)	IFI	Documentation Discrepancies
50-266/97025-05a(DRS); 50-301/97025-05a(DRS)	VIO	Reactor Coolant system goals
50-266/97025-05b(DRS); 50-301/97025-05b(DRS)	VIO	Residual Heat Removal system goals
50-266/97025-06(DRS); 50-301/97025-06(DRS)	IFI	Revised Structure Monitoring Procedure
50-266/97025-07(DRS); 50-301/97025-07(DRS)	URI	Intersystem Leakage Evaluation
50-266/97025-08(DRS); 50-301/97025-08(DRS)	IFI	Audit Follow-up

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
IFI	Inspection Follow-up Item
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
SSC	Structure, System, or Component
URI	Unresolved Item

PARTIAL LIST OF DOCUMENTS REVIEWED

AM 3-4, "Implementation of the Maintenance Rule at PBNP," Revision 2, May 30, 1997.

A-P-95-12, "Maintenance Rule Audit - Audit of the Development and Implementation Stages of the PBNP Maintenance Rule Program"

A-P-97-14, "Maintenance Rule Audit - PBNP Implementation of the NRC Maintenance Rule (10 CFR 50.65)"

NP 5.3.1, "Condition Reporting System," Revision 6, September 24, 1997

NP 5.3.2, "Industry Operating Experience Review Program," Revision 4, July 25, 1997

NP 7.7.4, "Scope and Risk Significant Determination for the Maintenance Rule," Revision 3, October 15, 1997.

NP 7.7.5, "Determining, Monitoring and Evaluating Performance Criteria for the Maintenance Rule," Revision 3, October 15, 1997.

NP 7.7.6, "Work Order Review and MPFF Determination for the Maintenance Rule," Revision 2, October 15, 1997.

NP 7.7.7, "Guideline for Maintenance Rule Periodic Report," Revision 2, October 15, 1997

NP 10.1.1, "TS Equipment OOS/Voluntary Entry into an LCO," Revision 7, September 27, 1996.

NP 10.2.1, "Outage Planning, Scheduling, and Management," Revision 9, October 15, 1997.

NP 10.2.2, "Non-Outage Work Planning," Revision 1, September 15, 1995.

NP 10.3.6, "Outage Safety Review and Safety Assessment," Revision 3, August 29, 1997.

SEM 4.3, "Development of Component Maintenance Programs," Revision 1, March 9, 1995.

PBF-9133a, "Cross Train Voluntary LCO & Testing Restrictions," Revision 1, October 27, 1997.

PBF-9700, "PBNP Shutdown Safety Assessment and Fire Condition Checklist," Revision 1, August 29, 1997.

Point Beach Nuclear Plant Units 1 and 2 Probabilistic Safety Assessment, Wisconsin Electric Power Company, June 1993.

SCIENTECH-PB-97-01, "Point Beach Nuclear Plant Re-baseline of the PSA Risk Significance Determinations for Maintenance Rule Using the Enhanced PSA93 Model," Revision 1, November 11, 1997.

SCIENTECH-PB-97-02, "Point Beach Nuclear Plant Use of PSA93 PRA Model to Determine Component and System Risk Significance for the Maintenance Rule," Revision 1, November 11, 1997.

NPC 95-13343, (A)(1) Systems

NPC 96-02176, "System Performance Criteria for Maintenance Rule"

NPC 96-03347, "Maintenance Rule Implementation"

NPC 96-07589, "Risk Significant Systems"

NPM 96-0075, "Expert Panel Review of PBNP Systems for Maintenance Rule Condition Assessment"

NPM 96-0076, "Summary of Expert Panel Meeting Results to Establish Maintenance Rule Performance Criteria"

NPM 97-0116, "Annual Report for Maintenance Rule 10 CFR 50.65"

NPM 97-0396, 'Maintenance Rule Risk Significant Expert Panel Minutes"

NPM 97-0802, "Structural Inspection Program Annual Report Point Beach Nuclear Plant"

PBM 96-0300, "Meeting Notes from Expert Panel Discussion of Risk Significance"

PBM 96-0357, "EOP Review for the Maintenance Rule"

PBM 96-0385, "Maintenance Rule Implementation Summary"

PLM-93-055, "Methodology Document"

PLM-93-086, "Transmittal of Final System Screening Procedure"

PLM-93-100, "System and Structure Screening Procedure"

PLM-93-117, "Component Screening Procedure"

WEP-11-14847, "Procedure for Identifying Systems and Structures Important to License Renewal and Within the Scope of the Maintenance Rule"

WEP-11-14891, "Methodology for Identifying Systems, Structures, and Components Important to License Renewal, the Maintenance Rule, and Power Production"

WEP-12-15380, "Summary Report for Screening Systems and Structures Important to License Renewal and within the Scope of the Maintenance Rule"

WEP-12-15818, "Screening Results for Identifying Systems and Structures"

PBF-7029, "Performance Criteria Report from MRLIN"

NDE-751, "Structural Surveillance of Containments and General Structures," Revision 5, October 31, 1997.

NDE-754, "Visual Examination of (VT-3) Nuclear Power Plant Components," Revision 5, dated October 31, 1997