U.S. NUCLEAR REGULATORY COMMISSION
REGION III

Docket No.:	50-440
License No.:	NPF-58
Report No.:	50-440/96014(DRS)
Licensee:	Cleveland Electric Illuminating Company
Facility:	Perry Nuclear Power Plant
Location:	10 Center Road Perry, OH 44081
Dates:	November 4 - 8, 1996
Inspectors:	Anthony N. Fresco, Research Engineer, BNL Martin J. Farber, Reactor Inspector, RIII (Team Leader) Charles H. Brown, Reactor Inspector, RIII Frank X. Talbot, Reactor Operations Engineer, NRR Tirupataiah Tella, Reactor Inspector, RIII Ronald A. Langstaff, Reactor Inspector, RIII
Approved by:	Wayne J. Kropp, Chief Engineering Specialists Branch 1 Division of Reactor Safety

9702050124 970129
PDR ADOCK 05000440
Q PDR

EXECUTIVE SUMMARY

Perry Nuclear Power Plant, Unit 1
NRC Inspection Report 50-440/96014(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 one week on-site inspection by regional and NRR inspectors.

Operations

- The team concluded that operators' knowledge of the maintenance rule (MR) was limited but 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

- The licensee generally identified the structures, systems, and components (SSCs) that were required to be within the scope of the maintenance rule. One exception, the communications system, was not included in the scope of the rule as required. This was considered a violation of the maintenance rule, but is not being cited because corrective action was in progress to put the system in scope.

- The composition of the expert panel, and the qualifications and experience of the panel members were considered to be appropriate. The panel members' Probabilistic Risk Assessment (PRA) knowledge appeared to be limited and they primarily depended on input from the PRA member. The expert panel was involved in several aspects of maintenance rule implementation, beyond those specified in NUMARC 93-01; this was considered a strength.

- Although the Expert Panel had added at least one additional system to the list of risk significant systems, overall, the risk determination process was weak. There was inconsistent implementation of the NUMARC 93-01 guidance in this area, and there appeared to be a potential for improper risk classification for some SSCs.

  This was exemplified by the inadequate discussion in the meeting minutes regarding the Expert Panel's exclusion from the list of risk significant systems, some apparently risk significant functions of Turbine Building Closed Cooling Water (TBCCW), the condensate and feedwater systems (other than the motor-driven feedpump), Control Rod Drive (CRD) pumps, and portions of the containment systems. Lack of appropriate justification for classifications and potential misclassification of functions was considered an unresolved item.

- The procedures for performing periodic evaluations met the requirements of the rule and the intent of the NUMARC implementing guidance. The team noted that the reports were comprehensive and would be a useful management tool for evaluating the effectiveness of maintenance. With the exception of the balance between reliability and availability, the reports contained all of the required evaluations.

- The licensee's process for conducting the reliability and availability balance was acceptable. The licensee was upgrading this process to perform the evaluation quantitatively, rather than qualitatively.

- Several weaknesses were identified in the licensee's program for on-line maintenance risk assessments. As noted in Section M1.2 the PRA was outdated. Lack of PRA staff involvement in the scheduling process, minimal knowledge levels of operators and scheduling staff with regard to PRA and maintenance rule requirements for risk assessment, and a weak risk assessment matrix for various unanalyzed configurations all contributed to a weak program. An occurrence of an unanalyzed configuration, although a risk evaluation was required by licensee procedures, was considered a violation of plant procedures.

- The licensee's program for collecting, distributing, and analyzing Industry Operating Experience (IOE) appears to be effective. While the IOE program and the MR program are properly linked procedurally, the Responsible System Engineer's (RSE) generally weak knowledge of the MR could impact the effectiveness of this linkage in practice.

- The licensee had established appropriate unavailability criteria for risk significant and standby systems and was monitoring performance on a cycle basis. The licensee chose to use functional failures per cycle as the primary reliability criteria and was attempting to correlate the criteria with the PRA. This was considered an unresolved item. Goals for (a)(1) systems were properly established and condition monitoring was appropriate.

- The licensee had not established an appropriate and defined criteria for when to transfer a structure from category (a)(2) to category (a)(1) for monitoring. An inspection follow-up item was identified pending review of the licensee's structure monitoring program.

- In general, the material condition of the systems examined was acceptable.

Quality Assurance (QA)

- Assessment activities were useful in identifying enhancements for expediting maintenance. However, the assessments did not evaluate the safety and risk impact of the maintenance activities reviewed.

Engineering

- System engineers were knowledgeable regarding their assigned systems and typically very involved in the resolution of issues and deficiencies. In most cases, system engineers' knowledge of the MR was commensurate with their involvement in MR activities. However, the team identified one example of a functional failure (FF) misclassification which may have been due to weaknesses in training. Considering the scope of maintenance rule responsibilities currently assigned to system engineers by station management, the team concluded that their knowledge level was adequate.

Report Details

Summary of Plant Status

The plant was operating at full power during the inspection.

Introduction

This inspection examined the licensee's implementation of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The report covers a one week on-site inspection by regional and NRR inspectors.

I. Operations

04 Operator Knowledge and Performance (62706, 71707)

04.1 Operator Knowledge of Maintenance Rule (MR)

a. Inspection Scope

During the inspection of the implementation of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," the team interviewed a licensed reactor operator and three senior reactor operators to determine if they understood the general requirements of the rule, their particular duties, and responsibilities for its implementation.

b. Observations and Findings

The team found that the operators understood their MR responsibilities in the control room with regard to tracking of unavailability. The Shift Technical Advisor (STA) was responsible for tracking the unavailability time for SSCs within the scope of the MR. SSCs were listed in the procedure for the STA. The Reactor Operator (RO) assisted the STA by logging the time of the unavailability for the SSCs. The RO also logged the out-of-service and inoperable time as the two were not counted by the same criteria. Unavailability was not logged against the SSC if the time was less than 10 minutes. The operators were less informed with regard to risk assessments for equipment out-of-service during power operation. The operators depended on the scheduling group to evaluate the risk involved for the maintenance of equipment, but did not let the work start if the initial conditions had changed due to emergent work. The operators, in general, were uncertain about the general requirements and the terminology of the MR.

c. Conclusions

The operators understood the specific MR requirements for the control room, but were weak in general understanding of the MR.

II. Maintenance

M1 Conduct of Maintenance (62706)

The primary focus of the inspection was to verify that the licensee had implemented a maintenance monitoring program which satisfied the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of the Maintenance at Nuclear Power Plants," (the maintenance rule). The inspection was performed by a team of four regional inspectors and one contractor. Assistance and support were provided by one member of the Quality Assurance and Maintenance Branch, NRR.

M1.1 SSCs Included Within the Scope of the Rule

a. Inspection Scope

The team reviewed the licensee's scoping documentation to determine if the appropriate SSCs were included within their maintenance rule program in accordance with 10 CFR 50.65(b). The team used inspection procedure 62706, NUMARC 93-01, and Regulatory Guide 1.160 as references during the inspection.

b. Observations and Findings

The licensee's maintenance rule program was described in Plant Administrative Procedure 1125, "Monitoring the Effectiveness of Maintenance Program Plan," Revision 2 (PAP-1125). This program described the methodology used to select the SSCs under the maintenance rule. The scoping criteria included whether the systems were: safety related, non-safety related systems that were relied upon to mitigate accidents or transients or were used in EOPs, systems whose failure could prevent safety-related SSCs from fulfilling their safety-related function, or systems whose failure could cause a reactor scram or actuation of a safety-related system. Based on the results of these evaluations, lists were developed of systems within the scope of the maintenance rule.

In general, the scoping of SSCs was good. The team reviewed the licensee's scoping documentation and determined that adequate justification for classification was available. The licensee initially considered about 233 SSCs in the scoping phase. Of these, 141 SSCs were placed within the scope of the maintenance rule, and 16 SSCs were placed in a(1) category. The team, however, noted that several non-safety-related SSCs were added to the scope as late as October 31, 1996. The team noted that the licensee's scoping of structures, included as system S-99, was good. This system included not only the plant buildings, but also pump houses, cooling towers, cranes, cable trays, and conduits. However, the licensee did not include the communications system in the scope. Communications were referred to in the Perry USAR, and were necessary to perform plant operations to maintain the plant in a safe condition following an accident.

The licensee's failure to include communications in the scope under the maintenance rule was a violation of 10 CFR 50.65 (b). However, the licensee had issued a Potential Issue Form (PIF) No.96-3396 on November 1, 1996, to include

communications in the scope of the maintenance rule. This licensee identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII of the NRC Enforcement Policy (NCV 50-440/96014-01(DRS)).

c. Conclusions

The team concluded that the licensee had identified most of the SSCs that were required to be within the scope of the maintenance rule. The team noted the licensee's scoping process was a properly structured program, but the implementation was not thorough. The licensee did not include communications in the scope of the rule. The team noted that the scoping decisions were well documented.

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

a. Inspection Scope

Paragraph (a)(1) of the rule requires that goals be commensurate with safety. Additionally, implementation of the rule using the guidance contained in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," requires that safety be taken into account when setting performance criteria and monitoring under paragraph (a)(2) of the rule. This safety consideration would be used to determine if an SSC should be monitored at the system, train, or plant level. The team reviewed the methods and calculations that the licensee established for making these risk determinations. The team also reviewed the risk determinations that were made for the specific SSCs reviewed during this inspection. NUMARC 93-01 recommended the use of an expert panel to establish safety significance of SSCs by combining Probabilistic Risk Assessment (PRA) insights with operations and maintenance experience, and to compensate for the limitations of PRA modeling and importance measures. The team reviewed the composition of the expert panel, and the experience and qualifications of its members. The team reviewed the licensee's expert panel process and the information available which documented the decisions made by the expert panel. The team interviewed several members of the expert panel to determine their knowledge of the maintenance rule and to understand the functioning of the panel.

b.1 Observations and Findings on the Expert Panel

The licensee used an expert panel process in conjunction with a PRA ranking methodology to determine the safety significance of SSCs within the scope of the maintenance rule. The expert panel consisted of seven voting members and three non-voting members. The expert panel met at least once weekly and the quorum required was four voting members. The expert panel included plant personnel experienced in engineering, PRA, operations, maintenance, and work scheduling. Most of the panel members had a degree in engineering or an SRO license and had between 9 and 17 years of nuclear power plant experience. The team noted that the members of the panel demonstrated a general knowledge of the maintenance rule and the functions of the expert panel. However, the team noted that there was no documentation, prior to October 1996, of training given to the members of

the expert panel on the maintenance rule or PRA. The team noted that the PRA knowledge of non-PRA members was limited; they heavily relied on the recommendations of the PRA member. The lack of appropriate PRA knowledge of the expert panel members was considered a weakness which could have contributed to the earlier scoping deficiencies. Also, as noted in Section M1.1 of this report, communications, though required to be included in the scope, were not included.

The expert panel reviewed the PRA results and approved the risk significant, in-scope, and out-of-scope SSCs. The team noted that the expert panel remained involved in some aspects of maintenance rule implementation, beyond those specified in NUMARC 93-01, such as:

- Review and approval of system performance criteria
- Review and approval of maintenance rule periodic reports.

The team noted that the minutes of the expert panel meetings were excellent. Though the decisions of the panel were generally by consensus, differing opinions were documented. The team considered this a strength.

c.1 Conclusions on Expert Panel

The team considered that the expert panel's composition, qualifications, and experience were appropriate. The team noted that training in MR or PRA given to the expert panel members prior to October 1996 was not documented. The team concluded that the expert panel non-PRA members' knowledge of PRA was very limited. The team noted that the panel generally functioned acceptably, although early omission of some SSCs from the scope of the maintenance rule and the potential misclassification of some risk significant functions (discussed in Section b.2.2) were weaknesses. The team also considered the panel's excellent documentation of the meeting minutes a strength.

b.2 Observations and Findings on Risk Determinations

b.2.1 Analytical Risk Determining Methodology

The Individual Plant Examination (IPE) was based on NUS' NUPRA code (version not identified) and was transmitted to the NRC in July 1992. The transmittal document was available to the plant staff. The document included the event trees used for the modeling, but not the fault trees in order to reduce the material to one volume. In general, all of the Level 1 event trees were Perry-specific, especially the Anticipated Transient Without Scram and the Loss of Offsite Power event trees. The transition trees between the Level 1 PRA and the Level 2 PRA were also Perry-specific. However, the Plant Damage States in the Level 2 PRA were based largely on the Grand Gulf model as presented in NUREG/CR-4550. The Perry containment design consists of a free standing steel containment structure, which was significantly different from that of Grand Gulf.

Since the time when the IPE was generated, the licensee maintained the PRA model with occasional revisions based on plant modifications. As discussed below, the

data have not been updated since the time when the IPE was transmitted to the NRC, i.e., July 1992. At the time of this inspection, the PRA was a calculation with the hard copy paper originals and supporting information maintained in two file cabinet drawers. The PRA computer model was available to the plant's PRA engineer on the plant's computer network. No controlled copies, similar to FSARs or USARs, were available to the remainder of the plant staff. Consequently, accessibility to the PRA for those plant staff not intimately involved in the PRA process but whose job functions may have significant impact on the accuracy of the plant's PRA, and hence on the Maintenance Rule implementation, was limited. Such staff members included Systems Engineers, Design Engineers, Planners and Schedulers, and Operations, Maintenance, and I&C staff engineers. It was also a contributing factor to the nearly consistent response of key individuals involved in implementing the Maintenance Rule, such as the Operations staff involved in planning and scheduling and the systems engineers, that they had very little knowledge or understanding of PRA in general, of the plant's specific PRA, or of the Maintenance Rule requirements. There was only one plant staff engineer actively involved in the PRA effort. This engineer was the only person who could calculate the plant's risk of core damage for emergent work situations The combination of these factors resulted in a weakness in the licensee's risk assessment program.

As part of the plant modification program, the licensee had in place procedure NEI-0330, "Interface Reviews and Evaluations," Revision 3. In Attachment 1 to this procedure were Checklists 25.0 "Probabilistic Risk Assessment" and 29.0 "Maintenance Rule, 10 CFR 50.65." Each checklist posed a series of questions which were worded for "Yes/No" answers. The procedure did not specify whether a "Yes" answer for 29.0 automatically required a review of the design modification for impact on the PRA, as detailed under 25.0. While not likely, it was possible that a "Yes" answer to 29.0 for Maintenance Rule impact might not have resulted in review for PRA impact if the answers for Checklist 25.0 were "No," and vice versa. The procedure would have been more effective if there had been a clear link between the two checklists. There was no written procedure or guidance document which defined the criteria to be used to determine whether a modification should result in a PRA update. The criteria for deciding whether a modification was significant were at the discretion of the PRA engineer. The team viewed these conditions as weaknesses in the PRA program.

The data in the PRA for the basic events was based on generic failure data that were developed from EPRI databases, NUREGs and NUREG/CRs, scrams occurring in other BWR 6 plants as documented in LERs, and Perry-specific plant history of events occurring between 11% - 100% power operation. The licensee did use Perry-specific data for the equipment unavailabilities. The historical review period for the Perry-specific data covered a 3.5 year period from the late 1980's to the early 1990s. The compilation of data was initiated in 1990 for the IPE effort and ended with the transmittal to the NRC of the Perry Individual Plant Examination (IPE) results in July 1992. In March 1996, the licensee calculated the importance measures for the maintenance rule, i.e., the Risk Achievement Worth, Risk Reduction Worth, and 90% of Core Damage Frequency (CDF) cutsets, using the same data that were the basis of the IPE. The team considered the lack of current reliability and availability data as a weakness in the PRA program.

b.2.2 Adequacy of Expert Panel Evaluations

In addition to determining which SSCs were within the scope of the rule, the licensee's expert panel established the risk significance ranking of SSCs, the performance criteria for SSCs, the goals for SSCs, and the lists of (a)(1) and (a)(2) SSCs. The licensee established the expert panel in accordance with Section 9.3.1 of NUMARC 93-01.

The final risk significance ranking was based on a combination of results from the licensee's PRA and expert panel judgement based on deterministic considerations.

To determine the risk significance of SSCs from the perspective of a Level 1 PRA (frequency of core damage), the licensee considered the risk importance measures identified in NUMARC 93-01. The licensee considered an SSC to be risk significant if it met any 2 out of 3 of the importance measures' quantitative limits:

- Its risk achievement worth (RAW) was ≥ 2,
- Its risk reduction worth (RRW) was > 1.005, or
- It appeared in the cutsets contributing to the top 90% of Core Damage Frequency.

NUMARC 93-01, Revision 0, page 18, stated that each of these methods would identify a different set of SSCs based upon differing concepts of importance. Each method was useful in providing insights into risk significant SSC selection, and consideration should be given to using all of them in the decision making process. NUMARC 93-01 indicated that it was equally important to identify as risk significant, those SSCs that prevent containment failure or bypass that could result in an unacceptable release. Examples might include the containment spray system, containment cooling system, and valves that provide the boundary between the reactor coolant system and low pressure systems located outside containment.

For the Level 2 analysis, the licensee considered as an indication of importance a $\pm 5\%$ change in the frequency of reactor pressure vessel (RPV) failure, containment failure, or containment bypass.

At the time of the inspection, based on the "Periodic Assessment Report of Maintenance Effectiveness for (Operating) Cycle 5 (August 14, 1994 - April 10, 1996)," out of 233 engineered systems, 141 systems were considered to be within the scope of the Maintenance Rule. Those 141 systems were classified into the Maintenance Rule performance monitoring regime of 145 Systems and Component Groups. Thirty-eight (38) systems were monitored strictly by Plant Level Performance Criteria, and 67 Systems and Components were monitored by Specific Performance criteria.

Of the 141 systems in the scope of the MR, 95 were considered to be less risk significant. Of the remaining 46 systems, 8 were considered to be risk significant systems (RSS), 16 were risk significant and standby, and 22 were standby systems. Thirty-eight (38) trains and systems were modeled in the PRA.

With respect to the inclusion or exclusion of SSCs by the expert panel, the following were noted:

- Although the Low Pressure Core Spray (LPCS) System did not meet any of the 3 importance measure criteria, the expert panel declared it to be a Risk Significant System. This was considered a strength

- The Turbine Building Closed Cooling Water (TBCCW) System, P44, met the criteria for inclusion in the Top 90% of Cutsets, but not in the Risk Achievement Worth (RAW) or Risk Reduction Worth (RRW) importance measures. Consequently, the licensee considered the TBCCW System, P44, not to be risk significant. The licensee's expert panel Meeting Minutes #50, May 6, 1996, stated that the systems ranging from E31 to T23 were reviewed and approved by consensus, with certain exceptions that were identified. There was no special discussion for system P44. Although not specifically stated in the NUMARC 93-01 guidance or in Regulatory Guide 1.160, it is the NRC's position that an SSC should be included as high risk significant if it meets any one of the three importance measures of RAW, RRW, or top 90% of cutsets, unless adequately justified by the expert panel. Given the inadequate discussion evident in the meeting minutes, the exclusion of the TBCCW System by the expert panel is considered a weakness.

- In Meeting #50 of the expert panel, May 6, 1996, the expert panel requested that the risk significant bases explain why the PRA did not conclude that the condensate and feedwater system pumps (other than the motor feedpump) were important. It was agreed to address this as part of the bases' documentation. It was also agreed that the bases should also consider the importance of the CRD purge flow to the RPV water level reference legs to ensure level flashing does not occur during depressurization, and noted that this concern was addressed in PAP-0201, "Conduct of Operations." No further documentation was provided to the team to address the condensate and feedwater issues or the reference leg fill function of the CRD pumps not being included in the RSS ranking.

- In Calculation 6.17, Revision 0, dated June 21, 1996, "Determination of Risk Significant systems from the Level II perspective," there was a table, "Backend PRA Risk Significance Reference Information," which listed various containment systems and the corresponding paragraph in the IPE which discussed that system. This table indicated that:

    - T23, "Containment Design," was not risk significant with respect to the Containment Failure Fragility Model.

    - B21, "Nuclear Boiler Instrumentation," was not considered risk significant with respect to RPV depressurization;

    - M56, "Hydrogen Control," was not considered risk significant with respect to Hydrogen Ignitor (No Recovery or Failure to Initiate and Failure to Recover);

- E22, "Residual Heat Removal" (RHR) was not considered risk significant with respect to RHR Spray Operation.

According to the licensee's table, "Risk Significant Functions and Basis (PRA or EP for expert panel)," dated November 5, 1996:

- T23-01, -04, -05, -06, and -08 were not listed;
- T23-02, Primary Containment power-operated isolation valves, automatic PCIVs and MSIVs were RSS per the expert panel
- T23-03, Primary Containment Venting, was RSS per the PRA.

In the "Maintenance Rule System Report - Cycle 5 (August 14, 1994 - April 10, 1995) Super System - System: T23 Containment," dated September 9, 1996, table entitled "Maintenance Rule Roll-Up and Transfer Functions":

- T23-01 was Primary Containment Integrity,
- T23-04 was Drywell integrity - drywell concrete structure, drywell head, penetrations, and penetration isolation devices (valves, flanges, air lock and equipment hatch),
- T23-05 was Drywell power operated and automatic isolation valves,
- T23-06 was Secondary Containment integrity,
- T23-08 was Containment pressure isolation valves in the Inservice Testing Program.

It appeared that the above T23 systems, i.e., T23-01, T23-04, T23-05, T23-06, and T23-08, were not considered to be RSS. However, the licensee did not provide any meeting minutes from the expert panel deliberations which would indicate that the expert panel deliberated and considered these portions of T23 to be not risk significant. Similarly, no meeting minutes were provided to the inspection team to indicate that the expert panel had deliberated specifically concerning B21 and M56, other than the broad statement in Meeting Minutes #50 that systems ranging from E31 to T23 were reviewed and approved by consensus and in Meeting Minutes # 48, April 29, 1996, that the panel reviewed B13 to E22R systems.

c.2 Conclusions on Risk Determinations

Although the expert panel had added at least one additional system to the list of risk significant systems, overall, the risk determination process was weak. There was inconsistent implementation of the NUMARC 93-01 guidance in this area, and there appeared to be a potential for improper risk classification for some SSCs.

This was exemplified by the inadequate discussion in the meeting minutes regarding the expert panel's exclusion from the list of risk significant systems, some apparently risk significant functions of TBCCW, the condensate and feedwater systems (other than the motor-driven feedpump), CRD pumps, and portions of the containment systems. Lack of appropriate justification for classifications and potential misclassification of functions was considered an unresolved item (50-440/96014-02(DRS)).

M1.3 (a)(3) Periodic Evaluations (62706)

a. Inspection Scope

Section (a)(3) of the 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 is required to be performed at least one time during each refueling cycle, not to exceed 24 months between evaluations. The team reviewed the procedural guidelines for these evaluations and the Cycle 5 assessment which was completed in April 1996.

b. Observations and Findings

The licensee's instructions for conducting periodic evaluations were contained in the following documents:

- PAP-1125
- Technical Administrative Instruction (TAI) - 0510, "Monitoring the Effectiveness of Maintenance - System Review," Revision 2
- TAI-0511, "Monitoring the Effectiveness of Maintenance - Periodic Assessment Report," Revision 1

These procedures provided appropriate guidance for preparing evaluations which would meet the requirements of 10 CFR 50.65 (a)(3) and the intent of NUMARC 93-01.

The team reviewed the following assessments:

- Periodic Assessment Report of Maintenance Effectiveness for the Period June 1992 to June 1995
- Periodic Assessment Report of Maintenance Effectiveness for Operating Cycle 5
- Periodic Assessment Report of Maintenance Effectiveness for Operating Cycle 6 (report in-process)

The assessments were comprehensive and well-organized. The team noted that neither of the two completed assessments contained a reliability and availability balance. (This is discussed further in Section M1.4.) All of the remaining aspects of the maintenance rule program were adequately addressed.

c. Conclusions

The procedures for performing periodic evaluations met the requirements of the rule and the intent of the NUMARC implementing guidance. The team noted that the reports were comprehensive and would be a useful management tool for evaluating the effectiveness of maintenance. With the exception of the balance between reliability and availability, the reports contained all of the required evaluations.

M1.4 (a)(3) Balancing Reliability and Unavailability (62706)

a. Inspection Scope

Paragraph (a)(3) of the rule requires that adjustments be made where necessary to assure that the objective of preventing failures through the performance of preventive maintenance is appropriately balanced against the objective of minimizing unavailability due to monitoring or preventive maintenance. The team reviewed the licensee's plans to ensure this evaluation was performed as required by the rule. The team also discussed these plans with the Maintenance Rule Project Engineer (MRPE) who was responsible for preparing this evaluation.

b. Observations and Findings

The licensee's instructions for conducting reliability and availability balances were contained in PAP-1125. The procedure provided adequate guidance for preparing evaluations which would meet the requirements of 10 CFR 50.65 (a)(3) and the intent of NUMARC 93-01. Reviewing the completed Cycle 5 assessment revealed that the balance had not been evaluated. Since this assessment was completed prior to July 10, 1996, this was allowable. The licensee was preparing a procedure to evaluate this balance quantitatively. This was an improvement over currently accepted methodology which provided qualitative evaluations. The licensee staff stated that the Cycle 6 assessment will contain the required reliability and availability balance. Review of this procedure, when completed, and review of the Cycle 6 evaluation is an Inspection Followup Item (IFI 50-440/96014-03(DRS)).

c. Conclusions

The licensee's process for conducting the reliability and availability balance was acceptable. The licensee was upgrading this process to perform the evaluation quantitatively, rather than qualitatively.

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

a. Inspection Scope

Paragraph (a)(3) of the Maintenance Rule states that the total impact of maintenance activities on plant safety should be taken into account before taking equipment out of service for monitoring or preventive maintenance. The inspectors reviewed the licensee's procedures and discussed the process with the MRPE, the reliability engineer performing PRA risk assessments, the expert panel members, the plant operators, system schedulers, and work week supervisors.

b. Observations and Findings

b.1 Procedural Guidance/"Risk Matrix"

The licensee's program for assessing the risk significance of on-line maintenance was contained in PAP-0130, "On-Line Schedule Development," Revision 1. Attachment 5 to this procedure was "Risk Matrix for Selected Equipment/System Combinations." Attachment 6 was "Risk Categories."

The "Risk Matrix" provided in PAP-0130, Attachment 5, contained 48 pre-analyzed configurations. For configurations identified by Attachment 6, "Risk Categories," as falling into Risk Category 1, the On Line Scheduling Unit (OLSU) Supervisor was to identify compensatory actions on the schedule. For Risk Category 2, an evaluation was to be performed to determine if compensatory actions were required. For Risk Category 3, no compensatory actions were required. For any activity not covered in Attachment 5, the OLSU Scheduler was to identify that an evaluation by Engineering was required. The OLSU was required to identify in the schedule any risk or compensatory actions recommended by Engineering.

Since the maintenance rule has been in effect, the team found that the licensee has taken equipment out of service without performing specific risk assessments for various unanalyzed configurations. Examples were:

- On August 7, 1996, the licensee's reliability engineer performed a risk assessment for a scheduled Division 1 outage which included removing from service residual heat removal (RHR) train A, the low pressure core spray system (LPCS), emergency closed cooling (ECC) Train A, emergency service water (ESW) Train A and the division 1 emergency diesel generator (EDG). This pre-analyzed configuration gave an estimated CDF of 7.7E-5 per year. The licensee considered this to be a risk category 3 configuration which was below the NRC CDF safety goal of 1.0E-4.

  The team found that on August 20 and 21, 1996, with the plant at power, the licensee performed a Division 1 outage which removed RHR train A, the LPCS, ECC system train A, ESW train A, the Division 1 EDG, the control rod drive (CRD) pump (train B), and the reactor core isolation cooling (RCIC) system.

  The configuration above was not analyzed, as required by PAP-0130, for the addition of a CRD pump and the RCIC pump out-of-service due to emergent work. Both the CRD pump and the RCIC pump provided coolant to the core under Station Blackout (SBO) scenarios; therefore, the CDF estimate above was not accurate. This configuration could have been a risk category 2 configuration requiring compensatory actions. Failure to perform the required evaluation is a violation (VIO 50-440/96014-04(DRS)).

- The team noted that the PAP-0130, Attachment 5, risk matrix contained a very limited number of pre-analyzed configurations. For example, on October 3 through October 4, 1996, with the plant at power, the licensee removed a diesel driven fire water pump and standby liquid control system

(SLC) train A from service for a total combined time of eleven and half hours. The matrix contained no pre-analyzed risk assessment for this configuration.

The licensee provided no evidence as to whether the need for an engineering evaluation had been considered, that an engineering evaluation actually had been requested by the OLSU Scheduler, or that Engineering had performed an engineering evaluation.

Since other risk significant equipment, or other equipment which was within the scope of the maintenance rule but was not considered risk significant, was not specifically identified in the procedure, plant operations and scheduling personnel depended on a "catch-all" statement at the bottom of Attachment 6, "Risk Categories," to request a safety assessment for scheduling activities which took equipment out of service for maintenance. Unanalyzed combinations of risk significant SSCs not identified in the risk matrix, or combinations of low risk SSCs removed from service during plant operation at power, could place the plant in a higher than acceptable risk-significant configuration. Based on the fact that unanalyzed conditions have occurred, the team concluded that the minimal number of pre-analyzed configurations on Attachment 5 constituted a significant weakness in the licensee's program for risk assessment of on-line maintenance.

b.2 PRA Staff Involvement in Scheduling

The team noted that the PRA engineer was routinely only involved in evaluating the divisional outage schedule approximately one week in advance. The PRA Engineer was not consistently involved in emergent work situations, nor did the PRA Engineer normally attend scheduling meetings. Nevertheless, the scheduling process relies heavily on only two people, the reliability (PRA) engineer and the Senior Operations Scheduler to determine risk significance.

b.3 Schedulers/Operators Knowledge and Training

The team interviewed a Unit Supervisor and a Shift Supervisor concerning their knowledge of the MR, specifically as it applied to procedure PAP-0130 and equipment out-of-service during power operation. The team found that there was generally weak knowledge of PRA or MR requirements with respect to consideration of balance-of-plant equipment, which was not controlled by the technical specifications, as significant to risk. In addition, the team found that training of licensed reactor operators (ROs) was limited to reading and signing that they understood procedure PAP-0130 (and its attachments 5 and 6).

c. Conclusions

Several weaknesses were identified in the licensee's program for on-line maintenance risk assessments. As noted in Section M1.2 the PRA was outdated. Lack of PRA staff involvement in the scheduling process, minimal knowledge levels of operators and scheduling staff with regard to PRA and maintenance rule requirements for risk assessment, and a weak risk assessment matrix all contributed to a weak program. The occurrence of an unanalyzed configuration,

although a risk evaluation was required by licensee procedures, was considered a violation.

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 rule. The team also discussed the program with appropriate plant personnel. The team reviewed the following systems:

(a)(1) systems

Reactor Core Isolation Cooling
Reactor Recirculation
Power Transformers
DC Systems

(a)(2) systems

Divisions 1 and 2 Standby Diesel Generators
Miscellaneous Structures
Control Room Emergency Recirculation
Control Rod Drive
Essential Service Water

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 (MPFF). The team also reviewed performance criteria for SSCs not listed above.

The team reviewed the licensee's process to evaluate onsite passive structures for inclusion under the MR. Structures evaluated by the team included buildings, enclosures, storage tanks, earthen structures, and passive components and materials housed in the aforementioned. In addition, the team assessed by what means performance of structures determined to be within scope was monitored for degradation.

b. Observations and Findings

b.1 Performance Criteria for Unavailability

The licensee had established fractional unavailability criteria for their high risk significant systems. For each cycle, the criteria were translated into a specified number of hours for each affected system, depending upon the expected length of

the fuel cycle. The MRPE stated that the hours would be adjusted if length of the fuel cycle changed.

Unavailability was monitored on a per cycle basis rather than a rolling average. In practice, the RSE and the scheduling organization understood that they only had a fixed number of hours during which systems could be unavailable without exceeding performance criteria. As such, unavailability times were a major consideration for planning on-line maintenance.

The licensee's PRA engineer stated that some sensitivity studies had been performed to ensure that unavailability criteria were reasonable. The team did not identify any examples where the unavailability criteria for systems was not commensurate with safety.

b.2. Reliability Related Performance Criteria

The licensee had established reliability performance criteria of allowing only two FFs per fuel cycle for most systems. The criteria were monitored on a per cycle basis rather than a rolling average. In some cases, the licensee had established reliability criteria which spanned two fuel cycles. For example, the licensee established a reliability performance criterion of allowing only one FF per two fuel cycles for their divisional Topaz inverters.

The licensee identified (as documented on PIF 96-3396) that their reliability performance criteria needed to be justified as consistent with PRA assumptions for reliability. At the time of the inspection, the reliability performance criteria for most systems were specified without regard to the number of demands per cycle nor PRA reliability assumptions.

As part of their corrective actions, the licensee was in the process of reviewing the performance criteria in relation to the PRA reliability assumptions. The inspectors reviewed a draft calculation, Calculation 6.20, which proposed that the performance criteria for most systems be revised to only allow one FF per fuel cycle. The use of one FF as performance criteria was consistent with suggested performance criteria presented at an October 16, 1996 Nuclear Energy Institute (NEI) Maintenance Rule Workshop. Although the proposed performance criteria would be an improvement, the team concluded that significant limitations would still exist.

The calculation proposed performance criteria such that, given the reliability assumed in the PRA, there would be less than 10% probability that the performance criteria would be exceeded. The team noted that the objective was to establish performance criteria such that the criteria would not be routinely exceeded rather than to ensure that the criteria were commensurate with safety. For example, the calculation proposed to set a performance criterion of one FF per fuel cycle for the Divisions 1 and 2 diesel generator building ventilation systems. The licensee identified these systems as having a minimum of 18 demands per fuel cycle and a PRA assumed failure probability of 1.2E-4 per demand (99.99% reliability). The team performed calculations using a binomial distribution (consistent with the licensee's calculation methodology) and noted that a system

with a statistical failure rate of 3.0E-2 per demand (97% reliability) would have a 90% chance of meeting the performance criteria and that a system with an actual failure rate of 9.2E-2 per demand (91% reliability) would have a 50% chance of meeting the performance criteria. Given that these systems could exhibit failure rates two orders of magnitude greater than that assumed in the PRA and still have a good chance of meeting the performance criteria, the team concluded that the performance criteria would not ensure that the PRA assumed reliability was being maintained. In this instance, the licensee failed to demonstrate that reliability, as monitored using MR performance criteria, would be acceptable in terms of safety.

In addition to the diesel generator building ventilation systems, the team identified a number of other systems for which the proposed performance criteria did not appear to be commensurate with safety. Some examples were:

System	Calculation Proposed Performance Criteria		PRA Assumed Failure Rate	Statistical Failure Rate Having Chance of Passing Under Proposed Performance Criteria	
	FF's Allowed	Minimum Demands per Cycle	Probability Failure/ Demand	Failure/Demand With 50% Chance	Failure/Demand With 90% Chance
E12, RHR C	1	6	9.794E-3	2.664E-1	9.926E-2
M39, ECCS Pmp Rm Cooling, RCIC	1	6	4.999E-4	2.664E-1	9.926E-2
M43, Div's 1& 2 D/G Bldg Ventilation	1	18	1.238E-4	9.151E-2	2.995E-2
P45, ESW C	1	18	7.044E-3	9.151E-2	2.995E-2

The team acknowledged the difficulty of establishing performance criteria which were commensurate with safety when FFs were used as a monitored parameter. The difficulties were especially apparent for systems having a very high reliability assumed in the PRA and for which there were a low number of demands during the period being monitored.

At the time of the inspection, the licensee was still in the process of reviewing their performance criteria and the calculational methods used to support those criteria. The PRA engineer stated that the number of demands specified in the draft calculation reviewed by the inspectors was based on surveillance frequencies for the specified systems. Demands due to routine operation and surveillances on other systems were not counted. Consequently, the actual number of demands per cycle would be significantly greater than that reflected in their draft calculation for many systems. The team recognized that the additional demands considered may provide a sufficient sample size upon which to judge reliability.

The issue of establishing reliability performance criteria commensurate with safety is an Unresolved Item pending review of the licensee's revised performance criteria and associated justifications (URI 50-440/96014-05(DRS)).

b.3 (a)(1) Goals and Performance Monitoring

The team examined the goals selected for category (a)(1) systems and noted that they were generally extensions or refinements of the systems' performance criteria. With the exception of reliability goals based on FFs (which are subject to the same concerns as reliability criteria based on FFs), the licensee's goals were appropriate and conservative. Reviewing the bases used for transferring systems from (a)(2) to (a)(1) and back revealed that the licensee was monitoring these systems on a frequent basis. This was most evident in the treatment of the Control Room Emergency Recirculation Ventilation System, the CRD system, and the Power Transformers.

b.4 Structures and Structure Monitoring

The licensee divided structures into two categories: safety related and non-safety related. Both categories used no FFs per fuel cycle as one of the performance criteria. For structures, the licensee defined a FF as a condition which caused a failure of a safety-related function or of a plant level performance criteria. In addition, the licensee specified an allowed number of "structural component failures" for each category: 25 for safety related and 3 for non-safety related per fuel cycle. The MRPE stated that the number of structural component failures permitted for non-safety-related structures was set proportional to the number non-safety-related structures being evaluated under their MR program.

The licensee took credit for existing surveillances and operator rounds for their structural condition monitoring program. In addition, the licensee was planning to perform periodic structure walkdowns for monitoring under the Maintenance Rule. However, the licensee had not established a frequency for periodic structure walkdowns nor had they performed any structure walkdowns for monitoring under the Maintenance Rule at the time of this inspection.

The team noted that the performance criterion of no FFs provided no predictive value. The criterion would only ensure that a structure was placed into a(1) monitoring after a significant failure. Consequently, the no FF criterion, by itself, provided little safety benefit.

The team noted that the criteria based on monitoring structural component failures had the potential to be predictive in nature. However, the licensee had not defined what constituted a "structural component failure." The inspectors noted that for the review conducted of the previous operating cycle, the licensee had listed 15 conditions which would be considered structural component failures. In general, these conditions were ones where the condition required repair and could not be left as found. However, these conditions were not proceduralized into the structural monitoring program outlined in procedure TAI-0513 nor carried over in the current operating cycle evaluation. In addition, the team identified one example (an Intermediate Building roof leak documented by PIF 96-2977) where the condition was one of the 15 conditions previously cited as being a structural component failure. However, in this instance, the licensee determined that the condition identified was not a structural component failure. Consequently, the

team considered the criteria based on "structural component failures" ambiguous and to not define when monitoring under (a)(1) was necessary.

The licensee's procedure for monitoring structures, TAI-0513, did not provide adequate guidance for when a structure should be transferred from category (a)(2) to category (a)(1) monitoring under the Maintenance Rule. This is a generic, industry-wide issue and is an Inspection Follow-up Item pending review of appropriate and defined criteria established by the licensee for when to transfer a structure from category (a)(2) to category (a)(1) for monitoring (IFI 50-440/96014-06(DRS)).

c. Conclusions

The licensee had established appropriate unavailability criteria for risk significant and standby systems and was monitoring performance on a cycle basis. The licensee chose to use FFs per cycle as the primary reliability criteria and was attempting to correlate the criteria with the PRA. Goals for (a)(1) systems were properly established and condition monitoring was appropriate.

The licensee had not established an appropriate and defined criteria for when to transfer a structure from (a)(2) to (a)(1) classification for monitoring. An inspection follow-up item was identified pending review of the licensee's structure monitoring program.

M1.8 Use of Industry-wide Operating Experience

a. Inspection Scope

Paragraph (a)(1) of the rule states that goals shall be established commensurate with safety, where practical, taking into account industry-wide operating experience. Paragraph (a)(3) of the 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-wide operating experience. The team reviewed the licensee's program to integrate industry operating experience into their monitoring program for maintenance.

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

The inspector reviewed licensee procedures PAP 1607, "Industry Operating Experience," and PAP-1608, "Corrective Action Program," and discussed the procedures and usage with the Independent Safety Engineering Group (ISEG) member responsible for receipt and distribution of the industry operating experiences (IOE) from all the sources. The ISEG member reviewed the IOEs and sent them out to engineering supervisors who in turn divided them by systems and sent each system IOE to the appropriate RSE. The ISEG member could also have sent the IOE to multiple groups. Each IOE report had a cover letter stating what was to be covered in the response to ISEG and each IOE report was tracked. Neither ISEG nor the supervisor reviewed the IOE reports for FFs; however, the RSE was tasked with identifying FFs and determining if the failure was applicable to a

Perry system. The MRPE did not routinely receive copies of the IOEs. Because RSEs exhibited generally limited knowledge of the MR (discussed in Section E4.1), their ability to identify FFs effectively was questionable.

c.2 Conclusions for Use of Industry wide Operating Experience

The licensee's program for collecting, distributing, and analyzing IOEs appears to be effective. While the IOE program and the MR program are properly linked procedurally, the RSE's generally weak knowledge of the MR could impact the effectiveness of this linkage in practice.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 General System Review

a. Inspection Scope

The inspectors conducted a detailed examination of several systems from a maintenance rule perspective to assess the effectiveness of the licensee's program when it was applied to individual systems.

b.1 Observations and Findings for the Control Rod Drive System

This licensee classified this system as a low risk system, except for the scram function (Subsystem C11-01). The team noticed that several PIFs were generated for this system, indicating varied problems, including CRD pump failures. The team noted that the post-accident EOP functions of the CRD pumps to provide cooling water were not adequately described and these pump functions were not monitored. The licensee classified CRD purge flow (Subsystem C 11-10), used to maintain reactor vessel water level instrumentation as a less-risk significant function. This level instrumentation provided both RPS reactor trip signals and ESFAS actuation signals to start emergency core cooling pumps. The team also found that CRD purge flow was not modeled in the PRA; therefore, CRD purge flow would not be picked up by any importance measure as risk significant. Further, no justification for the expert panel's low risk significance determination was available.

c.1 Conclusions on Control Rod Drive System

The team determined that the licensee had not adequately evaluated the risk significance of the CRD purge flow. This unresolved item was discussed in Section M1.2.b.2.2. The team considered the material condition of the CRD system to be acceptable.

b.2 Observations on Control Room Emergency Ventilation Recirculation System

This safety-related, non-risk significant system was initially combined with the Control Room Ventilation System (M-25) and was placed in category (a)(1), due to multiple failures in the systems. These systems were changed to category (a)(2) in September 1996, as they met the goals and performance criteria during Cycle 5. The expert panel also decided in September 1996 to consider them as two separate

systems for the maintenance rule. The licensee set the performance criteria for these systems as "No more than two FFs per fuel cycle and no repetitive FFs." System M-25 already had two FFs in Cycle 6. PIF No. 96-3276 issued for the latest failure in this system was under investigation at the conclusion of this inspection. One failure was noted in System M-26 in Cycle 6. The licensee stated that both systems continued to be in category (a)(2).

c.2 Conclusions on Control Room Emergency Ventilation Recirculation System

The licensee established FF-based reliability performance criteria for the Control Room Ventilation systems. With two failures so far in cycle 6, issuance of a PIF to evaluate system M-25 performance was appropriate. Review of the corrective actions emerging from PIF 96-3276 is an inspection follow-up item (IFI 50-440/96014-07(DRS)).

b.3 Observations and Findings on DC Electrical Distribution System

This safety-related, risk significant system continued to be in a(1) category, as the system's unavailability of 1314 hours in Cycle 5 exceeded the performance criteria of 1016 hours. Except for minor leaks at the battery top cover (which did not affect the battery operability), these 125 volt batteries performed satisfactorily. Monthly and quarterly tests were conducted on these batteries and they were adequately trended. The team noted that the housekeeping and equipment conditions in these systems were satisfactory

c.3 Conclusions on DC Electrical Distribution System

The team concluded that adequate goals and performance criteria were set for the D.C. System and the system's performance was being satisfactorily monitored, per the maintenance rule.

b.4 Findings and Observations on the Reactor Core Isolation Cooling System

During inspection of the RCIC pump room, no excessive leakage was observed nor were any leaks not previously identified by the licensee found. Only three material deficiency tags were noted with the oldest dating back to March 1996. No material conditions problems were identified from visual inspection.

The licensee considered RCIC a high risk system and classified RCIC as an (a)(1) system due to the unavailability criteria being exceeded during the previous fuel cycle. Performance criteria had been established as 1.8% for unavailability and two FFs per fuel cycle for reliability. The licensee was in the process of reviewing the reliability criteria for RCIC and other systems (see Section M1.6.b.2).

c.4 Conclusions on the Reactor Core Isolation Cooling System

The team considered the RCIC system to be in good material condition. The system was being properly treated under the maintenance rule. Appropriate unavailability criteria had been established; when these criteria were exceeded, the

system was categorized as (a)(1), acceptable goals and corrective action were established, and monitoring was implemented.

b.5 Findings and Observations on Emergency Diesel Generators

The licensee considered the diesel generators to be a high risk system and classified the diesel generators as (a)(2) for monitoring purposes. Performance criteria had been established as 1.8% for unavailability and two FFs per fuel cycle. The team noted that the reliability criteria were different, but comparable, to other licensing commitments relating to the diesel generators.

During inspection of the Emergency Diesel Generator, only a few minor fuel and lube oil leaks were observed; all had been previously identified by the licensee. No material conditions problems were identified from visual inspection.

c.5 Conclusions on Emergency Diesel Generators

The team considered the EDGs to be in good material condition. The system was being properly treated under the maintenance rule.

b.6 Findings and Observations on the Reactor Recirculation System

Based on walkdowns of accessible components, the Reactor Recirculation System (RRS) appeared to be well maintained, and the various pieces of equipment were kept clean. There was no evidence of oil leaks on the hydraulic skids. The relay panel and the relays were clean and well maintained. The low frequency motor generator (LFMG) sets were clean and the vibration analysis performed on the "B" LFMG during the monthly surveillance indicated the set was satisfactory. The RRS had been placed in category (a)(1) because of FFs that occurred in 1992 and 1993. Corrective action was modifications to the loop RTDs and the pump seal packages. The system will be maintained in category (a)(1) for cycle 6 with a criterion of no power reductions due the modified components.

c.6 Conclusions on the Reactor Recirculation System

The team considered the RRS to be in acceptable material condition. The system was being properly treated under the maintenance rule.

b.7 Findings and Observations on the Emergency Service Water System

The Emergency Service Water (ESW) system appeared to be well-maintained; no significant deficiencies were identified during the walkdown. Divers were working in the ESW screen house, de-mudding the intake bays. After each dive the diving gear was washed off, which necessitated periodically cleaning the northern part of the screenhouse. Review of the PIFs written during 1996 revealed that a PIF (96-2159) had been written by the ESW system engineer in May 1996 regarding ESW Pump "A." ESW pump "A" had degraded to the point where Division 1 Diesel Generator (DG) jacket water heat exchanger flow was at the minimum permitted by the USAR, 1000GPM. The PIF was written to request further evaluation for continued operation during cycle 6. The original design flow rate was 1000GPM

and the engineering analysis was performed for a flow rate of 900GPM to the DG jacket water heat exchanger. A 10 CFR 50.59 safety evaluation, 96-0098, found the analysis to be satisfactory for the period through refueling outage 6.

Pump performance continued to by tracked through August although the train was never put into (a)(1). The "A" pump continued to supply 1000GPM to the DG and approximately the same amount to the other loads. The "B" pump was rebuilt two years before and was supplying near the rated flow. The "A" pump had been rebuilt in 1990 and was about 8% degraded in August 1996.

c.7 Conclusions on the Emergency Service Water System

The team determined that the ESW system appeared to be in acceptable material condition. Degradation of the "A" pump was appropriately addressed. The team concluded that the system was being properly treated under the maintenance rule.

b.8 Findings and Observations on Power Transformers

The Power Transformer system, which included the Main, Startup, and Auxiliary Transformers, was initially considered a low risk (a)(2) system. In view of the failure of the Unit 1 Auxiliary Transformer in Cycle 6, the system was placed in category (a)(1) in October 1996; goals were set as no more than one failure in two cycles, and no repetitive transformer component failure. The team noted that the performance of these transformers was monitored regularly. In addition to routine inspections and clean up, thermography was utilized to identify any hot spots.

During the walk down of this system, the team identified two items:

- Extensive corrosion was observed on the structure of the neutral ground of the Unit 2 Start up Transformer.
- A large number of dry leaves were noted at the base of the Unit 1 Auxiliary Transformer. The team noted this as a potential fire hazard.

The licensee promptly issued a PIF to evaluate these conditions.

c.8 Conclusions for Power Transformers

The team concluded that conservative goals were set for the power transformers and their performance was monitored satisfactorily as per the maintenance rule. With the exception of the noted deficiencies, the condition of the power transformers was considered good.

M2.2 Material Condition

a. Inspection Scope

In the course of verifying the implementation of the maintenance rule using Inspection Procedure 62706, the team performed walkdowns to examine the material condition of the systems listed in Section M1.6.

b. Observations and Findings

Except as noted in Section M2.1, the systems were free of corrosion, oil leaks, water leaks, trash, and based upon external condition, appeared to be properly maintained.

c. Conclusions

In general, the material condition of the systems examined was acceptable.

M7 Quality Assurance in Maintenance Activities

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

a. Inspection Scope

The team reviewed the report of an assist visit by the Nuclear Energy Institute (NEI) which took place in 1995. The team also reviewed a sample of licensee self-assessment activities associated with maintenance. The documents reviewed included:

- Memorandum SES-SEU-96-00215, RCIC Outage Critique, dated July 29, 1996
- Surveillance 96-052, Maintenance-Surveillance/Division 2 Outage, dated October 15, 1996

b. Observations and Findings

The team reviewed the minutes from a critique of an RCIC maintenance outage. The minutes documented considerable discussion of improving scheduling performance and reducing outage time. The inspectors considered the critique useful in focusing attention on reducing system unavailability times due to maintenance activities.

Surveillance 96-052 identified instances where work could have been accomplished more efficiently thereby reducing the unavailable time of components. The surveillance identified problems with counting of Maintenance Rule unavailability hours in that the hours counted were in excess of actual. However, the surveillance did not address the risk significance of taking multiple systems out of service at once.

c. Conclusions

Assessment activities were useful in identifying enhancements for expediting maintenance. However, the assessments did not assess the safety and risk impact of the maintenance activities reviewed.

III. Engineering

E4 Engineering Staff Knowledge and Performance (62706)

E4.1 Engineers Knowledge of the Maintenance Rule

a. Inspection Scope (62706)

The team interviewed system engineers and managers to assess their understanding of PRA, the maintenance rule, and associated responsibilities.

b. Observations and Findings

The team interviewed several RSEs, walked down systems with them, and determined that they were knowiedgeable of their systems, the Maintenance Rule performance criteria for their systems, and where their systems stood with respect to the performance criteria. However, the RSEs only had weak understanding of what the bases were for the performance criteria. Training given to the system engineers on the MR and PRA appeared to be recent and minimal and system engineers' knowledge of plant PRA was very limited. The system engineers were generally involved with the resolution of PIFs in their assigned systems. The team also noted that some system engineers were assigned a large number of systems (as high as 15 systems).

The team identified one example where the engineering organization failed to properly classify a FF. PIF 95-1949 documented that the RCIC governor valve stem was binding and that RCIC was considered inoperable due to the stem binding problem. The MRPE stated that the event was originally classified as an FF as part of the historical review performed by consultants, but was reclassified to not be an FF by the RSE during subsequent review. After the team brought the misclassification to their attention, the licensee agreed that the event was misclassified and initiated PIF 96-3398 to investigate how this FF was missed. The team noted that this misclassified problem may have been due to lack of understanding regarding what constituted an FF and associated training weaknesses.

c. Conclusions

System engineers were knowledgeable regarding their assigned systems and typically very involved in the resolution of issues and deficiencies. In most cases, system engineers' knowledge of the maintenance rule was commensurate with their involvement in MR activities. However, the team identified one example of a FF misclassification which may have been due to weaknesses in training.

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 following the conclusion of the inspection on November 19, 1996. The licensee acknowledged the findings presented.

The team asked the licensee whether any materials examined during the inspection should be considered proprietary. The licensee indicated that the four program assessments and the ORAM Shutdown Risk program manual provided to the team were proprietary information.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

N. Bonner, Director, Engineering
P. Bordley, Nuclear Engineering Supervisor
R. Collings, Manager, QA
W. Colvin, Maintenance Rule Project Engineer
D. Dervey, System Engineering Supervisor
D. Gartner, Design Engineer
D. Gudger, Licensing Engineer
R. Hayes, Manager, SES
V. Higaki, Manager, ISEG
J. Icard, Maintenance Rule Consultant
K. Jury, Compliance Supervisor
H. Kelly, Shift Supervisor
J. Kloosterman, Manager, RAS
C. Kuester, Systems Scheduler
T. Lutkehaus, Maintenance Manager
J. Messina, Manager, Operations
S. Meyer, PRA Engineer
L. Meyers, Vice President - Nuclear
S. Moffitt, Shift Technical Advisor
C. Reiter, Radiation Protection Manager
S. Sanford, Engineer, ISEG
R. Sochia, Senior Operations Scheduler
F. Smith, Unit Supervisor
R. Stiffler, Quality Auditor
R. Tadych, Manager, Nuclear Engineering Section
R. Wheeler, Work Control SRO
L. Worley, Director, PNSD

NRC

R. Corriea, Section Chief, NRR
M. Farber, Reactor Inspector, RIII
A. Fresco, Research Engineer, BNL
D. Kosloff, Senior Resident Inspector, RIII
R. Langstaff, Reactor Inspector, RIII
S. Salicrup, Intern, NRR
F. Talbot, Reactor Operations Engineer, NRR
T. Tella, Reactor Inspector, RIII
R. Twigg, Resident Inspector, RIII

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

ITEMS OPENED, CLOSED, AND DISCUSSED

50-440/96014-01(DRS)	NCV	Communications system not in scope
50-440/96014-02(DRS)	UNR	Potential misclassification of risk significant functions
50-440/96014-03(DRS)	IFI	Reliability/Availability balance
50-440/96014-04(DRS)	VIO	Failure to perform risk assessment
50-440/96014-05(DRS)	UNR	Reliability performance criteria
50-440/96014-06(DRS)	IFI	Structure monitoring program
50-440/96014-07(DRS)	IFI	Corrective actions from PIF 96-3276

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
AOP	Abnormal Operating Procedure
CA	Compressed Air
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CW	Circulating Water
DG	Diesel Generator
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling Systems
ED	Electrical Distribution
EDG	Emergency Diesel Generators
EOP	Emergency Operating Procedure
ESW	Emergency Service Water
EPRI	Electric Power Research Institute
FF	Functional Failure
FW	Feedwater
IEEE	Institute of Electrical and Electronics Engineers
IFI	Inspection Follow-up item
INPO	Institute of Nuclear Plant Operations
IOE	Industry Operating Experience
IPE	Individual Plant Evaluation
ISEG	Independent Safety Engineering Group
LER	Licensee Event Report
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
MPFF	Maintenance Preventable Functional Failure
MRPE	Maintenance Rule Project Engineer
MS	Main Steam

LIST OF ACRONYMS (CONTINUED)

NOV	Notice of Violation
NPRDS	Nuclear Plant Reliability Data System
NUMARC	Nuclear Management Resource Council
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PCIS	Primary Containment Isolation System
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
QA	Quality Assurance
RHR	Residual Heat Removal
RPS	Reactor Protection System
RSE	Responsible System Engineer
SRO	Senior Reactor Operator
SSC	Structures, Systems or Components
TS	Technical Specifications
URI	Unresolved Item

LIST OF DOCUMENTS REVIEWED

- PEI-SPI 2.5, Bypass of RCIC Isolation and Suction Transfer, Revision 0, effective 8/19/94
- PEI-B13, RPV Control (Non-ATWS) flowchart, Revision B
- PEI-B13, RPV Control (ATWS) flowchart, Revision B
- Calculation 6.20, Recommended Performance Criteria for Standby Systems/Trains Based on PRA Failure Rates, draft, prepared 10/23/96
- TAI-0513, Monitoring the Effectiveness of Maintenance Structural Monitoring Program, Revision 0, effective 10/25/96
- Maintenance Rule Functions, Performance Criteria and Classifications, Revision 1.11, approved 10/31/96
- Maintenance Rule System Report, Cycle 5, System: 99, Miscellaneous Structures, approved 8/23/96
- Perry Individual Plant Examination (IPE), July 1992
- NEI-0330, Revision 3, dated 02/22/96, "Interface Reviews and Evaluations"
- DCP 94-0092, Revision 0, dated 6/22/94
- "Periodic Assessment Report of Maintenance Effectiveness for (Operating) Cycle 5 (August 14, 1994 - April 10, 1996)"
- Calculation No. 6.16, Revision 0, dated 06/21/96, "Determination of Level 1 PRA Risk Significant Systems for the PNPP Maintenance Rule"
- Expert Panel Meeting Minutes # 50, May 6, 1996
- Calculation 6.17, Revision 0, dated 6/21/96, "Determination of Risk Significant Systems from the Level II Perspective"
- "Risk Significant Functions and Basis (PRA or EP for Expert Panel)," 11/05/96
- "Maintenance Rule System Report - Cycle 5 (August 14, 1994 - April 10, 1995) Super System: T23 Containment," dated 9/9/96
- Expert Panel Meeting Minutes, Meeting # 48, April 29, 1996

LIST OF DOCUMENTS REVIEWED (CONTINUED)

- EPRI developed software program ORAM (Outage Risk Assessment and Management) Version 2.0 for Windows (© 1994 - contains EPRI PROPRIETARY INFORMATION)
- Internal memorandum, ISEG-96-00024, dated 05/22/96, "Post RF05 Shutdown Safety Report"
- PAP-0116, Revision 0, 10/12/95, (with Change No. 3, 07/14/96) "Shutdown Safety"
- Expert Panel Meeting Minutes, Meeting # 65, June 27, 1996
- Calculation No. 6.16, Revision 0, 07/05/96, "Establish Unavailability Performance Measures and Unavailability Criteria for the PNPP Maintenance Rule"
- Calculation No. 6.21, DRAFT REVISION 0, 10/28/96, "Recommended Performance Criteria for Operating Systems/Trains Based on PRA Failure Rates"
- Calculation No. 6.23, DRAFT REVISION 0, 04/26/95, "Availability and Reliability Balance"
- Procedure No. PAP-0130, Revision 1, effective date 07/31/96, "On-Line Schedule Development"
  Attachment 5, "Risk Matrix for Selected Equipment/System Combinations"
  Attachment 6, "Risk Categories"