U.S. NUCLEAR REGULATORY COMMISSION

REGION III

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Licensee:	Consumers Power Company 212 West Michigan Avenue Jackson, MI 49201
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
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EXECUTIVE SUMMARY

Palisades Nuclear Generating Station
NRC Inspection Report 50-255/97003(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 both a one-week and a three-day on-site inspection by regional and Office of Nuclear Reactor Regulation (NRR) inspectors.

Operations

- The operators were knowledgeable about the specific Maintenance Rule (MR) requirements for the control room, and were knowledgeable in general about the MR. Operators' knowledge was consistent with their responsibility for implementation of the MR. There was no indication that the MR detracted from the operators' ability to safely operate the plant.

Maintenance

- The team concluded that the licensee had correctly identified the Structures, Systems, and Components (SSC) that were required to be within the scope of the MR. Documentation of the technical basis for scoping decisions was adequate and justifications for decisions to include or exclude SSCs were acceptable.

- The expert panel was composed of well-qualified, experienced personnel. The Palisades Individual Plant Examination (IPE) was used in conjunction with their experience base to accurately assess the risk significance of the SSCs.

- The team concluded that the licensee's approach to establishing the risk ranking for SSCs within the scope of the MR was adequate, although it deviated from NUMARC 93-01 recommendations in its reliance on the Four-Quadrant plot of the Fussell-Vesely versus Birnbaum measures. However, weaknesses in the licensee's implementation included the use of an outdated IPE and inconsistent documentation of the expert panel's determinations.

- The procedures for performing periodic evaluations met the requirements of the rule and the intent of the Nuclear Management Resource Council (NUMARC) implementing guidance. The first periodic assessment had not been completed at the time of the inspection and will be reviewed as an Inspection Follow-up Item (IFI).

- The licensee's process for conducting the balance between reliability and availability was considered acceptable. The first periodic assessment had not been completed and its incorporation of balancing reliability and availability will be reviewed as a part of an IFI.

- The team viewed the licensee's process for assessing plant risk resulting from multiple equipment outages to be appropriate. The licensee's Control of Equipment procedure provided extensive guidance to the plant operators and schedulers to

control the risk of concurrent maintenance and testing activities. The procedures controlled risk evaluation for both scheduled and emergent work. The team did not find any examples of emergent work not being analyzed for risk significance.

- The establishment of performance criteria and goal setting was considered acceptable. Performance criteria for safety significant SSCs were validated through a sensitivity study of the Probabilistic Safety Assessment (PSA). Goals and monitoring for (a)(1) components were included in documented action plans and were appropriate.

- The licensee properly scoped buildings and enclosures as structures under the rule. Walkdowns had been completed; deficiencies were recorded, classified, evaluated, and dispositioned properly. Criteria for transferring structures from (a)(2) to (a)(1) were included in the criteria for classifying and dispositioning the deficiencies.

- The licensee's ongoing program for collecting, distributing, and analyzing industry operating experience (IOE) was consistent with NUMARC 93-01 guidance and appeared to be effective. The first periodic assessment had not been completed and its incorporation of industry operating experience will be reviewed as a part of an IFI.

- Walkdowns of plant systems, in conjunction with review of deficiency lists, revealed a sizable number of material condition problems. This was consistent with the (a)(1) classification of components in the Primary Coolant System (PCS), High Pressure Safety Injection system (HPI), Instrument Air System (IAS), and turbine Generator System (TGS).

<u>Quality Assurance (QA)</u>

- The team concluded that Nuclear Performance Assessment Department (NPAD) surveillances were detailed and thorough. The MR staff addressed the issues in a timely manner; corrective actions to the surveillance findings were completed or in progress.

<u>Engineering</u>

- System engineers (SEs) had been trained and appeared qualified to provide oversight of the implementation of the rule for their respective SSCs. The licensee's development of a MR qualification card for system engineers was good. However, some SEs were unfamiliar with performance criteria, and were not timely in classifying events as Maintenance Preventable Functional Failures or in recognizing that performance criteria had been exceeded.

Report Details

Summary of Plant Status

The plant was operating at full power at the start of the inspection but reduced power during the inspection to address problems with overheating of isophase bus connectors.

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 both a one-week and a three-day on-site inspection by regional and NRR inspectors.

I. Operations

04 Operator Knowledge and Performance

04.1 Operator Knowledge of Maintenance Rule

a. Inspection Scope (62706)

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 (RO) and two senior reactor operators (SROs) to determine if they understood the general requirements of the rule and their particular duties and responsibilities for its implementation. The team reviewed EM-25, Revision 1, "Maintenance Rule Program," and Administrative Procedure 4.02, Revision 13, "Control of Equipment."

b. Observations and Findings

The team found that the operators depended on operations scheduling personnel and PSA personnel to perform quantitative assessments of the risk of scheduled work. SROs understood their MR responsibilities in the control room for performing either quantitative or qualitative risk assessments for emergent work if scheduling or PRA personnel were not available. System engineers were responsible, in conjunction with the operators, for tracking the unavailability time for SSCs within the scope of the MR. SSCs were listed in the procedure for the SROs. All three of the operators understood their responsibilities as delineated in EM-25 and Administrative Procedure 4.02.

c. Conclusions

The operators were knowledgeable about the specific MR requirements for the control room, and were knowledgeable in general about the MR. Operators' knowledge was consistent with their responsibility for implementation of the MR. There was no indication that the MR detracted from the operators' ability to safely operate the plant.

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 five regional and headquarters inspectors and a consultant from the Brookhaven National Laboratory. 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 MR 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 MR program was described in Procedure No. EM-25, "Maintenance Rule Program," Revision 0 (January 20, 1997). This program described the methodology used and five scoping questions to select the SSCs under the rule. The questions considered whether the systems were safety related, nonsafety-related systems that were relied upon to mitigate accidents or transients or were used in Emergency Operating Procedures (EOP), 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 MR and of systems excluded from the scope.

The licensee generated a "Maintenance Rule Scoping Document" which identified specifically for each SSC whether or not that SSC had a MR function, was High or Low Safety Significant, whether it was modeled in the Probabilistic Safety Assessment (PSA), and the basis for the determination of whether the SSC was within the scope of the MR.

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 considered about 100 SSCs in the scoping phase. Of these, 80 SSCs were placed within the scope of the MR. Justification for exclusion of systems was contained in the notes of the scoping document. The scoped systems included both Level 1 PSA safety-related and balance of plant systems, and also Level 2 PSA containment systems.

The inspectors reviewed both the SSCs within the scope of the rule and those SSCs excluded from the scope as defined by the licensee's program. The team found no examples of SSCs that should be within the scope of the rule that were not identified by the licensee. Additionally, the SSCs were rescoped after the most recent plant outage.

c. Conclusions

The team concluded that the licensee had correctly identified the SSCs that were required to be within the scope of the MR. Documentation of the technical basis for scoping decisions was adequate and justifications for decisions to include or exclude SSCs were acceptable.

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," required that safety be taken into account when setting performance criteria and monitoring under paragraph (a)(2) of the rule. This safety consideration was to be used to determine if the 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 PSA insights with operations and maintenance experience, and to compensate for the limitations of PSA modeling and importance measures. The team reviewed the composition of the expert panel and 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 MR and to understand the functioning of the panel.

b.1 Observations and Findings on the Expert Panel

The team reviewed the licensee's process and procedures for establishing an expert panel and determined that the expert panel was established in accordance with the guidance contained in NUMARC 93-01. The expert panel's responsibilities included the final authority for decisions regarding MR scope, risk determinations, and performance criteria selection.

The licensee used the expert panel process in conjunction with a PSA ranking methodology to determine which SSCs were within the scope of the rule, to determine risk significance ranking, establish the lists of (a)(1) and (a)(2) SSCs, and assign performance criteria and goals.

The composition of the expert panel, and the qualifications and experience of the panel members were considered appropriate. The panel included plant staff who were experienced in engineering, PSA, operations, maintenance, and work scheduling. The expert panel possessed a total of greater than 145 years of nuclear power experience and had between 10 and 27 years of nuclear power plant experience. Most of the panel members had a degree in engineering or had an SRO license. The expert panel had received extensive MR training and the team noted that the panel members demonstrated a generally good knowledge of both the MR and the functions of the expert panel. The panel was involved in several aspects of MR implementation beyond those specified in NUMARC 93-01 such as review and approval of system performance criteria and MR periodic reports.

Expert panel meeting notes were generally adequate; however, the team noted that this was not consistent. Meeting notes from the early stages of implementation and from recent meetings were thorough and informative, but notes during the critical phase of risk determinations were less comprehensive. In particular the notes lacked justification for risk determinations made for systems that were not modeled in the PSA and thus had no risk importance measures.

c.1 Conclusions on Expert Panel

The expert panel was composed of well-qualified, experienced personnel. EM-25 was consistent with NUMARC 93-01 guidance and provided adequate instructions to govern the panel's activities. The team concluded that the panel generally functioned well, but inconsistent content of meeting notes was a weakness.

b.2 Observations and Findings on Risk Determinations

b.2.1 Analytical Risk Determining Methodology

The licensee began to develop a PSA for the Palisades plant beginning in 1983. In May 1985, the licensee submitted the PSA to the NRC as input to an issue concerning main steam isolation valves (MSIVs). The PSA was completed during the time period of 1985 to 1989. The event trees and fault trees were specific to the Palisades plant. The PSA was based on the Set Equation Transformation System code.

To support the licensee's response to Generic Letter 88-20 for the Individual Plant Examination (IPE), the licensee updated the PSA using licensee personnel with the assistance of TENERA, Gabor, Kenton and Associates, and ABB Impell. Both generic and plant-specific data were used. The data were current through 1990. The IPE, specifically for internal events and internal flooding, was transmitted to the NRC in January 1993. The IPE,included both a Level 1 PSA based on Core Damage Frequency (CDF) and a complete Level 2 PSA based on a detailed plant specific containment structural analysis.

In June 1995, the licensee completed an individual plant examination of external events (IPEEE). This examination addressed seismic, fire, and other external events. At the time of this inspection, the licensee was updating both the IPE and the IPEEE. The licensee hoped to have this update completed by July 1997. Both the IPE and IPEEE were available to the plant staff as maintained hard copy volumes. Access to the PSA computer model for revision was limited to the PSA staff; others could access the PSA on a read-only basis. The licensee was also in the process of adapting the Electric Power Research Institute (EPRI) developed Equipment Out of Service (EOOS) program to the Palisades-specific design and operating conditions. (Evaluation of this effort was not part of the scope of the inspection). Once completed, the licensee also intended to include the IPEEE PSA as part of its MR PSA. Therefore, any risk insights from the seismic, fire and other external events would be evaluated for impact on MR SSCs.

The licensee had in place administrative procedure No. 9.03, Revision 15, dated May 30, 1996, "Facility Change" which addressed the disposition of plant design modifications. Attachment 1 to this procedure was the "Facility Change Master Checklist." Attachment 7 was the "Design Review Checklist (DRC)." Attachment 10 was the "Notice of Modification." Both the PSA Engineer and the Maintenance Rule Engineer were included on the distribution for Attachment 10, "Notice of Modification." However, in neither Attachment 1, nor Attachment 7 was there a clear statement which indicated that the PSA Engineer had reviewed the modification specifically to consider whether or not it had an impact on the PSA.

The team also noted that there was no written procedure or guidance document which defined the criteria to be used to determine whether a modification should result in a PSA update. The criteria for deciding whether a modification was significant were at the discretion of the PSA engineer. Although not a formalized process, the PSA Engineer stated that modifications were reviewed for PSA impact based on the criteria in Section 4.2, "Quantitative Screening Criteria," of EPRI-TR-105396, "PSA Applications Guide," August 1995.

To determine the risk significance of SSCs from the perspective both of a Level 1 PSA (frequency of core damage) and of a Level 2 PSA (containment failure), the licensee chose to augment the recommendations of NUMARC 93-01, Section 9.3.1 "Establishing Risk Significant Criteria," with a Fussell-Vesely vs. Birnbaum four-quadrant plot with no human errors.

NUMARC 93-01, Section 9.3.1, specifically identified the Risk Reduction Worth (RRW) importance measure (Methods A and B), the cutsets contributing to the top 90% of CDF, and the Risk Achievement Worth (RAW) importance measure as the measures to use in establishing risk significant criteria. It should be noted that for the RRW and Top 90% of CDF importance measures, those events not specifically related to maintenance (e.g., operator error and external or initiating events) were eliminated from consideration. There was no corresponding restriction for the RAW importance measure. It has been commonly accepted that the Fussell-Vesely importance measure closely paralleled the RRW measure.

b.2.2 <u>Adequacy of Expert Panel Evaluations</u>

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 first risk ranking of those systems included within the scope of the MR was presented to the expert panel in June 1996. In September 1996, the expert panel upgraded three systems to High Safety Significance: VAS/CRV Control Room Heating, Ventilating, and Air Conditioning (HVAC); ESS/LPI Low Pressure Safety Injection; and ESS/SIT Safety Injection Tanks. In January 1997, the expert panel reviewed systems that were identified as important using the criterion of 90% of CDF.

The final risk significance ranking was based on a combination of results from the PSA and expert panel judgement based on deterministic considerations. Thirty-six (36) of the scoped systems were considered to be of High Risk Significance while forty-four (44) scoped systems were considered to be of Low Risk Significance.

With respect to the 90% of CDF measure, the licensee identified three PSA basic events which fell into this category. Although not stated in the expert panel Meeting Minutes of January 27, 1997, the expert panel downgraded these three basic events to Low Safety Significance. Apparently it was not until the expert panel meeting minutes of January 27, 1997 that any mention was made that the expert panel had ever considered the rankings obtained by the Top 90% of CDF importance measure. Although the licensee's procedure EM-25 indicates that the 90% of the overall CDF (eliminating operating errors), RAW, and RRW importance measures were to be used in the risk ranking process, it was apparent that in fact the expert panel relied heavily upon the Four-Quadrant plot of the Fussell-Vesely versus Birnbaum measures. The details of the importance calculations consisted to a large extent of justifications of the mathematical parallels between the Fussell-Vesely versus RRW measure and between the Birnbaum versus RAW measures. Since these details had not been prepared prior to the inspection, it was not clear to the team how the expert panel could have fully comprehended the differences between the Fussell-Vesely and Birnbaum measures from those identified specifically in NUMARC 93-01 and their corresponding threshold values.

The Four-Quadrant plots indicated a Fussell-Vesely threshold of 5.0E-03 (0.005) and a Birnbaum threshold equal to the plant CDF (5.15E-05). The detailed calculations indicated that at the component level a Fussell-Vesely measure of 0.005 (0.5% of CDF) was approximately equal to a RRW of 1.005 (the NUMARC 93-01 threshold) and that to obtain a threshold for the Birnbaum measure that yields results similar to an RAW of 2 (NUMARC 93-01 threshold), a threshold value for the Birnbaum measure should be between one and two times the baseline CDF. Therefore, the licensee concluded that the Birnbaum and Fussell-Vesely measures would yield at least equai if not more conservative results than the RRW and RAW measures. The licensee's justification for developing the Four-Quadrant plots was that it was clearer to present the issue of importance rankings in a graphical manner, particularly to members of the expert panel who were not familiar with PSA.

At the beginning of the inspection, the team requested that the licensee provide any calculations documenting the use of the importance measures and also the meeting minutes which documented the decisions made by the expert panel concerning the importance measures. The licensee responded that the details of the calculations supporting the importance ranking results were in preparation by a consultant from

TENERA who was present during the inspection. The details of the calculations and the meeting minutes were not presented to the team until late in the afternoon of the third day of the inspection, giving the team very limited time to review them. The details were not in the form of a formal calculation, i.e., there were no dates marked or any signatures.

c.2 Conclusions on Risk Determinations

The team concluded that the licensee's approach to establishing the risk ranking for SSCs within the scope of the MR was adequate, although it deviated from NUMARC 93-01 recommendations in its reliance on the Four-Quadrant plot of the Fussell-Vesely versus Birnbaum measures. However, weaknesses in the licensee's approach included the use of an outdated IPE and inconsistent documentation of the expert panel's determinations.

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

a. Inspection Scope

Section (a)(3) of the rule requires that performance and condition monitoring activities and 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 the procedural guidelines for these evaluations.

b. Observations and Findings

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

- EM-25, Revision 1, "Maintenance Rule Program," Section 7.9
- EM-20, "System Performance Monitoring"

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 first assessment could not be reviewed since the licensee planned to complete it by the summer of 1997. This will be tracked as inspection follow-up item (IFI) (50-255/97003-01a(DRS)).

c. Conclusions

The procedures for performing periodic evaluations met the requirements of the rule and the intent of the NUMARC implementing guidance. The first periodic assessment required by the MR had not been completed at the time of the inspection.

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

a. Inspection Scope

Paragraph (a)(3) of the MR 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 licensee's plans to ensure this evaluation was performed as required by the rule.

b. Observations and Findings

The licensee's instructions for conducting reliability and availability balances were contained in EM-25, Section 7.7. 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. Routine adjustments to balance reliability and availability were made as identified; however, more extensive evaluations of balancing reliability and availability will be made during the first overall assessment scheduled for completion in the summer of 1997. Therefore inspection of the balancing between availability and reliability will be reviewed as a part of the IFI identified in paragraph M1.3 above (IFI 50-255/97003-01b(DRS)). The present guidance was qualitative; however, the licensee was investigating several other potential methods, one being a quantitative method. If used, the quantitative method would be an improvement over currently accepted qualitative methodology.

c. Conclusions

The licensee's processes for conducting the ongoing reliability and availability balance and the first overall assessment were acceptable. The licensee was investigating several potential alternative methods, one being a quantitative process which if implemented would be an improvement over currently accepted qualitative methodology.

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

a. Inspection Scope

Paragraph (a)(3) of the MR states that in performing monitoring and preventive maintenance activities 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 MR program engineer, the PSA engineer performing PSA risk assessments, plant operators, and planning and scheduling personnel.

b. Observations and Findings

b.1 Procedural Guidance/"Risk Matrix"

The licensee provided elaborate qualitative guidance to the plant operators and schedulers to control the risk of concurrent maintenance and testing activities. The licensee had in effect Administrative Procedure No. 4.02, Revision 13, effective date October 10, 1996, "Control of Equipment." The procedure had the following attachments which were of direct importance to the scheduling of on-line maintenance activities:

Attachment 9: "Safety Assessment for Removal of Plant Equipment (SSCs) from Service." This attachment was completed if all of the following conditions existed:

- required entry into unplanned maintenance outages (i.e., not identified on plant 13-week schedule which has been reviewed by the Operations Scheduler(s) and the Probabilistic Safety Assessment Group);
- involved high safety significance systems (refer to Attachment 11);
- rendered the system incapable of performing its MR function (the Operations Schedulers had a copy of the "Maintenance Rule Scoping Document" that contained these functions);
- did not occur with the SSC positioned into its designed safety position prior to removal of its power source (e.g., failing open a valve to work on a breaker).

This attachment contained a flow chart to guide the user on the proper scheduling of the proposed on-line maintenance activity. The attachment was completed by an Operations Scheduler or On-Shift SRO. It was reviewed by the Operations Support Supervisor.

Attachment 10: "Documentation of Safety Assessment by System Engineering/Probabilistic Safety Assessment Group for Impact of Equipment Inoperability on CDF." This attachment consisted of two memoranda. The first memorandum was directed to the PSA Group/System Engineering from the Operations Scheduler requesting a safety assessment for impact on CDF due to removal from service of the identified SSCs for maintenance and/or testing on the specified days of the week for a specified Work Week. The second memorandum was the return to the Operations Scheduler from the PSA Group/System Engineering identifying the largest increase for that day above the baseline daily CDF. A value of 1 to 2.35 indicated Low Risk, 2.35 to 10 indicates Medium Risk, and >10 indicates High Risk.

Attachment 11: "Palisades Systems Maintenance Rule Safety Rankings." This attachment listed the systems/trains in order of their safety ranking. The systems/trains were ranked based on their importance to the achievement of plant safety functions (anti-core-melt, containment integrity). The user was advised that the operability of high safety significant systems/trains have the most impact on reducing plant risk (prevention of radiation release). The user was also advised that inoperability times were required to be tracked to assess their impact on assumptions made in the Palisades PSA for CDF. The attachment provided additional notes:

- Palisades tracked inoperability on the train level (for high safety significant systems);

- Systems were designated by their primary system designator and a secondary designator specifically prepared for the MR;

- Only one (1) "High Safety Significant" system/train could be removed from service at a time without a documented evaluation per Attachment 9 or 10 on overall impact on plant safety.

Attachment 12: "SSCs Separated by Critical Safety Function." In this attachment, SSCs were listed separately by LEFT and RIGHT channel (where applicable) and listed as COMMON where circuit separation criteria were not met and/or common piping configurations existed. SSCs were also separated by CRITICAL SAFETY FUNCTION to evaluate impact on that particular safety function. The safety functions were:

- Reactivity Control
- Maintenance of Vital Auxiliaries (MVA) Electric
- Inventory Control
- Pressure Control
- Heat Removal
- Containment Isolation
- Containment Atmosphere
- MVA Water
- MVA Air

The 13-Week Rolling Schedule was a master schedule which included the separation of critical safety functions on a train basis for each week of the rolling schedule. The Plant On-Line Schedule was a schedule network consisting of plant activities scheduled for a minimum of six months, usually through the next refueling outage. The activities included Technical Specification Tests, as scheduled on the Two Year Schedule, all open work orders, meetings, maintenance training, projects, etc. In this manner, maintenance and testing activities were performed on a nearly continuous basis as opposed to a single week in which major portions of entire trains were taken out of service simultaneously.

The Generic Quarterly Safety Train Schedule was a 13-week schedule of Technical Specification tests, system windows, and other significant activities that were performed on a quarterly or less basis. Scheduling of Technical Specification tests were first scheduled on a weekly safety train (Left, Right, Common) basis. Whenever applicable, Technical Specification tests were used as post maintenance tests to eliminate redundant testing. The Generic Quarterly Schedule was reviewed by the PSA group to ensure proper train separation for concurrent system window scheduling during a work week. The licensee identified and grouped Maintenance

Rule High Risk, Medium Risk, Low Risk, and No Risk (i.e., not MR scope equipment) on this schedule.

System windows were developed for on-line and outage scheduling. These were selected portions of plant systems that could be removed from service under specific plant conditions to perform maintenance and testing activities. The boundaries ranged from complete systems to specific plant components. The scoping of the system windows aligned with the MR program.

Several examples of completed Attachment 9s were reviewed by the team to determine how emergent work situations were handled by the licensee. Licensee personnel involved in the process demonstrated good knowledge of the MR and their assigned functions in completing and reviewing the Attachment 9 examples. For emergent work which occurred on off-hours and weekends, the appropriate licensee personnel were contacted off-site by telephone to obtain their concurrence with the scheduling of the emergent work together with the equipment already out of service or scheduled to be out of service. Off-site contact appeared to be a routine, but not a frequent occurrence, with the appropriate personnel making themselves available. There were only two people routinely contacted: the Operations Scheduler SRO, or as a backup, the PSA Engineer. Others could also be contacted, such as the Operations Support Supervisor. In the future, once the EPRI-sponsored EOOS software was made operational, the licensee hoped to reduce the dependence of the plant operators on offsite personnel to obtain approval for emergent work situations. At that time, operators should be able to determine the on-line risk profile using the EOOS software. The team did not find any examples of emergent work not being analyzed for risk significance.

The team noted that in the IPEEE, Table 4.1.11.1, "Palisades Plant Response to Area Specific Fires," the licensee provided the results of the CDF for each fire area or zone based on a particular Ignition Frequency for that particular fire area or zone. The total CDF due to fires was 1.7E-04. In Section 8.0, "Results and Conclusions," paragraph 8.1.2, "Fire," the licensee stated the following:

> Eighty-five percent of the CDF associated with internal fires can be traced to five rooms/burn areas:
>
> (1) turbine building
> (2) main control room
> (3) cable spreading room
> (4) spent fuel pool equipment room
> (5) auxiliary building 590 corridor

The results of the Fire IPEEE accident sequence quantification were derived from a methodology that included a number of conservative assumptions. Fires were assumed to increase until they completely engulfed the area where they were located. In addition, with the exception of the main control room, cable spreading room and the 2.4kV switchgear rooms (fire areas 3 and 4), the effects of suppression were not credited. Therefore, while the CDF due to internal fires was higher than desired, the methodology as applied resulted in a conservative CDF.

The CDF in several fire areas was reduced due in large part to Palisades plant specific implementation of the requirements of 10 CFR 50, Appendix R. These requirements, including separation of alternate/redundant trains of safe shutdown equipment, fire barriers, and an alternate shutdown location (outside of control/cable spreading rooms) combined to limit the total CDF due to fires. The administrative control of transient combustibles was also a contributing factor to the low fire CDF in certain key areas.

The team questioned the licensee as to whether a risk assessment was performed under the on-line scheduling process whenever transient combustibles were located inside any given fire area or zone so as to avoid or minimize the out-of-service time of equipment required to achieve safe shutdown as a result of a fire in that particular area or zone. Such safe shutdown equipment could either be located within the particular area or zone or outside that particular area or zone.

The licensee responded that the fire PSA was not used for risk assessment of weekly maintenance activities. Transient fire loading was controlled by an administrative procedure (FPIP-7, "Fire Prevention Activities"). The quantity was not tracked but the Fire Protection System walkdowns assured that oversight and correction occurred, as necessary. An administrative procedure (AP 5.01, "Initiation and Planning of Work Requests/Work Orders") was used to determine if an amount of combustible material exceeding the specified limits was placed in a fire area. A Hot Work Permit was issued for welding, burning, cutting or grinding. In addition, under the Hot Work Permit, fire watches were placed, explosive gases were sampled, combustible material was limited, adjacent electrical systems were deenergized, surrounding systems were protected, and fire extinguishers were specified and located. Fire pumps were controlled under Administrative Procedure 4.02, "Control of Equipment."

In the future, the fire PSA will be used for plant activities. Any transient fire loads added to a fire area or zone will be considered and require additional risk analysis if the quantity of the fire loading is below the limits set in the fire PSA. Administrative connections were planned between the PSA section and Maintenance Planning to assure that risk assessment, including fire risk assessment, is performed when transient combustible materials and Hot Work Permits were involved with maintenance activities.

b.2 <u>Involvement of PSA Engineer in Scheduling Process</u>

The PSA engineer was only involved routinely in evaluating the weekly outage schedule about one week in advance. The PSA Engineer was not consistently involved in emergent work situations but adequate qualitative guidance for risk assessment was provided by Attachments 9 through 12 of Administrative Procedure 4.02. Although the scheduling process relied heavily on two people, the PSA engineer and the Operations Scheduler SRO, to determine risk significance, the qualitative guidance provided adequate assurance that on-line maintenance and testing activities were controlled as effectively as possible, short of the on-line risk and unavailability monitoring software in the process of being adapted.

The licensee performed scheduled surveillance testing activities as identified above for the Generic Quarterly Safety Train Schedule. In addition, the PSA Engineer issued testing windows which provided preanalyzed allowable outage times for surveillance testing in case the testing could not be completed as originally scheduled.

b.3 Schedulers/Operators Knowledge and Training

The team interviewed an RO and two Shift Engineers (equivalent to a Shift Technical Advisor), one of whom was on-duty in the control room, concerning their general knowledge of the MR, specifically as it applies to procedure AP 4.02 and removing equipment out of service during power operation. The team found that these individuals' knowledge of PSA or MR requirements, with respect to considering certain BOP equipment which was not under control by the technical specifications as significant to risk, was adequate for them to perform their job functions.

The team noted that the licensee had several different lists in the control room which identified equipment which was out of service: Limiting Condition of Operations (LCO) reports, Switching and Tagging Orders Index, and the Operations LCO Board. This could increase the possibility that the control room operators might not be aware that MR scope nonsafety-related balance of plant equipment was out of service at the same time as technical specification related equipment.

c. Conclusions

The team viewed the licensee's process for assessing plant risk resulting from multiple equipment outages as appropriate. The licensee's Control of Equipment procedure provided extensive guidance to the plant operators and schedulers to control the risk of concurrent maintenance and testing activities. The procedures controlled risk evaluation for both scheduled and emergent work. The team did not find any examples of emergent work not being analyzed for risk significance.

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) components/systems

High Pressure Safety Injection
Reactor Protection System
Instrument and Service Air

(a)(2) systems

Critical Service Water
Turbine Generator
Emergency Lighting Units
Primary Coolant System
125 Volt Vital DC Power

The team reviewed each of these systems to verify that goals or performance criteria were established in accordance 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 were monitored for degradation.

b. Observations and Findings

The inspectors reviewed the licensee's performance criteria to determine if the licensee had adequately set performance criteria under (a)(2) of the MR consistent with the assumptions used to establish the safety significance. Section 9.3.2 of NUMARC 93-01 recommended that risk significant SSC performance criteria be set to assure that the availability and reliability assumptions used in the risk determining analysis (i.e., PSA) were maintained.

As with the details concerning the importance measure calculations, the licensee provided to the team on the third day of the inspection an unsigned and undated document entitled "Palisades Maintenance Rule - Quantitative Evaluation of Performance Criteria." The document provided the results of an analysis of the potential effect of Performance Criteria developed under the MR on the overall CDF of the Level 1 internal events PSA for trains of equipment modeled in the PSA. Seventy-one (71) systems were listed in the "Maintenance Rule Performance Monitoring Results" document, based on their MR designators. 47 systems were identified in the scoping document as being modeled in the PSA.

According to the analysis, trains of equipment modeled in the PSA were matched with the MR functions identified in the Results document. Approximately 200 individual trains of equipment were selected from the PSA to test the performance criteria. (The selected trains represent 37 systems compared to 47 systems modeled in the PSA.) The trains in each system were grouped into trains or reliability blocks based on the similarity of function and equipment contained with the trains. For example, the auxiliary feedwater system was broken up into four injection paths, three pump trains, and three suction sources. The analysis was performed assuming that all trains degrade simultaneously to that allowed by the performance criteria with no credit given for any corrective action that may take place as individual trains of equipment approach the availability or reliability performance criteria.

To validate the performance criteria, the licensee performed sensitivity studies using the reliability and availability criteria and evaluating the resulting changes in CDF. The licensee determined that the CDF (baseline value 5.15E-05/year) rose by a factor of three for each of the availability and reliability performance criteria sensitivity studies. The licensee concluded that the importance measures generated using the reliability and availability results identified the same systems and components as important to the CDF that were identified in the original PSA. Few new systems or components were identified as risk significant that were not already classified as such by the Expert Panel.

The team pointed out that because the data used were not updated since at least 1992 for Palisades-specific unavailabilities or unreliabilities, if there were in fact specific systems or components with higher unavailabilities or unreliabilities than the Performance Criteria assumptions then the risk ranking of SSCs could change.

Furthermore, among the 10 PSA systems apparently not included in the analysis was the Reactor Protection System (RPS). The RPS was identified in the "Maintenance Rule Performance Monitoring Results" document as having performance criteria of <5 RPS channel MPFFs per 24 months and as having actually experienced 5 channel MPFFs in the last 24 months.

In addition, the licensee did not provide any indication that the expert panel had deliberated upon the setting of performance criteria for systems not modeled in the PSA or upon the results and scope of the quantitative evaluation of the performance criteria analysis.

As a result of the above review, the NRC conducted a telephone conference with licensee personnel on March 17, 1997. The issues discussed with the licensee concerned:

- absolute magnitude of the CDF as a result of the sensitivity study
- truncation value of 1.0E-08
- adequacy of performing only separate availability and reliability analyses versus also performing a combined analysis considering all affected components at their performance criteria failure rates simultaneously for both availability and reliability

Some of the issues raised by the team included the following:

- adequacy of the scope of analysis with respect to the number of systems included in the analysis
- whether the results of the Individual Plant Evaluation of External Events (IPEEE) should be factored into the performance criteria analysis
- increasing random failure rates very likely will result in a corresponding increase in unavailabilities
- the common cause failure rates may increase as the random failure rates increase

- while a factor of three increase in CDF may not be generally considered to be very significant, if the contribution of random failures and maintenance activities to the overall CDF is small in the baseline CDF calculation, in a relative sense, a factor of three increase due to random failures and maintenance activities may actually be quite large in proportion.

In response to the NRC concerns, the licensee staff indicated that they had revised their analysis so that in some cases, the number of demands on components had been increased, while the functional failure rates were maintained the same, resulting in lower failure rates per demands. The licensee stated that the resulting CDF considering reliability alone increased by a factor of 1.5 (50%) as opposed to the first analysis factor of three. With respect to the availability analysis, the licensee stated that by eliminating illogical cutsets from their PSA model, e.g., scheduled simultaneous maintenance on both trains of diesel generators, the resulting CDF considering availability alone increased by a factor of 1.6 (60%). (They did not indicate how eliminating the illogical cutsets impacted the baseline CDF.) The truncation value used was 1.0E-09, which satisfied the NRC guideline of four orders of magnitude below the CDF. The licensee staff stated that they are considering a calculation which combined the effects of both the reliability and availability performance criteria. They agreed that an increase in unreliability would result in an increase in unavailability.

With respect to the concerns raised by the team, the licensee responded for the specific case of the Reactor Protection System (RPS) that system appears in the Anticipated Transient Without Scram (ATWS) event tree, but that there is no fault tree in the PSA for that system. Similarly, systems within the scope of the maintenance rule such as main feedwater and condensate are reflected in the PSA only in terms of the initiating event frequencies. These systems are monitored only at the plant level under the maintenance rule. There are no specific fault trees for these systems. In the analysis, the initiating event frequencies were not changed from those in the PSA.

Regarding the application of the IPEEE analysis to the performance criteria quantification, the licensee staff indicated that they do plan to incorporate the IPEEE analysis as part of their maintenance rule program, and therefore, at that time, the performance criteria will be reanalyzed accordingly.

With respect to the effects on the common cause factors, the licensee staff indicated that they did not believe that there would be any increase in the common cause failure rate solely because of an increase in unreliabilities. With respect to human errors, they had been asked by the NRC during the review of the IPE to reanalyze their human error portion of the IPE analysis. In the reanalysis, they did not take credit for operator actions, and 90% of the operator actions were performable from inside the control room.

The licensee provided the most recent calculation file entitled "Palisades Maintenance Rule Quantitative Evaluation of Performance Criteria" for the team to review. The team compared this latest calculation file to the one received during the on-site inspection. The following significant changes were noted:

- With respect to methodology statement (3) "Calculate failure probability from Performance Criteria and testing frequency," the licensee added the following paragraph:

Some SSCs had a relatively small number of demands over a two-year period. Simply dividing the number of MPFFs by the estimated number of demands is not an accurate representation of the actual failure probability of the component. SSCs having less than 20 demands were examined to see if there was a relatively significant chance of a failure occurring in a two-year period assuming the original failure probability used in the PSA. A binomial distribution was assumed to perform this test. If the probability of occurrence of the number of allowable MPFFs over the specified period of time was at least 1% then a relatively significant chance of a failure occurring was considered possible and the PSA random failure probability was assigned. Below 1%, then it was not considered to be appropriate to use the PSA value. A higher probability between the PSA value and the performance criteria should be selected and justified. Selected SSCs having greater than 20 demands were examined in this manner also. Attachment B1 contained the selection of the failure probability for all SSCs tested using this method.

- Attachment B1, which did not appear in the first calculation, was entitled "Components Having Low to Moderate Number of Demands." The column headings were: Train, Failures/Demand, $P_{random}/P_{surrogate}$, P_0, P_1, and $P_{>1}$. Five of the ten components were specifically identified under the "Train" column:

  - Instrument Air Compressor 2B
  - RAS (recirculation actuation signal)
  - SDC Suction (shutdown cooling suction)
  - PORVs
  - DG Vent

Of the above, only the PORVs had a $P_{>1}$ (probability of greater than 1 failure in 24 months) value greater than 1%, i.e., 1.8%. (the team assumed this term referred to the Pressurizer Power Operated Relief Valves). Pumps P66A&B (unidentified) also had a $P_{>1}$ value greater than 1%, i.e., 2.8%. The remaining components did not have verbal identification and had values less than 1%. From other portions of the calculation, it appeared that the other components are Compressor Air Valves (CV-3025 and CV-3030), manual valves 750 and 755 (Auxiliary Feedwater System), containment spray valves MV 3001 & 3002 (for recirculation mode), and High Pressure Safety Injection Pumps P66A & B.

It was noted that this calculation was a limited sampling effort. The licensee's methodology was acceptable to address the concern. However, the calculation should clearly identify which components are described in Attachment B1.

- A truncation limit of 1.0E-09 was used for all systems and accident sequences in the requantification. This was four orders of magnitude below the baseline CDF and met the NRC guideline.

- The Steam Jet Air Ejectors no longer appeared above the licensee's threshold values for the Fussell-Vesely and Birnbaum importance measures.

- The previously identified pump P-936, which provides automatic makeup to the relatively small capacity (80,000 gallon) Condensate Storage Tank, continued to appear above the licensee's threshold values for the Fussell-Vesely and Birnbaum importance measures. The licensee committed to present this result concerning this pump to the Expert Panel for reconsideration of its risk significance.

- With respect to the reliability performance criteria, the following changes identified in Attachment A, "Summary of Performance Criteria and Identification of Basic Events by Maintenance Rule Function," and Attachment B, "Conversion of Performance Criteria to Failure Probabilities," appeared to have the most impact on the revised results:

  (1) For system CVC/CVC, the number of demands for the Safety Injection Refueling Water Tank (SIRWT) was increased from 3 demands to 45 demands, thereby decreasing the failure rate per demand from 2 failures/3 demands (0.67 = 6.7E-01) to 2 failures/45 demands (4.4E-02).

  (2) For the Recirculation Actuation Signal (RAS), the number of demands was increased from 16 to 32, thereby decreasing the failure rate from 1 failure/16 demands (6.2E-02) to 1 failure/32 demands (3.1E-02).

  (3) For the ESS/SDC, shutdown cooling suction path, the number of demands was increased from 2 to 6, thereby decreasing the failure rate from 1 failure/2 demands (0.5 = 5.0E-01) to 1 failure/6 demands (1.7E-01). For the discharge path to the heat exchangers, the number of demands was increased from 10 to 15, thereby decreasing the failure rate 1 failure/10 demands (0.1 = 1.0E-01) to 1 failure/15 demands (6.7E-02).

  (4) For the FWS/AFW auxiliary feedwater system, the number of demands for the injection paths was increased from 26 demands to 68 demands. This change appeared to result from assuming 2 paths per test. The failure rate was decreased from 2 failures/26 demands (7.7E-02) to 2 failures/68 demands (2.94E-02).

- With respect to the "Quantification Results," the licensee stated the following:

  When the failure rates associated with the reliability Performance Criteria are applied, the total CDF rose approximately 50%. This rise was small given that maintenance rule implementation was expected to confirm existing maintenance practices and not result in significant changes to historical component failure rates. When the availability criteria was applied, the total CDF rose approximately 76%. Again this change was judged to be small given that all failure rates simultaneously are assumed to be at the maximum allowed by the performance criteria and are considered to be bounding.

The licensee's conclusion did not indicate what impact the elimination of illogical cutsets, i.e., scheduled maintenance outages not allowed by the Technical Specifications, had on the baseline CDF.

The licensee's calculation was generally well organized and presented the assumptions, data, and methodology in a generally clear manner. The licensee's methodology with respect to the use of the binomial theorem is acceptable to gain a more accurate representation of the actual failure probability of the component. SSCs having less than 20 demands were examined to see if there was a relatively significant chance of a failure occurring in a two-year period assuming the original failure probability used in the PSA. Based on acceptability of the assumptions used in the revised sensitivity studies, and the resulting small increases in CDF, the team concluded that the licensee's performance criteria were appropriately justified.

b.1 <u>Performance Criteria for Unavailability</u>

Section 9.3.2 of NUMARC 93-01 recommended that risk significant SSC performance criteria be set to assure that the availability and reliability assumptions used in the risk determining analysis (i.e., PSA) were maintained. The team evaluated the licensee's performance criteria to determine if they had been adequately set under (a)(2) of the MR, consistent with the assumptions used to establish SSC safety significance.

Availability Criteria were specified as the fraction of time that the equipment in a given train was expected to be in service. Failure probabilities associated with the unavailability of each train were derived simply by subtracting this fraction from 1.0. Unavailability values were assigned to each train within each functional grouping. Unless specifically noted, no reduction in the unavailability values was performed by sharing the total time out of service between trains in a functional group or across the system. A number of systems and their trains were never expected to be removed from service during power operation. Rather than assign an availability of 100%, the licensee assumed that some out of service time could be tolerated with little impact on risk.

The team noted that the majority of unavailability criteria established for high safety significant SSCs were less conservative than the unavailability values assumed in the PSA. The licensee had recalculated the CDF value using the MR performance criteria unavailability values. The results of that sensitivity study indicated that the CDF changed by a factor of three. Based on this increase in the CDF, the team questioned the acceptability of the unavailability performance criteria established for high safety significant SSCs. The team's concerns and the results of a revised sensitivity study were addressed in detail in Section b. above.

b.2 Performance Criteria for Reliability

The licensee's process for determining reliability criteria was contained in EM-25, Section 7.3.3 and Attachment 6. The reliability criteria were generally specified as the number of maintenance preventable functional failures (MPFFs) over a given period of time (generally 24 months) that would trigger an evaluation as to whether the equipment should have goals established and be monitored under 10 CFR 50.65(a)(1). The number of failures assumed in deriving the performance criteria failure probabilities was one less than this value. The allowable number of MPFFs specified for the system was applied to each set of functional trains within a system and was assumed to be shared between the trains. (The analysis notes that no reduction in failure probability was performed by distributing MPFFs over the entire system; each functional grouping of trains is assumed to fail at a rate dictated by the number of allowable MPFFs).

The team expressed similar concerns with the reliability criteria as were expressed with the availability criteria, i.e., questions on the assumptions used and the resulting large increases in CDF. These concerns and the resulting resolution were discussed in detail in Section b. above.

The team noted two SSCs where reliability criteria were not specified in MPFF/24 months. Emergency lighting criteria included a limitation of <10 units out-of-service at any one time. The reliability criterion for the radiation monitors was <8 component failures in 24 months.

Reviews of reliability monitoring for the High Pressure Safety Injection and the Reactor Protection systems raised questions with regard to the administration of reliability criteria. The team noted examples in the documentation for both systems where system engineers had not accurately tracked the number of MPFFs on their systems and consequently, had not recognized when the performance criteria for those systems were exceeded. The team also identified some cases where the time period for evaluations for FFs and/or MPFFs extended for several months. As a result, delays were incurred in classifying components in these two systems as (a)(1). To evaluate the extent of these issues a detailed review of Condition Reports (CR) and CR MR evaluations was conducted on four systems: Engineered Safety Features, Control Rod Drive, Auxiliary Feedwater, and the Chemical and Volume Control System. Additional examples where in excess of three months to classify a FF as an MPFF were identified as well as one example in the Control Rod Drive system where the system engineer did not recognize that an MPFF had occurred which exceeded the <3 MPFF in 24 months criteria until the quarterly system health assessment was being prepared. This resulted in a seven-week delay in initiating an evaluation for classifying the system as (a)(1). Based on these findings the team viewed the administration and control of reliability performance as a significant weakness in the licensee's MR program.

b.3 Performance Criteria for Non-risk Significant Normally Operating SSCs

EM-25, Section 7.3.3 identifies plant level performance criteria which could be used for non-risk significant normally operating SSCs. The procedure specified that if the failure of a MR function for one of these systems caused a scram, system actuation, shutdown, or derate, plant level performance criteria could be used to monitor the overall maintenance effectiveness of these systems. Palisades plant level performance criteria were based on unplanned automatic reactor trips, safety system actuations, and generation loss. Limits were as follows:

- Unplanned reactor trips per 7000 critical hours

  <1 for systems whose failure is infrequent and extremely undesirable
  <2 for systems whose failure has been a contributor to trips due to design

- Unplanned safety system actuations per year

  <1 for systems whose failure is infrequent and extremely undesirable
  <2 for systems whose failure was considered less significant

- unplanned generation loss

  <2% for systems whose failure is infrequent and extremely undesirable
  <4.5% for systems whose failure was considered less significant

The team reviewed performance criteria assigned for all non-risk significant, normally operating SSCs and noted that the licensee had chosen not to rely exclusively on plant level criteria for these systems but mixed specific and plant level criteria. Where plant level performance criteria were viewed as providing little information, specific reliability criteria were assigned. For systems where plant level criteria were clearly applicable, the licensee assigned specific reliability criteria and a combination of suitable plant level performance criteria.

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

MR paragraph (a)(1) states that appropriate corrective action shall be taken when performance or condition of an SSC "does not meet established goals." In this context, "established goals" was equivalent to MR performance criteria. The goal setting process ensures appropriate corrective action is taken for category (a)(1) SSCs. Goals are established to bring about necessary improvements in SSC performance. Problematic or poor performing SSCs are placed in category (a)(1) until their performance returned to acceptable levels. The licensee's MR program, EM-25, established goal setting and corrective action requirements.

EM-25, Attachments 4 and 5 provided a detailed description of the licensee's failure identification and goal-setting process. Attachment 4 was a flow chart and instruction for evaluating each of the decision blocks in the flow chart. Attachment 5 was a detailed discussion of FF and MPFF identification, complete with examples.

An allowable number of MPFFs over a given period of time (generally 24 months) would trigger an evaluation as to whether the equipment should have goals established and be monitored under 10 CFR 50.65(a)(1). The allowable number of MPFFs specified for the system was applied to each set of functional trains within a system and was assumed to be shared between the trains. The analysis noted that no reduction in failure probability was performed by distributing MPFFs over the entire system; each functional grouping of trains was assumed to fail at a rate dictated by the number of allowable MPFFs.

When a component was placed in category (a)(1), the responsible system engineer developed a specific action plan to correct the problem. The plan included the corrective actions (generally with target dates), goals, and monitoring instructions. These action plans were presented to and approved by station management.

b.5 Structures and Structure Monitoring

The team reviewed procedure EM-25, Maintenance Rule Program, EM-20, System Performance Monitoring, and the results in Sargent and Lundy Report SL-5106, and other associated licensee programmatic controls to determine which onsite structures were evaluated for inclusion under the Rule. EM-25, Section 7.3.4 defined the organization of the structure monitoring program. Criteria for categorizing the significance of deficiencies were included as well as criteria for dispositioning. Part of these criteria specified condition for classifying a structure as (a)(1).

Sargent and Lundy (S&L) report SL-5106, "Maintenance Rule for Structures," contained the baseline examination results from a joint walkdown conducted by S&L and Consumers Energy in October/November 1996. Deficiencies identified during the walkdowns were tabulated and transcribed onto plant drawings for ease in reexamination. The deficiencies were then evaluated and dispositioned, based on criteria contained in EM-25. SL-5106 also contained guidelines for future condition monitoring of the structures and structural elements in the 1996 baseline examination. The examination methods and frequencies are also provided in SL-5106.

c. Conclusions

The establishment of performance criteria and goal setting was considered acceptable. Performance criteria for safety significant SSCs were validated through a sensitivity study of the PSA. Goals and monitoring for (a)(1) components were included in documented action plans and were appropriate.

The licensee properly scoped buildings and enclosures as structures under the rule. Walkdowns had been completed; deficiencies were recorded, classified, evaluated, and dispositioned properly. Criteria for transferring structures from (a)(2) to (a)(1) were included in the criteria for classifying and dispositioning the deficiencies.

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 and, 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 (IOE) into their monitoring program for maintenance. The team also interviewed two system engineers.

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

The team reviewed Administrative Procedure 3.16, "Industry Experience Review Program," and EM-20, "System Performance Monitoring." The procedures provided adequate guidance to the system engineers for ensuring that industry operating experience was integrated into the monitoring program. The inspectors observed condition reports documenting that system engineers had analyzed events and information at other plants for applicability to Palisades. The first periodic assessment had not been completed and the incorporation of industry operating experience will be reviewed as a part of the IFI identified in paragraph M1.3 (IFI 50-255/97003-01c(DRS)). The team observed that the quarterly system health assessments documented the industry operating experience that the system engineers had reviewed.

c.2 Conclusions for Use of Industry wide Operating Experience

The licensee's ongoing program for collecting, distributing, and analyzing IOEs was consistent with NUMARC 93-01 guidance and appeared to be effective. The first periodic assessment had not been completed and its incorporation of industry operating experience will be reviewed as a part of an IFI.

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

M2.1. General System Review

a. Inspection Scope

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

b.1 Observations and Findings for the Primary Coolant System (PCS)

The licensee classified this safety-related system as high safety significant. The performance criteria was less than 3 MPFFs per 24 months, unidentified leakage less than .5 gpm, unplanned capability loss less than 2% per year, and less than one reactor trip per 7000 hours critical. The 1996 Fourth Quarter System Health Assessment (SHA) classified the PCS as (a)(2) with two MPFFs since April 1994. Condition Report C-PAL-97-0132, initiated January 31, 1997, documented that a third MPFF occurred December 29, 1996 when two Core Exit Thermocouple (CET) cables were discovered swapped. The CR documented that under current performance criteria the PCS should be classified (a)(1). However, PCS was so broad that performance criteria should be redefined to more accurately identify common issues and screen out unrelated ones. The recommendation from the MRE to the Manager Review Board on whether to make PCS (a)(1) was due March 14, 1997. The team noted the extended time period taken by the licensee for the evaluations.

c.1 Conclusions for the Primary Coolant System

The team concluded that the MR was properly implemented for the system. Performance criteria were varied to allow adequate monitoring of the system. Although progress in evaluations was slow, the licensee's process for identifying FFs and MPFFs was being followed.

b.2 Observations and Findings for the Critical Service Water (CSW) System

The licensee classified this system as high safety significant and the performance criteria as less than 3 MPFFs per 24 months with each CSW pump (CSWP) available more than 98% per 12 months. The 1996 Fourth Quarter CSW SHA classified the CSW system as a(2) with two MPFFs since April 1994. Condition Report C-PAL-97-0064 was initiated January 21, 1997 and documented that unavailability of the CSW system should be tracked during all operational modes. Using this new criteria the licensee determined that the 7C CSWP had less than 98% availability during the last 12 months due to being rebuilt. Each CSWP was rebuilt on a 5-7 year schedule as needed. Previously the system availability had only been tracked when the PCS was above 300 degrees F, so tracking during all modes was an improvement. The evaluation on whether to put the CSWP in a(1) for exceeding its unavailability time was still ongoing. The due date for the evaluation to be presented to the MRB was March 15, 1997. The team also noted the extended time period taken to complete this evaluation.

c.2 Conclusions for the Critical Service Water System

The team concluded that the MR was properly implemented for the system. Performance criteria were appropriate and the process for classifying a system/component as (a)(1) was being followed.

b.3 Observations and Findings for the Turbine-Generator System

The system was placed in MR category (a)(1) due to overheating of the flexible laminations between the isophase bus and the generator bushing, bushing failure, transition piece heating, inadequate cooling in the bushing box, vibration, and cracked bushing bellows. The licensee developed an evaluation of the contributing factors of the generator flex strap failure with an action plan which included any design changes and additional monitoring needed.

c.3 Conclusions for the Turbine-Generator System

Based on repetitive failures which caused unplanned generation loss, the licensee appropriately classified the turbine generator system as (a)(1).

b.4 Observations and Findings for the Emergency Lighting Units (ELU)

The emergency lighting system included 131 ELUs in the scope of the rule. The licensee had set performance criteria of less than 3 MPFFs in 24 months and less than 10 units out-of-service at any one time. This equated to 6.87% of the ELUs out-of-service at any one time. The team questioned the adequacy of the reliability performance criteria for ELUs. The comments section of the Performance Criteria Basis document stated that with overlap and the classification as low risk significance, that this percentage of units out-of-service at any one time was acceptable. The inspector identified an error in the comment section of the "Maintenance Rule Scoping Document" for ELUs. The performance criterion for less than 10 ELU out-of-service did not match the ratio of 124/131 (7 ELUs out-of-service) allowed in the comments. The licensee corrected this number to 122/131.

There was no apparent consideration of the number of demands in these criteria. As a result of inspectors concerns, the licensee revised the Performance Criteria Basis document to incorporate a consideration of 6 demands per year for the ELUs. The team did not consider this performance monitoring criteria predictive in nature because it did not detect degradation before failures occurred. A review of licensee surveillances indicated that there was a fairly low failure rate of ELUs. The licensee did not replace ELU components on a periodic basis as a result of aging. A review of CRs indicated most ELUs components, such as batteries, chargers, and lamps, were operated until failure. ELU surveillances were somewhat predictive of battery and charger condition because of evaluations performed for water loss and voltage readings, but only represented a minor part of component replacements. The "run to failure" was contrary to a statement in the licensee's procedure EM-25, which states that no items in the scoping plan were operated until failure. The licensee did not agree with the inspectors' assessment.

The MR program did not provide a clear definition of what would be considered a functional failure of the emergency lighting system. The functions listed in the Scoping Document only specified providing emergency AC, DC, and emergency ELU lighting to plant areas. The Performance Criteria Basis document indicated that functional failure of the emergency lighting system was tied to groups of lights related by their power supply. Failures of individual ELUs have not been considered a functional failure of the system. Maintenance records revealed numerous replacements and repairs to ELUs but these were not listed as FFs and were not evaluated for MPFF. Although there was an overall criterion of <10 ELUs out-of-service, it was not clear to what extent ELUs could be out-of-service in a given plant area before a functional failure was identified. Lack of clarity in identifying the functions of the emergency lighting system was a weakness in the MR program.

c.4 Conclusions for the Emergency Lighting Units

The team noted that the emergency lighting system was in generally good condition. While the performance criteria were clearly specified, monitoring MPFFs was complicated by the lack of clarity in the function statements for the system.

b.5 Observations and Findings for the 125 Volt Vital DC Power System

The 125 volt DC system was in MR category (a)(2). Battery charger 1 was degraded due to an irreparable current limiter, however the other three chargers were operating properly. The availability of charger spare parts was limited by obsolescence and the original vendor being out of business. The licensee replaced the DC breakers, and batteries. The batteries were one year old and in excellent condition. Planned improvements include charger replacement, and battery room air conditioning to extend battery lifetime.

c.5 Conclusions for the 125 Volt Vital DC Power System

The team concluded that adequate goals and performance criteria were set for the 125 Volt Vital DC Power System and the system's performance was being satisfactorily monitored, per the maintenance rule.

b.6 Observations and Findings for the Instrument and Service Air System

Reliability criteria were set at <3 MPFFs in 24 months; availability criteria for each of the trains were set at 95% in 12 months. Six MPFFs have been identified during the past two years. Four of the MPFFs were related to the instrument air compressor C-2A/C train. Repeat failures of the air compressor unloader and discharge valves caused the MR reliability performance criteria to be exceeded. C-2A and C both independently failed in February 1995. C-2A also failed in August 1995. Due to repetitive MPFFs, the MR expert panel concluded that the instrument air compressors C-2A and C be placed in MR category (a)(1). The system engineering manager approved this decision, and the system engineer prepared an action plan, "Palisades Nuclear Plant Action Plan 05," titled, "Instrument Air Compressors C-2A and C-2C," dated January 13, 1997.

The action plan detailed the corrective actions to be taken to ensure that the MR goals are met. These included the following:

(1) Reroute C-2A outlet piping to reduce moisture collection
(2) Clean C-2A, B, and C's corroded outlet piping
(3) Adjust C-2A/C outlet piping to allow moisture to drain
(4) Replace the air compressor's aftercoolers
(5) Increase operating time of C-2A/C to prevent moisture buildup
(6) Clean air receiver tanks to remove rust buildup

Placing C-2A and C in continuous operation and C-2B in standby improved quarterly availability to greater than 98% for both trains.

c.6 Conclusions for the Instrument and Service Air System

Reliability and availability criteria were appropriately established. Performance problems with the "A" and "C" instrument air compressors were appropriately addressed. The team concluded that the system was being properly treated under the maintenance rule.

b.7 Observations and Findings for the High Pressure Safety Injection System

The performance monitoring criteria for the High Pressure Safety Injection System (HPI) was less than 2 MPFFs per 24 months. This criteria had been exceeded on several occasions during the past year. Availability of the HPI as of January 31, 1997 was greater than 98.5 percent.

The inspectors were concerned with MPFFs not being added to the HPI performance monitoring until months after the problems occurred. For example, in the example listed below the number of MPFFs was at 2 during the time period that the third MPFF went unevaluated. The inspectors noted that the time period to evaluate issues to see if they were MPFFs was untimely as indicated by the following example:

- Breaker 152-113 for HPI pump P-66B had exhibited repeated closing coil failures which would have prevented a subsequent start of the pump. This breaker had failed 4 times (July 6, 1993, October 31, 1993, March 4, 1995, and April 9, 1996) in 38 months. The failure on March 4, 1995 had not been evaluated as an MPFF nor added to the MPFF data base, Consequently, corrective action was not initiated for this problem. The licensee initiated CR C-PAL-96-0444 on April 9, 1996, for the fourth failure. It was determined to be an MPFF on September 20, 1996. Breaker 152-113 was placed in MR (a)(1) status with Action Plan AP-28 developed to determine root cause and improve performance. A spare breaker was subsequently installed. On December 9, 1996, an MR Evaluation of C-PAL-96-0444 was updated in the data base to reflect the FF as an MPFF. On February 1, 1997, the HPI System trend graph was updated to reflect the MPFF. The graph reflected the MR criterion of 3 MPFFs for the period April 9, 1996, to June 12, 1996.

The inspectors noted that the time frame for the completion of some classifications coincided with the preparation of the quarterly SHA rather than on an ongoing basis.

The inspectors were concerned that some SEs did not understand that functional failures on their systems were to be recorded even though the systems were not required to be available. These failures were reliability data for the systems regardless of availability requirements; therefore, the SEs should count these failures as functional failures. This was identified in reviewing the circumstances of the following example:

- C-PAL-96-1533 was written on the HPI system when it was identified that SC-96-050 MOV control circuit upgrade resulted in a train separation problem for MO-3009 and MO-3011. This problem was corrected through the Engineering Design Change process for these and other valves affected by the specification change. The issue was identified after post-maintenance testing for these valves, but before the HPI was required to perform its MR function (plant was still in cold shutdown mode). This condition report was therefore not considered a functional failure by the system engineer as evidenced by his correspondence with the MRE.

The number of CRs written on HPI during the 4th quarter of 1996 was 17. Ten of these failures were equipment related. In addition, a large number of problems with the HPI system were listed on the licensee's degraded condition list. In the SHA the licensee characterized HPI performance as marginal.

c.7 Conclusions for the High Pressure Safety Injection System

The team concluded that the material condition of the HPI system was not good. Observations during walkdowns were consistent with a maintenance rule record of problems. This confirmed the need to have portions of the system considered (a)(1). The team noted untimely MPFF evaluations, inaccurate tracking of MPFFs, and untimely recognition of exceeding performance criteria.

b.8 Observations and Findings for the Reactor Protection System

The inspectors had a concern that the licensee did not have adequate justification for increasing the reliability performance criteria for the reactor protection system from <2 MPFFs to <5 MPFFs. The system engineer believed that this number of failures would be acceptable based on the material condition of the RPS and the number of components and subsystems. The SE was evaluating whether to allow additional MPFFs for subsystems of the RPS. In addition, the licensee did not provide any indication that the expert panel had deliberated upon the setting of performance criteria for systems not modeled in the PSA or upon the results and scope of the quantitative evaluation of the performance criteria analysis. After being informed of this finding, the licensee provided the appropriate justification. This justification was based on a careful accounting of the number of components in the system and the corresponding opportunity to fail. It also indicated that consideration was being given to further raising the reliability criteria. Since this effort was still ongoing, it will be reexamined as an IFI (50-255/97003-02(DRS)). This issue was of particular concern because the RPS system was only modeled in the PSA as a "black box" and was not included in the sensitivity study used to validate performance criteria.

The licensee identified that an MPFF had not been evaluated. C-PAL-97-0136 documented that while compiling RPS Health Report statistics it was discovered that the total number of MPFFs was 6.

The six MPFFs were:

Channel "B" TMM Failure, C-PAL-95-1636
Channel "C" TMM Failure, C-PAL-94-1050
Failure of RPS to trip on Containment High Pressure, C-PAL-95-1117

Channel "A" TMM Failure, C-PAL-96-0061
Failure of I/I-0010C, C-PAL-96-0897
Failure of I/I-0010D, C-PAL-96-1134

The inspectors identified that the failure of I/I-0010B, C-PAL-95-1056 had not been reclassified as an MPFF from a FF after the two additional failures of this subsystem. Five MPFFs had been previously classified as MR (a)(1). As a result of the listed failures, 3 subsystems of the RPS had been included in a(1):

- Nuclear/Delta T Power Comparitors due to repeat failures;
- Thermal Margin Monitors due to repeat lockups;
- RPS VHPT reset pushbutton isolators due to repetitive failures.

The inspector reviewed the corrective action for these failures, and the goals and monitoring under the (a)(1) status, and concluded that the corrective action, goals, and monitoring were appropriate for those subsystems moved to a(1). Although 7 MPFFs had been identified, the licensee had justified not moving the entire RPS system to (a)(1) because of the subsystems of RPS were moved to (a)(1). The team was concerned that three subsystems classified as (a)(1) could be indicative of overall system problems and raised this issue with the licensee staff. The licensee informed the team that CR C-PAL-96-0136 had been written on January 31, 1997, to evaluate reclassifying the entire RPS as category (a)(1). Because this evaluation was not completed at the close of the inspection, it will be followed as an IFI (50-255/97003-03)

c.8 Conclusions for the Reactor Protection System

The team concluded that three subsystems of the RPS being classified as (a)(1) were indicative of overall system problems and that an evaluation to determine if the entire system should be classified as (a)(1) was appropriate. The team also identified additional examples of untimely processing of MPFF evaluations and inaccurate administration of reliability criteria.

M2.2 Material Condition

a. Inspection Scope

In the course of verifying the implementation of the MR using Inspection Procedure 62706, the team performed walkdowns using Inspection Procedure 71707 to examine the material condition of the systems listed in Section M1.6.

b. Observations and Findings

The team performed material condition walkdowns on selected portions of each system that related to the areas inspected. Housekeeping in the general areas around system and components was adequate, with some minor examples of trash, debris, and tools in work areas. Some indications of corrosion, oil leaks, or water leaks were evident. Examination of control panels, distribution panels, and motor control centers showed no signs of extraneous material.

During the inspection, the licensee reduced power to take additional corrective for an ongoing problem with the main generator isophase bus. Recurrent problems had forced the licensee to reduce power or shut the plant down on three previous occasions.

c. Conclusions

Walkdowns of plant systems, in conjunction with review of deficiency lists, revealed a sizable number of material condition problems. This was consistent with the (a)(1) classification of components in the PCS, HPI, IAS, and TGS.

M7 Quality Assurance In Maintenance Activities (40500)

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

a. Inspection Scope

The team reviewed the licensee's self-assessments to determine if MR independent evaluations were conducted and the findings of the audits were addressed.

b. Observations and Findings

The team reviewed three self-assessment reports:

- NEI assist team assessment conducted on October 3, 1995, through October 5, 1995;
- Palisades Maintenance Rule Implementation Surveillance, NPAD/P-95-018, conducted July 17 through July 21, 1995, and
- Palisades 10 CFR 50.65 (Maintenance Rule) Compliance Program Surveillance, NPAD/P-96-011.

The overall quality of the audits was good. The audits were detailed, and the issues identified were clearly described. NPAD/P-95-018 was performed early in the implementation schedule and identified a number of significant weaknesses in the general program, scoping, performance monitoring, and on-line scheduling. These were appropriately addressed by the licensee's MR staff. NPAD/96-011 identified issues in the areas of structure monitoring, procedure status, availability tracking, and aging of the PSA. The team noted that action on the structure monitoring finding had been completed and that action on the remaining issues was in process.

c. Conclusions

The team concluded the NPAD surveillances were detailed and thorough. The issues were addressed in a timely manner; corrective actions to the surveillance findings were completed or in progress.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Review of Updated Final Safety Analysis Report (UFSAR) Commitments (62706)

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the inspections discussed in this report, the team reviewed the applicable portions of the UFSAR that related to the areas inspected. The team verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

E4 Engineering Staff Knowledge and Performance (62706)

E4.1 Engineer's Knowledge of the Maintenance Rule

a. Inspection Scope (62706)

The team interviewed system engineers (SE) and managers to assess their understanding of PSA, the MR, and associated responsibilities.

b. Observations and Findings

The team interviewed the SEs assigned responsibility for selected SSCs, walked down systems with them, and determined that they were knowledgeable of their systems. During the implementation phase, the licensee conducted several intensive training sessions on the MR, and also developed a MR qualification card for the SEs. Despite this training effort, SEs did not display a uniform understanding of the MR or the Palisades program. During the interviews and walkdowns, the team noted that some of the SEs were not familiar with the MR performance criteria for their systems, the basis for the criteria, or where their systems stood with respect to the performance criteria. The team identified cases where SEs were untimely in evaluating events for FF or MPFF and were inaccurate in the administration of the reliability criteria. Also, as discussed above in Paragraphs M2.1.b.7 and b.8, the team identified that some SEs did not understand the need to identify and record FFs when the system was not required to be in service.

c. Conclusions

System engineers had been trained and appeared qualified to provide oversight of the implementation of the rule for their respective SSCs. The licensee's development of a MR qualification card for system engineers was good. However, some SEs were unfamiliar with performance criteria, and were not timely in classifying events as Maintenance Preventable Functional Failures or in recognizing that performance criteria had been exceeded.

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 February 27, 1997 and on March 20, 1997. The licensee acknowledged the findings presented.

The team asked the licensee whether any materials examined during the inspection should be considered proprietary; a contracted evaluation was identified.

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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

50-255/97003-01a(DRS)	IFI	Periodic Assessment
50-255/97003-01b(DRS)	IFI	Balancing Reliability/Availability
50-255/97003-01c(DRS)	IFI	Industry Operating Experience
50-255/97003-02(DRS)	IFI	Reactor Protection System Reliability Criteria
50-255/97003-03(DRS)	IFI	Reactor Protection System Overall Classification

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Reports
CSW	Critical Service Water
CSWP	Critical Service Water Pump
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
CCW	Component Cooling Water
DRS	Division of Reactor Safety
ELU	Emergency Lighting Unit
EOOS	Equipment Out of Service
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
FF	Functional Failure
FW	Feedwater
HPI	High Pressure Safety Injection
IAS	Instrument Air System
IFI	Inspection Follow-up Item
INPO	Institute of Nuclear Plant Operations
IOE	Industry Operating Experience
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Examination of External Events
MPFF	Maintenance Preventable Functional Failure
MR	Maintenance Rule
MRE	Maintenance Rule Engineer
MSIV	Main Steam Isolation Valve
MVA	Maintenance of Vital Auxiliaries
NPAD	Nuclear Performance Assessment Department
NUMARC	Nuclear Management Resource Council
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PCS	Primary Coolant System
PSA	Probabilistic Safety Assessment
QA	Quality Assurance
RAW	Risk Achievement Worth
RO	Reactor Operator
RPS	Reactor Protection System
RRW	Risk Reduction Worth
S&L	Sargent and Lundy
SE	System Engineer
SER	Safety Evaluation Report
SHA	System Health Assessment
SRO	Senior Reactor Operator
SSC	Structures, Systems or Components
TGS	Turbine Generator System

LIST OF DOCUMENTS REVIEWED

1. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 0, May 1993.
2. Palisades Individual Plant Examination (IPE), Volumes 1, 2, and 3, November 1992, submitted January 29, 1993.
3. Palisades Individual Plant Examination (IPE) Additional Information, July 22, 1994.
4. NRC letter 02/07/96 from M. Gamberoni to R. W. Smedley, Consumers Power with Staff Evaluation Report of Palisades Plant Examination Submittal and Appendices A, B, and C.
5. Palisades Nuclear Plant Individual Plant Examination of External Events (IPEEE), June 1995.
6. Administrative Procedure No. 9.03, Revision 15, 05/30/96, "Facility Change."
7. EPRI-TR-105396, "PSA Applications Guide," August 1995.
8. Palisades Plant Surveillance Report, NPAD/P-96-011, 09/13/96, "Palisades 10 CFR 50.65 Maintenance Rule Compliance Program."
9. Palisades Maintenance Rule Expert Panel Meeting Minutes - June 14, 1996.
10. Palisades Maintenance Rule Expert Panel Meeting Minutes - Sept. 26, 1996.
11. Palisades Maintenance Rule Expert Panel Meeting Minutes - Jan. 27, 1997.
12. Palisades "Maintenance Rule Scoping Document," Revision 2.
13. Palisades Engineering Manual Procedure EM-25, "Maintenance Rule Program," Revision 0, 01/20/97.
14. "Calculation of System Importance Measures"
15. "Calculation of Component Importance Measures"
16. "Disposition of Basic Events"
17. "Palisades Maintenance Rule Quantitative Evaluation of Performance Criteria."
18. Palisades "Maintenance Rule Performance Monitoring Results," 01/22/97.
19. Palisades General Operating Procedure GOP-14, "Shutdown Cooling Operations," Revision 7, 10/16/96.
20. Palisades "Shutdown Operations Protection Plan 96-5090 Refout," 10/07/96.

LIST OF DOCUMENTS REVIEWED (cont'd)

21. Palisades Administrative Procedure No. 4.02, "Control of Equipment." Revision 13, effective date 10/10/96.

22. "Palisades 13 Week Schedule Program - Planning & Scheduling Guidelines 2.0," July 1, 1996, Revision C.

23. Off Normal Procedure, ONP-25.1, "Fire Which Threatens Safety-Related Equipment," Revision 8, 11/13/96.

24. Off Normal Procedure, ONP-25.2, "Alternate Safe Shutdown Procedure," Revision 12, 12/13/96.