U. S. NUCLEAR REGULATORY COMMISSION

REGION III

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Licensee:	Consumers Energy Company
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
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EXECUTIVE SUMMARY

Palisades Nuclear Generating Plant
NRC Inspection Report 50-255/98012

The purpose of the inspection was to evaluate the effectiveness of the engineering and technical support (E&TS) organization in their performance of routine and reactive site activities including identification and resolution of technical issues and problems. The inspection focused on system engineering functions, modifications, technical problem resolution, and engineering support to other plant organizations. The criteria use to assess E&TS performance was understanding of plant design and involvement in preventing and solving plant problems. In addition, the effectiveness of the corrective action program in identifying, resolving, and preventing problems was evaluated.

Engineering

- Overall, for the 51 CRs reviewed, the corrective actions taken were good and root cause determinations were effective. The team also noted that a low threshold existed for identifying problems and issuance of condition reports. However, two minor examples were noted where corrective actions could have been improved (Section E1.1).

- Although the need for testing of molded case circuit breakers had been licensee identified in 1993, from review of industry operating experience information, a testing program was not developed until 1997, after 44 of 72 molded case circuit breakers failed to trip during testing. The failure to assure that this condition adverse to quality was promptly identified and corrected was considered a violation (Section E2.1).

- Surveillance Test Procedure MO-7A-1 for the diesel generator went beyond the specific warning contained in IN 97-16 to assure that any adverse condition found concerning liquid in the cylinders would be formally documented and evaluated (Section E2.1).

- Overall, the 10 CFR 50.59 screenings and safety evaluations reviewed for the past two years were of good quality and a good program had been established for ensuring that trained and qualified personnel prepared and reviewed 50.59 screenings and safety evaluations (Section E3.1).

- The team reviewed 24 modifications and nine temporary modifications and concluded that they were of good quality, properly installed and tested (Section E3.2).

- Based on interviews with station personnel and review of corrective action documents, the team concluded the licensee's corrective action, audit, and self-assessment programs were effective. The team concluded that quality assurance activities were of appropriate depth and scope (Section E7.1).

- The corrective action program at Palisades had shown improvements in identification, resolution, and prevention of problems in the past two years. Personnel interviewed indicated a willingness to identify problems, considered the process to be owned equally by all plant staff, and did not consider CRs written against themselves to be negative. Overall, the licensee has been effective in the identification and resolution of problems (Section E7.2).

- The program for screening, analyzing and dispositioning industry experience issues appeared to be effective; however, the team noted two examples where Engineering concluded that concerns were not applicable to Palisades because the conditions were not precisely the same as those at Palisades, rather than taking the broader view of how and where there were similarities (Section E2.1 and E7.3).

- The team concluded that the Self-Assessment Program was effective and capable of providing valuable performance insights. The team also found that the audit program covered the required areas and was identifying problems and concerns. Audit findings were documented on condition reports, which were used for tracking and to obtain corrective actions. Areas of concern identified by audit findings were promptly and effectively corrected (Section E7.4).

Report Details

III. Engineering

E1 Conduct of Engineering

E1.1 Problem Identification and Root Cause Determination

a. Inspection Scope (37550)

The team reviewed 51 condition reports (CRs) and verified whether the CRs were properly evaluated for root cause determination and corrective actions. Documents reviewed are listed at the end of this report.

b. Observations and Findings

Overall, for the sample reviewed, the corrective actions taken for CRs were good and root cause determinations were effective in determining root causes. The team noted that a low threshold existed for identifying problems and issuance of CRs. However, the team had concerns with the disposition and corrective actions for the following CRs.

During the NRC architect/engineer (A/E) design inspection (Inspection Report No. 50-255/97201) CR No. C-PAL-97-1568 was issued which noted that overcurrent relays for breakers 152-105 and 152-106 had not been calibrated since 1992. These relays were required to be calibrated in 1995, but the scheduled test was missed. The corrective actions for the CR committed to calibrating the relays during the 1998 refueling outage (April 24 - June 7) concluding that this was acceptable because past calibration records indicated no history of problems. Subsequently, a violation was issued in the A/E inspection follow up (Inspection Report No. 50-255/98003) for failure to test these relays. The June 24, 1998, response to this violation committed to test these relays by December 31, 1998; during the refueling outage, a decision was made to delay their calibration until then. While this was consistent with the violation commitment, no consideration was given to the prior commitment made in CR 97-1568 to test the relays during the refueling outage.

CR No. C-PAL-97-1112 documented a torque wrench lost in August 1997. The CR was closed on September 3, 1997 with the conclusion stating "no equipment operability was affected." The team noted that without as-found data for the torque wrench, the evaluation could not conclude that no equipment operability was affected. The procedure that controlled measuring and test equipment (M&TE), required prompt notification should an M&TE item be nonconforming and defined lost M&TE to be considered potentially nonconforming. However, no nonconformance evaluation was initiated. In March 1998, the torque wrench was found and testing determined that it was in tolerance. The team determined that the significance of the lost torque wrench was low because it was found to be in tolerance; however, the initial disposition of the CR was without basis.

c. Conclusions

Overall, for the 51 CRs reviewed, the corrective actions taken were good and root cause determinations were effective. The team also noted that a low threshold existed for identifying problems and issuance of CRs. However, two minor examples were noted where corrective actions could have been improved.

E2 Engineering Support of Facilities and Equipment

E2.1 Licensee Response to Industry Initiatives

a. Inspection Scope (37550)

The inspection scope included review of responses to notifications of issues of generic interest, particularly NRC originated notifications, such as Information Notices (INs) and Generic Letters (GLs). Documents reviewed are listed at the end of this report. Review of the licensee's Industry Experience Program is documented in Section E7.2 of this report.

b. Observations and Findings

1. Molded Case Circuit Breaker (MCCB) Failures

On April 7, 1993, the NRC issued IN 93-26, "Grease Solidification Causes Molded Case Circuit Breaker Failure To Close," which discussed General Electric MCCBs that failed to close because of grease solidification. A Palisades industry experience traveler was issued; however, its disposition stated that the MCCBs were not greased, were not disassembled, and the grease in them was per vendor requirements. Therefore, no further action was taken at this time. However, on April 28, 1993, action item record A-PAL-93-017 was issued to evaluate and develop a program for periodic testing of safety-related and non-safety-related MCCBs. On August 12, 1993, the NRC issued IN 93-64, "Periodic Testing and Preventative Maintenance of Molded Case Circuit Breakers," which discussed numerous failures with Westinghouse MCCBs due mainly to age related degradation and infrequent exercising of MCCBs, some of which may not have been exercised since initial plant startup.

On November 15, 1996, in response to corrective actions for licensee event report (LER) 96005, "Appendix R Enhancement Analysis - DC Panels Breaker/Fuse Coordination Issue," the licensee tested four MCCBs and all four failed to trip on overcurrent. The licensee determined that there was a potential that failure of a downstream breaker could result in the loss of an entire DC distribution panel. Therefore, failure of the four MCCBs resulted in a concern that all DC MCCBs installed in the distribution panels, 72 in total, could fail to trip when subjected to a short circuit. These 72 MCCBs were tested and all magnetic-only MCCBs, 44 in total, would not trip under high fault currents. The remaining 28 thermal-magnetic MCCBs passed the as-found testing within

specifications. The root causes for the failure to trip were determined to be hardening of breaker lubrication which restricted movement of breaker internal components and an oversight by the plant's preventative maintenance program by not identifying the degraded breaker conditions. In addition, discussions with the licensee indicated that, in general, the MCCBs had not been exercised since initial plant startup. LER 96013, "DC Breaker Failure During Testing For As-Found Trip Setting," and a supplement were issued documenting the MCCB failures. Unresolved item (URI) 50-255/96018-01 was issued by the NRC to track this issue.

The licensee replaced all 72 breakers and established a preventative maintenance program on December 1, 1997, to test approximately a third of the breakers every refueling outage so that all the breakers would be tested in a six year period. The team considered these corrective actions were acceptable and comprehensive. Failure to promptly evaluate a previously identified generic problem with MCCBs, a condition adverse to quality, and to develop a program for their periodic testing in a timely manner was a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" (50-255/98012-01). However, Inspection Report No. 50-255/94002 had already adequately addressed on the docket corrective actions taken and actions to prevent recurrence for a significant breakdown in the control of the Palisades corrective action program which occurred in the same time frame as this violation. Therefore, no response to this violation is required.

2. Analyses of Safety-Related ECCS Pump NPSH

The team reviewed the response to GL 97-04, "Assurance of Sufficient Net Positive Suction Head [NPSH] for Emergency Core Cooling and Containment Heat Removal Pumps," and its NPSH calculation, EA-A-PAL-96-003, "ECCS Evaluation in Post-RAS Recirculation Modes Using Pipe-Flo," Revision 1. The team also reviewed sixteen generic communications relative to emergency core cooling systems (ECCSs) and NPSH. Eleven of these sixteen communications, including GL 97-04, discussed the potential for debris inside containment to be transported to the pumps' suction screens post loss-of-coolant-accident (LOCA), thereby reducing available NPSH below acceptable limits. GL 97-04 required licensees to provide the NRC with detailed information regarding their NPSH analyses, including the methodology for calculating strainer head losses and the required versus the available NPSH ($NPSH_R$ versus $NPSH_A$ respectively). The team identified two concerns. First, per the licensee's response letter, containment spray pump P-54A and high pressure safety injection pump P-66A had inadequate available NPSHs (3.5 feet and 3.3 feet less than the required NPSH respectively). The response letter stated that this condition was acceptable; however, there was insufficient information in the response to substantiate this conclusion. Second, the supporting analysis contained several discrepancies.

The GL 97-04 response stated that the inadequate NPSH condition was acceptable since it would exist for only a short period during pump suction switchover at the initiation of the accident recirculation phase, until the operators placed the subcooling lineup in service (up to 30 minutes). It also stated that operation with this degree of NPSH deficit for this period of time had been approved by the pump manufacturers. However, no documentation could be provided that the containment spray pump manufacturer had ever given this approval. As a result of the team's inquiries, the licensee contacted the containment spray pump vendor who provided approval to operate the pump in this mode for this amount of time, and the licensee performed additional evaluations that indicated that, even with degraded flow, the design basis flows would be provided. However, these evaluations were based on EA-A-PAL-96-003 and the team identified the following concerns during review of this calculation:

- No allowance was made for air entrainment. Both NUREG-0897, "Containment Emergency Sump Performance," Revision 1, and Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 2, provided formulae to correct upward NPSH required under these conditions. A licensee letter to the NRC dated July 9, 1982, concerning Unresolved Safety Issue A-43, "Containment Emergency Sump Reliability," documented that the level of air ingestion that could be expected would be approximately 2%, a level which would have produced a small increase in the required NPSH.

- Although RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," Revision 0, indicated an assumed blockage of 50% should be used in sump screen design, the licensee's analysis assumed only 10%. This assumption was based on a qualitative analysis which stated that the blockage would be very low because LOCA debris generation would be low based on the following: 1) most of the insulation was jacketed or encapsulated; 2) low debris transport flow velocities to the sump; and 3) a torturous path to the sump. However, GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," stated that using even 50% blockage usually will result in a non-conservative analysis for screen blockage effects, and that such assumptions should be replaced with a more comprehensive requirement to assess debris effects on a plant specific basis. It also singled out plants with debris screens less than 100 square feet area as being particularly vulnerable; Palisades' screens are 35 square feet. Further, NUREG-0897, "Containment Emergency Sump Performance," Revision 1, showed that significant LOCA debris generation from fibrous insulation could be expected even with encapsulation, and it could be transported at near neutral buoyancy conditions and very low transport velocities, and would be deposited uniformly across screens. Such deposition would tend to progressively decrease the effective screen

opening size, thereby filtering out increasingly smaller particles and thus significantly increasing the pressure drop over what would be calculated using a simple percentage blockage assumption such as that used by Palisades. The containment at Palisades had three types of fibrous of insulation installed; low-density, molded, encapsulated mineral wool; jacketed, molded calcium silicate (non-jacketed on main steam); and encapsulated fiberglass.

In order to gain an understanding of the magnitude of the blockage that might be expected on the Palisades screens, the team performed an order-of-magnitude calculation using information available since 1985. The debris generation value assumed was taken from NUREG-0897 at the very low end of the values tabulated for a typical PWR, and an assumption was made that only 10% of this debris would be transported to the sump. The calculation was performed using a pressure drop formula also from this NUREG for mineral wool insulation, a fibrous insulation type that produced the least pressure drop. Even with these relatively simplistic assumptions, the pressure drop calculated was 8.6 feet. A comparison with the less than 0.1 feet from the licensee's original calculation and a subsequent licensee calculation performed during the inspection using 50% blockage which yielded approximately 0.5 feet of pressure drop illustrated the apparent non-conservatism of a simple percentage blockage analyses. It was also noted that the licensee's qualitative analysis regarding the low debris transportation velocities considered only the post-LOCA flow rates provided by the various pumps; however, initial relatively high LOCA blowdown flow rates, which would significantly increase the probability of debris transport to the sump, were not considered.

The team also performed an informal structural scoping calculation on the screen supporting structure. This indicated that, at the 8.6 feet differential pressure from the scoping calculation, the load on the screens' structure was quite high, i.e., approaching its material limits. This was an area that also had not been evaluated by the licensee.

At the conclusion of the inspection the licensee generated CR No. C-PAL-98-1408 to address this concern and committed to perform rigorous quantitative analyses of the debris generation, transportation, and containment sump screen loading, and to reassess the resultant pump NPSH conditions as required by GL 97-04. Pending completion of these analyses and subsequent NRC review, this was an Unresolved Item (50-255/98012-02(DRS)).

3. <u>Unanticipated Effect of Ventilation System on Tank Level Indications and Engineering Safety Features Actuation System Setpoint</u>

IN 97-33, "Unanticipated Effect of Ventilation System on Tank Level Indications and Engineering Safety Features Actuation System Setpoint," described a condition where the level indication for the safety-related refueling water storage pool changed as a result of an auxiliary building controlled area ventilation

system starting during a test. This apparent level change resulted from the level instrument reference leg being vented to an area whose pressure changed when the ventilation system started.

The licensee reviewed this document and concluded that, at Palisades, variations of compartment space pressure affecting level instruments was not a concern for two reasons: First, the automatic switching of ECCS pumps to the containment sump suction was actuated by safety injection refueling water tank level conductivity probes, whose signals were independent of pressure. Second, it was stated that the only ventilation system that created either a positive or negative pressure was the control room heating, ventilation, and air conditioning (HVAC) system, and there were no safety-related level instruments located in spaces pressurized by the control room HVAC.

The team questioned if the control room HVAC system was the only ventilation system that created a differential pressure in ventilated spaces. The licensee responded that the control room HVAC was the only emergency ventilation system that affected pressures in ventilated spaces. Since the auxiliary building HVAC system was non-safety-related, the initial response to the IN was correct. The licensee further responded that, since the non-safety-related auxiliary building HVAC system had less capacity than the system described in the IN, the effect on tank levels would be insignificant, and therefore, the initial response was still valid.

The team concluded that the licensee's initial response to this IN and the subsequent reviews in response to team questions were too narrowly focused, and as such may not have detected all potential problems related to GL 97-33 concerns such as the following:

- The evaluation considered only spaces affected by safety-related HVAC systems; it should have considered instrumentation that could be affected by any HVAC system, regardless of its safety classification.

- The evaluations considered only tank level instrumentation; the condition described could potentially affect any pressure instrumentation that could be affected by HVAC systems.

- The evaluations did not consider the various failure scenarios of the auxiliary building HVAC system and how these could potentially create greater-than-normal negative pressure, or even positive pressure, such as, the collective and individual failures of the supply and exhaust fans and various system dampers.

- The evaluations considered that this was a concern only for higher capacity HVAC systems; however, even small capacity systems had the potential to create high differential pressures with low building leakage.

At the conclusion of the inspection the licensee agreed that the initial reviews were not thorough and committed to readdress this question taking a broader view and a more rigorous evaluation of potential problem areas. Resolution of this concern was identified as an Inspector Follow up Item (IFI) (50-255/98012-03(DRS)).

4. Potential for Ground-Level Radiation Release

During the review of the licensee's evaluation of IN 97-033 described above, an additional concern was identified with the design and operation of the auxiliary building HVAC.

The auxiliary building HVAC system is a non-safety-related system. During a LOCA, if offsite power and control air remained available, it would continue to operate normally, maintaining the auxiliary building at a slightly negative pressure exhausting to the environment through the plant vent stack. If high radiation was detected, the supply fan would be automatically tripped, potentially increasing the negative pressure, but the release would still be through the vent stack. The final safety analysis report (FSAR) values for accident offsite doses strongly indicated that a system operating in this manner had been assumed in the accident dose calculations. However, for a LOCA with a loss-of-offsite-power (LOOP), the system would cease to operate, allowing the building negative pressure to be lost, thus creating the potential for ground-level releases. This would substantially increase the offsite and control room doses.

Additionally, for certain failure scenarios, such as loss of control air pressure (the instrument air system was also non-safety-related), all air-operated system dampers would reposition closed, thereby pressurizing some building areas and ductwork sections relative to the outside environment. Individual failures of certain dampers could also cause even higher building pressurization. Pending licensee analysis of these scenarios and subsequent NRC review, this was identified as an Inspection Follow up Item (50-255/98012-04(DRS)).

5. Diesel Generator Operability Evaluations

IN 97-16 "Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing," discussed various unacceptable test preconditioning practices observed in the industry. Such preconditioning could render testing invalid.

Included as an example in the IN was rolling over diesel generators with the cylinder blowdown petcocks open prior to testing to assure that there was no accumulated liquid in the cylinders that could damage the engine. In this case, the NRC concluded that the safety benefit of rolling the diesels outweighed the benefit of testing in the as-found condition. However, if liquid were detected in any of the cylinders, this would call into question the as-found operability of the diesel generator, i.e., would the quantity of liquid found have been sufficient to cause hydraulic lock that could have prevent the diesel from starting or could

have caused damage? In order to properly assess this condition, any liquid detected would have to be quantified and an operability evaluation performed.

The team inquired if the diesel generator surveillance test procedure contained provisions to quantify any liquid detected and to perform an operability evaluation. An excerpt from Surveillance Test Procedure MO-7A-1, Rev 48, "Emergency Diesel Generator 1-1 (K-6A)," Section 5.2.3, was provided which described the rolling of the diesel with the petcocks open. It required that, for any moisture detected greater than a fine mist, the amount and the fluid type were to be recorded, and a CR was to be initiated. By initiating a CR, an operability evaluation would automatically be performed as a part of the CR process.

c. Conclusions

The team concluded that in general, response to industry initiatives was good; however, several cases were identified where the responses could have been more thorough.

Although the need for testing of molded case circuit breakers had been licensee identified in 1993, from review of industry operating experience information, a testing program was not developed until 1997, after 44 of 72 molded case circuit breakers failed to trip during testing. The failure to assure that this condition adverse to quality was promptly identified and corrected was considered a violation.

Two examples were noted where Engineering concluded that the generic concerns were not applicable to Palisades because the conditions were not precisely the same as those at Palisades, rather than taking the broader view of how and where there were similarities.

The team also noted one example of where a response to a generic concern went beyond the immediate issue. Surveillance Test Procedure MO-7A-1 for the EDG went beyond the specific warning contained in IN 97-16 to assure that any adverse condition found concerning liquid in the cylinders would be formally documented and evaluated.

E3 Engineering Procedures and Documentation

E3.1 10 CFR 50.59 Program Review

a. Inspection Scope (37001)

The team reviewed implementation of the 10 CFR 50.59 program including procedures for screening changes, tests, and experiments (CTEs) and preparing safety evaluations; the processes for maintaining records of CTEs, reporting CTEs to the NRC and updating the updated FSAR; and training and qualifications of 10 CFR 50.59 safety evaluation preparers and reviewers. In addition, the team reviewed 10 CFR 50.59 screenings and/or safety evaluations associated with procedure changes and facility changes to the FSAR. Documents reviewed are listed at the end of this report. 10 CFR

50.59 Safety Evaluations were also reviewed during the team's review of Design Change Packages, Modifications, and Temporary Modifications (See Section E3.2 of this report).

b. Observations and Findings

1. 10 CFR 50.59 Program

The plant was performing 10 CFR 50.59 screenings and safety evaluations in accordance with Administrative Procedure (AP) 3.07, Revision 9. The team reviewed the procedure and verified that the guidance in this procedure was in conformance with 10 CFR 50.59. The licensee used the updated FSAR to make 10 CFR 50.59 applicability determinations and included the FSAR Change Request Log, Technical Specifications Change Requests, Design Basis Documents, and the proposed Improved Technical Specifications for the analysis. The program included an independent review of unreviewed safety question evaluations by the Nuclear Performance Assessment-Independent Safety Review Group following approval of 10 CFR 50.59 screenings and safety evaluations by the Safety/Design Review Section.

2. 10 CFR 50.59 Program Reporting Review

10 CFR 50.59(b)(2) requires that licensees submit a report containing a brief description of any CTEs including a summary of the safety evaluation of each. The team's review concluded that the reporting of completed safety evaluations to the NRC were as required by 10 CFR 50.59(b)(2).

3. 10 CFR 50.59 Program Training Review

The team reviewed the materials used in the training course for personnel that prepared and reviewed 10 CFR 50.59 safety evaluations and verified that the information presented in the course was consistent with corporate procedures and NRC guidance. The training course, which included computer based training, appeared to be comprehensive. The inspector noted that in addition to successful completion of the training course, reviewers were required to be a General Engineer/Technologist or above and have at least five years of nuclear industry experience, or to hold a senior reactor operator license with at least ten years nuclear industry experience, and to prepare a satisfactory 10 CFR 50.59 screening or safety evaluation before becoming qualified to review 10 CFR 50.59 safety evaluations. The team concluded that the licensee had an good program for ensuring that trained and qualified personnel prepared and reviewed 10 CFR 50.59 screenings and safety evaluations.

4. 10 CFR 50.59 Safety Evaluation Review

The team reviewed a sample of 10 CFR 50.59 screenings and safety evaluations for the past two years and determined that, overall, the screenings and safety

evaluations were appropriately prepared and were consistent with licensee procedures. In particular, the team determined that appropriate documents were reviewed during the preparation of 10 CFR 50.59 screenings and safety evaluations; the 10 CFR 50.59 screenings and safety evaluations adequately addressed the effects of the proposed changes on plant operations, interactions with other systems and components, any new failure modes, and the effects on accidents and transients; and the 10 CFR 50.59 safety evaluations adequately addressed unreviewed safety question criteria.

c. Conclusions

Overall, the team concluded that the 10 CFR 50.59 screenings and safety evaluations reviewed for the past two years were of good quality. In addition, the team concluded that the licensee had a good program to ensure that trained and qualified personnel prepared and reviewed 10 CFR 50.59 screenings and safety evaluations.

E3.2 Design Change Packages, Modifications and Temporary Modifications

a. Inspection Scope (37550)

The team reviewed a sample of 24 modifications and nine temporary modifications to determine if they were designed, installed, and tested appropriately and whether canceled modifications were adequately justified. The team also reviewed whether temporary modifications were installed and removed in accordance with established procedures. In addition, the team reviewed 10 CFR 50.59 evaluations for the modifications, temporary alterations and procedure changes. Documents reviewed are listed at the end of this report.

b. Observations and Findings

1. Review of Modifications

The team's review of modifications Nos. FES-97-094 and FES-970-095 produced questions regarding the adequacy of testing of two emergency diesel generator (EDG) breaker closure permissive auxiliary relays (one per diesel) 162-107X and 162-213X. The modifications had changed the original relays from Westinghouse type SG to General Electric type HMA relays. These relays functioned to prevent the auto closure of the EDG breaker during an undervoltage condition; however, a test to verify this function could not be located. Subsequent to the on-site portion of the inspection, the licensee provided additional information on this issue as follows:

- The purpose of this relay is to block closing of the EDG breaker for 1.5 seconds during a fast transfer from the Safeguards transformer to the startup transformer. A fast transfer also generates an immediate trip

signal to the EDG breaker. During normal plant operations (diesel not running), if a fast transfer were to occur, the block signal would not be required as the EDG is not running.

- During surveillance testing (diesel running), if a fast transfer were to occur, the EDG would trip and the relay would prevent the breaker from closing until three conditions are met: 1) the breaker from the safeguards transformer must be open; 2) the breaker from the startup transformer must be open; and 3) the EDG must have attained greater than 2000 volts. However, this is not considered a safety function since the EDG is considered inoperable during surveillance testing.

- The original solenoid operated EDG breakers were replaced in 1995 with stored energy breakers. These breakers required approximately 300 cycles or five seconds for their springs to charge after opening or closing. Since this is longer than the 1.5 second block signal, it does away with the any required function of the blocking relay.

CR C-PAL-98-1507, "No Documentation of Functional Testing of Relay 162-107X (162-213X) was generated on August 12, 1998 to evaluate the current requirements for post maintenance testing and operability or full functional tests. The team considered that this issue had been adequately addressed and had no further questions.

2. Review of Temporary Modifications

Temporary Modification No. 98-009 installed a temporary strainer in the spent fuel pool (SFP) tilt pit drain in order to minimize the spread of contamination while working in the tilt pit. No technical concerns were identified with this modification; however, a procedural deficiency was identified as a result of review of this modification.

In reviewing the modification's 10 CFR 50.59 safety evaluation, the team noted that it stated that Off-Normal Procedure (ONP) 23.3, "Loss of Refueling Water Accident," Revision 3, directed the operators that, if level was being lost from the fuel pool, and the south tilt pit gate was not installed, to realign the spent fuel pool cooling system suction to the south pit. The procedure also allowed a system suction lineup to the reactor cavity tilt pit.

The team was concerned that this procedure had no direction to the operators to identify the cause of the level loss before realigning the system suction. If the cause of the level loss was a breach in the SFP cooling system pressure boundary, this realignment would allow loss of inventory down to the sill of the SFP gate rather than the maximum inventory loss of only approximately four feet if the system suction were left at its normal point near the pool surface. With the water level at the gate sill, in addition to normal cooling being lost, there would be only approximately one foot of water above the top of the fuel assemblies

(calculated fuel pool area radiation level - 893 Rem/hour), for the design basis pool heat load there would be only slightly more than one hour until the pool would boil and approximately one more hour before fuel assemblies would begin to be uncovered due to evaporative losses.

The licensee acknowledged the weaknesses in this procedure and offered the fact that when it was written, the only failures envisioned that could cause such water losses were a reactor cavity seal failure or a steam generator nozzle dam failure, in which case switching the suction point would not matter. However, since the procedure, as written, would cover level loss due to any cause, its structure as found was inadequate to cover any cause. Pending NRC review of licensee action on this procedural weakness, this was considered an Inspector Follow up Item (50-255/98012-05(DRS)).

c. Conclusions

The team reviewed 24 modifications and nine temporary modifications and concluded that they were of good quality, properly installed and tested.

E7 Quality Assurance in Engineering Activities

E7.1 General Comments

a. Inspection Scope (40500)

The inspectors reviewed the licensee's assessment activities to evaluate the effectiveness of licensee controls in identifying, resolving, and preventing issues that degrade the quality of plant operations or safety. These controls included the corrective action and self-assessment programs, implementation of timely and effective resolution of technical issues, active involvement in ensuring the reliability of plant systems, and awareness of industry events and how they impact the plant.

The inspectors selected a sample of issues/problems for detailed analysis to assess the licensee's ability to identify and correct problems. Additionally, the inspectors evaluated the process for initial identification and characterization of the specific problems, elevation of the problems to proper levels of management for resolution, disposition of any operability/reportability issues and implementation of corrective actions, including evaluation of repetitive conditions. Items reviewed included:

(1) Deficiencies requiring safety evaluations, root cause assessments or operability determinations.

(2) QA audits and self-assessments.

(3) Deficiencies tracked in the corrective action programs, including the evaluation of deferred items, or interim resolutions.

(4) Results of licensee audits that evaluated the effectiveness of the associated corrective action programs.

(5) Interviews with selected individuals involved with the problem identification process to determine the extent of the individual's understanding of the process and willingness to report problems.

The documents reviewed are listed at the back of this report.

b. Observations and Findings

The inspectors reviewed a number of licensee documents which focused on compliance of the plant departments with applicable requirements of the corrective action process, audits, and self assessments. The inspectors noted that these documents identified both strengths and areas requiring improvements. The team noted that the level of detail in the reports indicated an in-depth review of the issues.

c. Conclusions

Based on interviews with station personnel and review of the above documents which indicated that problems were being identified and corrective actions for those problems were being implemented, the inspectors concluded the licensee's corrective action, audit, industry experience and self-assessment programs were effective. The inspectors considered that quality assurance activities were of appropriate depth and scope.

E7.2 Corrective Action Process

a. Inspection Scope (40500)

The team assessed the Corrective Action Process (CAP) through review of implementing procedures, CRs, corrective action management reports, corrective action effectiveness reviews, Corrective Action Review Board (CARB) and Condition Review Group (CRG) activities, and action taken for previously identified trends. Documents reviewed are listed at the end of this report. The team also attended one CARB meeting and one Plant Review Committee meeting during the on-site inspection period and interviewed cognizant personnel concerning the corrective action and CR processes. In addition, the team assessed corrective actions taken for problems previously identified in resident reports.

b. Observations and Findings

The team reviewed Procedure 3.03, "Corrective Action Process," Revision 20, which described the methods used for documenting problems and the corrective action process. This procedure described the use of the CR for problem identification and tracking and indicated that a CR would be categorized as level 1, 2, 3 or 4 based on the importance and priority of the problem. Level 1 CRs were used to document the most

significant problems, and Level 2, 3, and 4 CRs, those problems of decreasing importance and priority. Problems documented on Level 4 CRs did not require cause investigations and actions to prevent recurrence.

The team noted that during two Joint Utility Management Audits (JUMA) conducted in 1996 and 1997, the licensee identified that improvements in managing the CR process were necessary. Other issues identified during these audits were that internal audit reports lacked definitive support for some findings; the backlog of low significance CRs was high; and that feedback to the CR initiator after a CR had been closed out was weak. The team concluded that the licensee's response to these issues was prompt and effective.

The team noted that in response to a JUMA audit finding relative to the high backlog of low significant CRs, the licensee established new expectations and revised the applicable plant procedures to reflect these expectations. The expectations for closure of Level 3 and 4 CRs were 30 to 40 days. The team noted that since 1996, close out times for Level 4 CRs was reduced from 105 to 61 days and for Level 3 CRs from 151 to 122 days. The team also noted that the threshold for identifying problems via CRs was considered low and the number of CRs generated was high. Of the sample of CRs reviewed by the team, the root cause analyses appeared to be thorough and effective. When corrective actions were extended, due dates were appropriately extended.

The team also reviewed a Corrective Action Log, dated February 11, 1998, which listed condition reports issued from October 1997 to March, 1998. The list contained 775 CRs which had been written during this period. The number of CRs written and a cursory review of the type of problems documented indicated that the threshold for writing CRs was appropriately low. A listing of open CRs, which included scheduled completion dates, was also reviewed. None of the listed CRs indicated a major problem with the completion date assignments.

The team reviewed the licensee's Correction Action Review Board (CARB) charter and noted that the CARB had been established to review CRs which had been categorized as Level 1 or 2. The duties of the CARB included evaluating the appropriateness of operability and reportability determinations and assuring that appropriate immediate corrective actions were taken to resolve these important matters.

On July 7, 1998, the team attended a meeting of the Condition Review Group (CRG) to observe their review of CRs issued the previous day. The discussions during this meeting appeared to be appropriate and individuals were assigned to follow and expedite required actions. The CRG appeared to be a valuable tool to ensure prompt and thorough management review of significant problems. A total of 9 condition reports were reviewed/evaluated during the CRG meeting. While none of these CRs reached the level that required CARB reviews, the CRG appeared to be effective in its oversight of CR prioritizations and corrective action recommendations.

The team also observed several noteworthy practices that contributed to the effectiveness of the corrective action process. Palisades employed trending at all

levels, including monthly corrective action management reports and periodic NPAD audits of corrective action effectiveness. Other examples included daily reviews of new CRs, the review of existing significant CRs, generation of quarterly reports of performance indicators, and senior management involvement in all Level 1 and 2 CRs.

The team observed that the corrective action process at Palisades had improved since enhancements to the corrective action process were implemented in 1996 and 1997. Problems were identified via the CR process, the more significant issues were investigated for root causes, trends were identified and tracked, significant corrective actions received interdisciplinary review through the CARB, observations were made in the field to improve problem prevention, and the overall collective significance of issues and trends was assessed quarterly. A review of the past two-year period indicated that corrective action process improvements at Palisades have been effective.

c. Conclusions

The corrective action program at Palisades had shown improvements in identification, resolution, and prevention of problems in the past two years. Personnel interviewed indicated a willingness to identify problems, considered the process to be owned equally by all plant staff, and did not consider CRs written against themselves to be negative. Overall, the licensee has been effective in the identification and resolution of problems.

E7.3 Industry Experience Program

a. Inspection Scope (40500)

The team evaluated the adequacy of the licensee's programs that implement industry experience information. Documents reviewed are listed at the end of this report.

b. Observations and Findings

The team reviewed AP 3.16, "Industry Experience Review Program," Revision 5, and noted that the procedure provided adequate guidance for ensuring that industry operating experience was integrated into the plant operating, engineering and maintenance activities. The team observed that CRs documented that system engineers had analyzed events and information at other plants for applicability to Palisades. The team also observed that the quarterly system health assessments incorporated the industry operating experiences.

The team noted that the Industry Experience (IE) group under the leadership of the Industry Experience Coordinator provided the initial screening of all industry experience documents. As possible relevant subjects applicable to Palisades were identified, the IE group coordinated a review/evaluation effort and assessed the adequacy of these reviews/evaluations.

The team observed an IE group daily meeting to discuss industry current news and the status of on-going industry experience evaluations and the backlog of these evaluations.

The team was informed that on a weekly basis, the IE group discussed on-going industry issues at the Managers' Morning Meeting.

Additionally, the licensee informed the team that they were addressing a number of issues identified with the IE program. These issues were: 1) Palisades was not internally using IE information effectively to prevent occurrences or improve the process, 2) Palisades was not always consistent or timely in reporting events to the industry data bank and 3) Self-Assessments of the IE program were not sufficiently critical to identify/quantify the Program's effectiveness. The team noted that the licensee's on-going responses to these issues appeared to be adequate.

The team reviewed several CRs and noted that industry experiences were being properly evaluated and addressed. For example, the team reviewed condition report C-PAL-98-0700 titled, "Parker Fittings Used with Swagelok Fittings," dated April 26, 1998 that correctly identified and evaluated the root cause for this issues. Specifically, the licensee accurately noted that the issue of interchanging hardware from different manufacturers may potentially damage the fittings. However, the team also identified examples of less than adequate response to industry initiatives (See Section E2 of this report).

c. Conclusions

The team concluded that the licensee's program for screening, analyzing and dispositioning industry experience issues appeared to be effective; however, two examples of where responses could have been better were identified during the engineering review.

E7.4 Self-Assessment and Audit Activities

a. Inspection Scope (40500)

The team evaluated the effectiveness of the licensee's self-assessment capability by reviewing department self-assessment reports, quality and self-assessment (Q&SA) quarterly self-assessment reports, and Q&SA audits. In addition, the team interviewed cognizant personnel. Documents reviewed are listed at the end of this report.

b. Observations and Findings

The team reviewed Procedure No. 1.09, "Self-Assessment," Revision 4, which was the implementing procedure for the self-assessment program. The procedure was revised to provide better guidance regarding training of self-assessment evaluators, management's expectations for the respective plant departments, assessment criteria and assessment objectives. The team concluded that the new procedure appeared to be more rigid on how self-assessments should be conducted which should provide further improvements in the Palisades Self-Assessment Program.

The team sampled a number of self-assessment activities conducted by the various plant departments over the previous year and concluded that improvements were being made in this area. For example, the Maintenance, Planning and Scheduling, Operations and Engineering Departments were all generating good quality self-assessment reports. These reports contained the purpose and scope for the assessment, personnel assignments, standards/expectations assessed against, facts supporting where deviations from standards or expectations existed, conclusions based on identified facts, and recommendations for performance improvements.

The team's initial observation of several Operations self-assessments suggested that the assessments did not provide enough details of the identified problem or what corrective actions were needed. Further review of this matter revealed that the licensee had self-identified these concerns and had initiated the appropriate corrective actions. The team verified that these issues were addressed in the department's self-assessment report dated March 23, 1998.

The team also noted the following strengths and improvement initiatives related to the Self-Assessment Program: A number of plant management personnel had received valuable experience during rotational tours of duty at INPO; Engineering Aid Administrative Procedure EGAD-ADM-08, "Guidelines for Performing Self-Assessments," was in the process of being issued; "Hot Button" reports were being utilized to identify adverse trends; and Operations had implemented detailed guidance for Management Monitoring of Supervisory Skills.

The team reviewed the licensee's audit logs and schedules and found that they adequately covered the appropriate plant activities. Records of selected audits that were reviewed indicated that the audits were adequately performed.

c. Conclusion

The team concluded that the Self-Assessment Program was effective and capable of providing valuable performance insights. The team also found that the audit program covered the required areas and was identifying problems and concerns. Audit findings were documented on condition reports, which were used for tracking and to obtain corrective actions. Areas of concern identified by audit findings were promptly and effectively corrected.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Violation 50-255/94002-01: Five examples of SWSOPI Inadequate corrective action which represented a significant breakdown in control of the corrective action program. Failure to recognize the significance of and to take corrective actions to resolve the single failure vulnerability of ESS pump seal cooling and lubrication heat removal, which could result in eventual ESS pump failure. Failure to take prompt corrective action to incorporate Non-critical Service Water System header isolation valve, CV-1359 onto a leakage test program. Failure to appropriately question and take action to evaluate the seismicity of bent instrument and unistrut supports routed in front

of the CCW HXs. Failure to recognize the significance and take corrective actions to couple the Service Water Inservice Test pump reference values to the Service Water flow balancing test. Failure to take prompt corrective actions to resolve a concern that Service Water System flow verification test T-216 balance flow to the CCW HXs at or very near their required flow rates and did not allow for pump degradation.

The licensee responded to this violation on June 6, 1994, committing to corrective actions in management direction and oversight, upgrades to the corrective action process, and evaluation of the failure to take adequate or prompt corrective actions to the significant conditions identified during the service water operational performance inspection. In addition, corrective actions for the five specific examples cited in the Notice of Violation were addressed.

The team reviewed the corrective actions to the specific examples and Event Report E-PAL-94-012, "Service Water System Operational Inspection (SWSOPI) - Inadequate Corrective Action," which evaluated the broad issue of failure to take adequate or prompt corrective action with a more global perspective. The team considered the corrective actions to be acceptable and comprehensive. This item is closed.

E8.2 (Closed) Violation 50-255/95010-01: Failure to use an updated and controlled wiring list. An uncontrolled wire list was used to implement Facility Design Change FC-888. This modification was initiated because of the high maintenance and obsolescence of the reactor protection system (RPS) trip logic. The licensee sent an old wire list that had not been updated to the manufacturer. This wire list contained 12 incorrect module circuit connections that bypassed the six matrix logic channels for the RPS containment high pressure (CHP) trip function. The errors in the CHP trip logic occurred, in part, because the original plant design did not include a CHP trip. Although Combustion Engineering was aware of the CHP trip, they installed printed circuit jumpers that bypassed the RPS matrix trip logic for containment high pressure on all four independent safety channels (A, B, C and D).

The licensee's corrective actions included revising the administrative procedures to clearly establish the roles of design engineers when implementing design changes or modifications. Changes were also made to the design change program that required that information in vendor manuals, drawings and wiring lists shall be verified and validated prior to its use in design changes. The team verified that the corrective actions were acceptable. This item is closed.

E8.3 (Closed) Violation 50-255/95010-02: The post modification test for Facility Design Change FC-888 was not suitable to verify or check the adequacy of design. During implementation of FC-888, the licensee introduced an inadvertent change to the existing RPS matrix logic that bypassed the CHP trip; however, the post modification test was inadequate because it did not include testing to determine whether any portion of the two out of four (2/4) RPS CHP trip logic was functional. The licensee determined that the inadequate post modification test occurred because of the reliance on the Technical

Specification (TS) surveillance that did not test all combinations of the RPS trip logic. In addition, the licensee concluded that the role of the system engineer was not clearly defined when determining the adequacy of the post modification test.

As part of the corrective actions, the licensee modified the RPS matrix channels to restore the CHP trip function. The TS surveillance test for the RPS matrix logic was revised to provide adequate testing. This assured that the requirements of the TS Table 4.17.1 were being adequately verified during the monthly test. Additionally, the licensee assigned responsibility for review of all modification testing to system engineers who will act as the testing authority. This item is closed.

E8.4 (Closed) Violation 50-255/95010-03: The licensee did not demonstrate operability of the six matrix logic trip unit channels for high containment pressure from April 11, 1992, to May 22, 1995. The surveillance test required by TS 3.17 and as implemented by Procedure No. MO-3, "Reactor Protection Matrix Logic Tests," had not demonstrated operability of the RPS trip logic during this period. This occurred because the 2/4 logic combinations for only 1 of the 11 trips on the RPS logic system was testing at random each month. Consequently, if a channel such as the high power trip, high rate trip or high containment pressure were to become inoperable, surveillance testing would not promptly identify this condition because the testing method was random and only one channel was selected each month.

The team verified that the licensee changed surveillance procedure No. MI-3, "Reactor Protection Logic Tests," Revision 0, so that all 11 channels and each of the 4 RPS logic combinations (A, B, C and D) were checked. The licensee now tests monthly each combination (AB, AC, AD, BC, BD and CD) in the ladder logic to provide adequate overlap in testing for all 11 RPS trips. The team reviewed a completed copy of the last RPS surveillance test and found it to be acceptable. This item is closed.

E8.5 (Closed) IFI 50-255/94014-53: Primary coolant system (PCS) cooled below 70°F temperature limit. As discussed in inspection report 50-355/94014, the PCS was cooled to below the temperature limit of 70°F (21°C) on two occasions with the reactor vessel (RV) head bolts fully tensioned. During this inspection, the team reviewed Engineering Analysis EA-D-PAL-94-170-01, "Impact of Less Than 70°F Shutdown Cooling Water on Reactor Vessel," which concluded that no ASME code limits for the RV had been exceeded. Consequently, there had been no cumulative effects on the ductility or integrity of the RV. This item is closed.

E8.6 (Closed) IFI 50-255/94014-60: Lack of Design Basis Information/No Clear Roles & Responsibilities. The causes for weaknesses included a historical lack of design basis information, lack of clearly defined roles and responsibilities between NECO and System Engineering, ineffective technical reviews, and an ineffective process to assure documents, processes, and activities affected by the modification were appropriately revised.

The team reviewed Palisades Performance Enhancement Plan (PPEP) action items 1.2, 2.4, and 4.2. The Design, Systems, Programs and Plant Support Engineering groups

now report to the Plant Engineering and Modifications Manager. A multi-discipline review and required training and a qualification card process was instituted to address the ineffective technical reviews. Administrative Procedures (AP) 9.00, "Design Engineering and Configuration Management Program Description and AP 9.03, "Facility Change," were rewritten and strengthened in the area of updated of documents and processes affected by modifications. This item is closed.

E8.7 (Closed) IFI 50-255/94014-61: Plant Configuration Control Weakness.

The team reviewed PPEP action items 2.4, 4.2, and 5.1. AP 9.0 was revised to add specific management expectations to specific engineering departments and an overview of the Design Engineering and Configuration Management Process. In addition, the plant now has a dedicated Configuration Control Manager and department. This item is closed.

E8.8 (Closed) LER 50-255/91017: Potential inter-system loss of coolant accident (ISLOCA) with primary coolant pump. As discussed in LER 50-255/91017-00, on August 5, 1991, the licensee identified that a postulated break in the PCP integral heat exchanger could result in a primary coolant system leak outside the containment building. Four identical Byron-Jackson primary coolant pumps are installed at Palisades. The primary coolant at the integral heat exchanger is pressurized to about 2060 psia; the component cooling water (CCW) system has a design pressure of 150 psig. In the event of a postulated ISLOCA, primary coolant would enter the CCW and pressurize the CCW system beyond its design pressure resulting in a potential for leakage path outside the containment building.

The licensee performed a probabilistic risk assessment PRA of this scenario to determine the likelihood of core damage occurring from the postulated event. The initiating event frequency was determined to be 100 times smaller than that of a small break LOCA. The licensee determined that the risk of core damage could be limited to acceptable levels by updating operator training on how to respond to the ISLOCA. The team verified that the scenario was included in the training module for continued licensed operator training in Instructor Lesson Plan LOCT01.93D, "Mitigating Core Damage Review." This item is closed.

E8.9 (Closed) LER 50-255/94003: Lack of environmental qualification for the position switches for the service water inlet and outlet valves to the containment air coolers (CACs). As discussed in LER 50-255/94003-00, on February 3, 1994, the licensee identified that the position switches for the CAC service water inlet and outlet valves were not environmentally qualified for submergence. The valve position switches provide the operator with CAC service water inlet and outlet valve status indication as one of the inputs to verify operability of the containment heat removal system. The licensee performed an operability determination and determined that the service water inlet and outlet valves would be operable in the event of the failure of the position switches. In addition, justification was provided to show that submergence would not have an effect on the switches until after the valve positions were verified and documented. The switches were qualified for submergence for a period of three hours.

This provided ample margin since the switches that change position will be verified approximately 30 minutes after the start of an accident. This item is closed.

E8.10 (Closed) IFI 50-255/93020-01: Evaluation of fuel assembly I-024 failure. As discussed in inspection report 50-255/93020(DRS) the licensee had not completed the root cause analysis in time for the teams assessment and inclusion into the report. On September 30, 1993, the licensee provided a detailed description and the results of the root cause analysis in a letter to the NRC. In addition, on October 14, 1993 the licensee issued a supplemental LER to include a summary of the results of the root cause analysis which was performed to determine the reasons for the failure of fuel assembly I-024 and the subsequent discharge of nuclear fuel into the primary coolant of the Palisades reactor. This supplemental LER was closed in report 50-255/98006(DRS). This item is closed.

E8.11 (Closed) IFI 50-255/94014-63: Fuse control program weaknesses. As discussed in inspection report 50-255/94014(DRP) the Palisades Performance Enhancement Program (PPEP) Action Plans 2.2 and 4.2 required further NRC review and evaluation. Action plans 2.2, "Establish an Improved Planning and Prioritization Process," and 4.2, "Establish Strong Sensitivity to Design Basis," were reviewed and evaluated by the team. In addition, action item A-PAL-94-152, "Calculations For Fuse Size and Type" and guideline EGAD-ELEC-10, "Sizing of Control and Power Fuses," were also reviewed. Palisades Maintenance Standards Handbook uses references from AP 4.02, "Control of Equipment," and EGAD-ELEC-10 to address fuse replacement. This item is closed.

E8.12 (Closed) LER 50-255/93007: Degradation of boraflex neutron absorber in surveillance coupons. As discussed in LER 50-255/93007-02, on August 17, 1993, the licensee identified that a Boraflex surveillance coupon, upon removal form the spent fuel pool (SFP), had disintegrated approximately 90%. Boraflex is the trade name of a boron impregnated, polymer-based sheet material that is utilized as a neutron absorber in the construction of SFP storage racks. The use of Boraflex allows minimal center to center cell spacing in the SFP storage racks.

The cause of the event was determined to be flow induced deterioration of the full length surveillance coupons due to inadequate holding canister design. The corrective actions included completion of neutron attenuation testing (blackness testing) on the SFP racks and a change to the surveillance method for verification of rack Boraflex condition. Blackness testing is the most effective method of boraflex surveillance since it "looks" at the actual rack Boraflex panels and does not rely upon secondary methods such as surveillance coupons. Blackness testing uses a neutron source to verify the presence of Boraflex in the walls of the spent fuel rack. This item is closed.

E8.13 (Closed) LER 50-255/96002-01: Initiation of TS required shutdown due to safeguards cable fault. As discussed in LER 50-255/96002-01, on January 16, 1996, the licensee identified a loss of the safeguards power source caused by a phase-to-phase fault in the 2400V AC safeguards bus. The loss of the safeguards power source placed the plant into a 24 hour TS action statement which led to a plant shutdown.

The cable failure was determined to be caused by localized water and contaminant treeing (treeing is a condition where microscopic voids in cable insulation look like tree branches when a wafer-thin cable insulation sample is viewed under a microscope) which initiated in the cable insulation. The initiating points for the treeing degradation were localized foreign matter (contaminants) and voids found in the cable insulation. Corrective actions included cable replacement, testing of the other cable, and a review of plant equipment response. This item is closed.

E8.14 (Closed) LER 50-255/96013-01: Testing of four molded case circuit breakers (MCCBs) revealed that the breakers would not trip on overcurrent. These failures resulted in concerns that all 72 DC MCCBs could fail to trip when subjected to a short circuit. Eventually all 72 breaker were tested and the licensee found that all 44 magnetic only MCCBs would not trip when subjected short circuit currents. The remaining 28 breakers were of the combination thermal-magnetic type and these breakers tested within the manufacturer requirements. The licensee replaced all 72 breakers and established a preventative maintenance program on December 1, 1997, to test approximately a third of the breakers every refueling outage so that all the breakers would be tested in a six year period. This item is closed (See Section E2.1).

E8.15 (Closed) Unresolved Item 50-255/96018-01: The failure of the MCCBs and related problems was considered unresolved pending the licensee investigation and determination of the extent and significance of the problem. This URI was determined to be a violation in Section E2.1. This item is closed.

X1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of the inspection on July 24, 1998. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any material examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. DePuydt, Design Engineering
R. DesJardins, Design Engineering
R. Gerling, Manager Design Engineering
K. Haas, Engineering Director
N. Haskell, Licensing Director
M. Nordin, Design Engineering
K. Osborne, Engineering Programs
T. Palmisano, Site Vice President
D. Rogers, General Manager - Plant Operations
G. Szczotka, Manager NPAD
K. Toner, Licensing Supervisor
S. Wawro, Director Maintenance and Planning
R. Westerhof, Configuration Control

INSPECTION PROCEDURES USED

IP 37001:	10 CFR 50.59 Safety Evaluation Program
IP 37550:	Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 92700:	Onsite Follow up of Written Reports of Non-Routine Events
IP 92703:	Follow up - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-255/98012-01	VIO	Failure to implement testing of MCCBs in a timely manner.
50-255/98012-02	URI	Licensee to perform rigorous quantitative analyses of the debris generation, transportation, and containment sump screen loading and to reassess resultant pump NPSH conditions
50-255/98012-03	IFI	Licensee to re-evaluate IN 97-33
50-255/98012-04	IFI	Licensee to evaluate the potential for ground-level radiation release
50-255/98012-05	IFI	Licensee to evaluate weakness identified in ONP 23.3, "Loss of Refueling Water Accident"

Closed

50-255/94002-01	VIO	Five examples of SWSOPI Inadequate corrective action which represented a significant breakdown in control of the corrective action program.
50-255/95010-01	VIO	Failure to use an updated and controlled wiring list
50-255/95010-02	VIO	Failure to perform an adequate post modification test
50-255/95010-03	VIO	Failure to implement and perform an adequate surveillance test
50-255/96013-01	LER	DC MCCB Failures
50-255/96018-01	URI	DC MCCB Failures
50-255/94014-53	IFI	PCS cooled below 70°F temperature limit
50-255/94014-60	IFI	Lack of Design Basis Information/No Clear Roles & Responsibilities
50-255/94014-61	IFI	Plant Configuration Control Weakness
50-255/94014-63	IFI	Fuse control program weaknesses.

50-255/94003	LER	Lack of environmental qualification for position switches
50-255/93020-01	IFI	Evaluation of fuel assembly I-024 failure
50-255/93007	LER	Boraflex degradation
50-255/96002-01	LER	Initiation of Technical Specifications (TS) required shutdown due to safeguards cable fault

LIST OF ACRONYMS USED

A/E	Architect Engineer
AC	Alternating Current
AP	Administrative Procedure
CAC	Containment Air Cooler
CAP	Corrective Action Program
CARB	Corrective Action Review Board
CCW	Closed Cooling Water
CFR	Code of Federal Regulations
CHP	Containment High Pressure
CRG	Condition Review Group
CTE	Changes, Tests, Experiments
CR	Condition Report
DC	Direct Current
DRS	Division of Reactor Safety
E&TS	Engineering and Technical Support
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FSAR	Final Safety Analysis Report
GL	Generic Letter
HVAC	Heating, Ventilation, and Air Conditioning
IFI	Inspection Follow Up Item
IN	Information Notice
ISLOC	Intersystem Loss of Coolant Accident
JUMA	Joint Utility Management Audit
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
M&TE	Measuring and Test Equipment
MCCB	Molded Case Circuit Breaker
NPSH	Net Positive Suction Head
PDR	Public Document Room
PPEP	Palisades Performance Enhancement Program
PCS	Primary Coolant System
RPS	Reactor Protection System
RV	Reactor Vessel
SFP	Spent Fuel Pool
TS	Technical Specifications
URI	Unresolved Item
V	Volt
VIO	Violation

PARTIAL LIST OF DOCUMENTS REVIEWED

Procedures

1.09, "Self-Assessment Program," Revision 4
3.03, "Corrective Action Process," Revision 20
3.07, "Safety Evaluations," Revision 9
3.15, "Design Basis Document Maintenance," Revision 6
3.16, "Industry Experience Review Program," Revision 5
4.00, "Operations Organization, Responsibilities and Conduct," Revision 20
5.07, "Control of Measuring and Test Equipment," Revision 8
10.41, "Procedure Initiation and Revision," Revision 27
MI-3, "Reactor Protection Logic Tests," Revision 0
MO-7A-1&2, "Emergency Diesel Generators 1-1 & 1&2," Revision 6
MO-7A-2, "Emergency Diesel Generator 1-2 (K-6B)," Revision 46
SOP-22, "Emergency Diesel Generators," Revision 24
T-SC-95-090-01, "SIS Actuation Logic Modification Test," Revision 1
COP 3 ,"Determination of T-103 Volume to Add to Sump Following a LOCA," Rev 8, Attach 5
SAP 4, "Containment Spray and Iodine Removal System," Revision 8
GOP-2, "Plant Heatup (Cold Shutdown to Hot Shutdown)," Revision 20
SOP-1, "Primary Coolant System," Revision 41
EOP-9.0, "Functional Recovery Procedure," Revision 9
EOP-4.0, "Loss of Coolant Accident," Revision 9
ONP-23.3, "Loss of Refueling Water Accident," Revision 3
EGAD-ADM-08, "Guidance for Performing Self-Assessments," Revision 0

Condition Reports

C-PAL-93-0017, "Develop Program for Periodic Testing of Molded Case Circuit Breakers"
C-PAL-94-0564, "Diesel Generator Failed to Reach Full Load"
C-PAL-96-1453, "72-228 Breaker Failed Lab Services Test"
C-PAL-96-0885, "Pressure Control Valves Out-of-Tolerance"
C-PAL-97-0087, "Main Generator Wire Insulation Damage at Current Transformer"
C-PAL-97-0910, "Breaker 52-2239 Failed During Testing"
C-PAL-97-1112, "Loss of M&TE Control"
C-PAL-97-1261, "Found Breaker 52-277 Out of Tolerance"
C-PAL-97-1430, "Inadequate Implementation of Battery Calculation"
C-PAL-97-1566, "Diesel Generator for Design Base Load Range Incorrect"
C-PAL-97_1568, "Failure to Verify Calibration of 152-105 and 152-106 Breaker Relays"
C-PAL-97-1596, "Station Batteries Unanalyzed for DBA Conditions with Battery Chargers Crosstied"
C-PAL-97-1619, "Electrical Engineering Calculations Were Not Updated"
C-PAL-97-1620, "Lack of Analysis for 125 VDC Loads at Degraded Voltages"
C-PAL-97-1652, "125VDC Distribution Panel Breakers Short Circuit Capability"
C-PAL-97-1656, "DC Load Flow with Incorrect Temperature Correction Factor"
C-PAL-97-1685, "Documentation for Station Battery Short Circuit Current not Available"
C-PAL-98-0724, "Peak Inrush Motor Current Exceeds Motor Locked Rotor Rating"

C-PAL-98-1153, "Test Overcurrent Relays for Breakers 152-105 and 152-106"
C-PAL-98-1358, "Essential Safeguards Room Cooler Breaker Found Tripped"
C-PAL-96-1648, "HPSI Operability in Question Immediately Following RAS Until Subcooling Established"
C-PAL-97-0091, "Main Steam Safety Valve Inlet Line Losses"
C-PAL-97-0582, "Potential Preconditioning of EDGs During MO-7A-1/2"
C-PAL-97-1209, "Less Than adequate Guidance for D/G Room Temperature at Which to Repower the D/G Rooms Ventilation Fans"
C-PAL-97-1210, "Diesel Injection Tube Part 21 Notification"
C-PAL-97-1396, "CCW Pump Discharge Pressure Below SOP16 Requirement"
C-PAL-98-0394, "Lack of Overpressure Protection Vulnerability in HP Air System to ESS Valves"
C-PAL-98-0733, "VHX-2 and VHX-4 Outlet Valves Failed ASME Section XI Testing"
C-PAL-96-0664, "Design and Installation Questions Associated with C-3B, "Diesel Generator 1-2 Air Compressor" Replacement Under FES-95-239"
C-PAL-97-0196, "No Calculations to Support FSAR Functional Statements"
C-PAL-97-1450, "SOP-2A Lacks Adequate Guidance to Adjust Purification Demineralizer D/P"
C-PAL-98-1323, "Potential Lifting of Letdown Relief Valve RV-2013"
C-PAL-97-1438, "PCS Leakage From Letdown Relief Valve"
C-PAL-96-0853, "Purification Ion Exchanger High D/P"
C-PAL-98-0121, "Failed Fuel Monitor Low Flow Condition"
C-PAL-97-0919, "Charging Pump P-55B Seal Lube Return Line Plugged"
C-PAL-97-1027, "Nitrogen Station 3B Relief Valve Failures"
C-PAL-97-1116, "Documentation of Design Function and Operability of Guard Pipe on Containment Sump Suction"
C-PAL-97-1172, "Debris Found in Containment"
C-PAL-97-1370, "CCW Flow to ESS Pumps Not Analyzed for IST Pump Degradation in Calc. Of Record"
C-PAl-97-1499, "Improper Air Pressure Used for Actuator Capability Calculation in T-372"
C-PAL- 96-0883, "Containment Spray Pump & Sump Check Valve Flow Rate Discrepancies"
C-PAL-97-1571, "Potential Flow Paths That Bypass Containment Sump Screens Following a DBA"
C-PAL-98-1408, "Adequacy of ECCS Pump NPSH Under Increased Screen Blockage"
C-PAL-98-0007, "ONP 6.2 (Loss of Component Cooling) Immediate Action not Performed"
C-PAL-98-0032, "ONP 6.2 Inadequate for Loss of Inventory"
C-PAL-98-0573, "Inadequate Job Preparation (FW Bypass Valve Position)"
C-PAL-98-1160, "Steam Generator Level Below SOP-7 Requirement"
C-PAL-98-1316, "Work Order Readiness Review Inadequate"
C-PAL-98-1358, "RCS West safeguards Room Cooling V-27C Breaker Found Tripped"
C-PAL-98-1409. "Manual Reactor Trip due to Loss of "A" Main FW Pump"

Modifications

EAR 96-0123, "Configuration Problem with Breaker 52-8226"
EAR 96-0196, "Temporary Covers for Containment Floor Drains"
EAR 96-0214, "Breaker Replacements"
EAR 96-0264, "15 VDC Power Supply for Comparator/Averager"

EAR 97-0592, "Pressure Switch Model 680 No Longer Available"
EAR 98-0303, "LS-1440 Wiring Problem"
FES 96-059, "Calculation for Second Level Undervoltage Time Relay"
FES 97-092, "Replacement of Load Shedding Relay 194-108 due to SQUG Relay Program"
FES 97-095, "Replacement of Diesel Breaker Auto Close Permissive Relay 162-213X"
FES 97-107, "Replacement of Circuit Breakers on Panel D11A"
FES 98-035, "Replacement Power Supply for Comparator/Averager"
SC 95-090, "Provide Left Channel Safety Injection Signal Actuation"
EAR 98-0008, "Containment Sump Vent Screen"
FES 98-048, "Allow Use of HeliCoil Inserts in Steam Generator Secondary Side Handhole Stud Mounting Holes"
EAR 98-0259, "SW Pump Impeller Metal Buildup"
FES 98-031, "Traveling Screen Basket Material Change"
FES 98-001, "MSIV Disk Assembly Belleville Washers"
EAR 96-0576, "Sealing Pipe Penetrations in Lube Oil Storage Blockwall"
EAR 95-0268, "Circulating Water Discharge Radiation Detector"
EAR 96-0809, "Pressure Regulator Has a Blown Diaphragm"
EAR 96-0645, "Justification of Heat Exchanger Cover Plant Thickness"
EAR 95-0512, "LPSI Pump Seal Enhancement"
EAR 96-0727, "Safety Injection Refueling Water Tank Low Level Switches"
EAR 97-0634, "Document Available NPSH for 200°F CCW Temperature"

Temporary Modifications

TM 97-027, "Install Mechanical Blocks to Hold Damper Open"
TM 97-040, "Install Blank Flange on Discharge Piping"
TM 97-052, "Disable Low Flow Alarm for Failed Fuel Monitor"
TM 97-017, "Install Jumper to Bypass 25F7 Interlocks to Disconnect Switch"
TM 98-018, "Install Hydrostatic Plug for In-Core Instrumentation"
TM 97-031, "Block Closed CV-0825 & CV-0878"
TM 98-009, "Install Temporary Filter in Spent Fuel Pool Tilt Pit Drain"
TM 96-032,"Modification of Plant Interface Connections to Supply Hook-Up Locations Needed to Install the Temporary VRS Unit"
TM 97-026, "Install Bypass Line Around TC-0852"

Licensee Event Reports

LER 96013, DC Breaker Failure During Testing For As-Found Trip Setting
LER 96005, Appendix R Enhancement Analysis - DC Panels Breaker/Fuse Coordination Issue

Licensing Documents

Palisades Design Inspection Report No. 50-255/97-201, 12/30/97.
Palisades Letter to USNRC dated 10/30/97, 30-Day Response to Generic Letter 97-04.
Palisades Letter to USNRC dated 1/5/98, 90-Day Response to Generic Letter 97-04.
Palisades Letter to USNRC dated 6/3/93, Response to NRC Bulletin 93-02.
Palisades Letter to USNRC dated 9/11/97, Supplemental Response to NRC Bulletin 93-02.

FSAR Table 14.22-2, Parameters Used in the Offsite Radiological Consequences Analysis of the Palisade Plant
Technical Specification 6.5.2, Primary Coolant Sources Outside Containment.

Drawings

M74, Sheet 1, "Underground Piping, Reactor Building," Revision 12
C-155, "Reactor Refueling Cavity & sump Liner," Revision 12
M202, Sheet 1, "Chemical & Volume Control System," Revision 60
M225, Sheet 1, "High Pressure Air Operated Valves," Revision 36
M221, Sheet 2, "Spent Fuel Pool Cooling System," Revision 30

Calculations

EA-A-PAL-96-003, "ECCS Evaluation in Post-RAS Recirculation Modes Using Pipe-Flo," Revision 1
EA-C-PAL-96-088301, "Containment Spray Pump Runout and Impact of Low Flow Rates on Pump," , Revision 0
EA-DBD-1.05-03, "Engineering Analysis for DBD-1.05 Open Item 2," Revision 0

NRC Generic Letters

85-22, Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage
97-04, Assurance of Sufficient Net Positive Suction Head [NPSH] for Emergency Core Cooling and Containment Heat Removal Pumps
93-64, Periodic Testing and Preventative Maintenance of Molded Case Circuit Breakers

NRC Information Notices

87-63, Inadequate Net Positive Suction Head in Low Pressure Safety Systems
88-74, Potentially Inadequate Performance of ECCS in PWRs During Recirculation Operation Following a LOCA
90-07, New Information Regarding Insulation Material Performance and Debris Blockage of PWR Containment Sumps
92-71, Partial Plugging of Suppression Pool Strainers at a Foreign BWR
93-02, Debris Plugging of Emergency Core Cooling Suction Strainers
93-26, Grease Solidification Causes Molded Case Circuit Breaker Failure To Close
93-34, Potential Loss of Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment
93-64, Periodic Testing and Preventative Maintenance of Molded Case Circuit Breakers
96-55, Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps Under Design Basis Accident Conditions
97-27, Effect of Incorrect Strainer Pressure Drop on Available Net Positive Suction Head

NRC Bulletins

95-02, Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode
96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors

Audits and Assessments

PA-96-29, "Palisades JUMA Audit," dated 11/9/96
PA-97-09, "Palisades Mechanical Maintenance Program Audit," dated 3/20/98
PA-97-010, "Palisades JUMA Audit," dated 7/18/97
PA-98-03, "Palisades Operations," dated 8/21/97
Assessment Report, Operations/Shift #3 - Compliance with Alarm Response Standards, dated 2/28/98
Assessment Report, Operations/Shift #4 - Quarterly Caution Tag Verification, dated 3/30/98
Assessment Report, Operations/Shift #4 - Caution Tag Administration Review, dated 3/31/98

Miscellaneous

Close-out Memo for IN 90-007, 8/18/92.
Specification M-136, Rev 9, 4/12/95, Furnishing and Installing Conventional Type Insulation.
Industry Experience Traveler PS 32948, Potential for Fibrous Insulation Material to Cause ECCS Sump Blockage at the Cook Nuclear Plant.
PRC Meeting 98*03 Minutes Dated 1/22/98.
Operation Department, Self-Assessment Plan
"Hot Button" Trend Data
Maintenance Organization Critical Self-Assessment Plan-Fuel Cycle 14
Maintenance "Hot Button" Report
Palisades Performance Monitoring - Management Summary
1997 JUMA Audit PA-97-10 Recommendation Status
1996 JUMA Audit PA-96-29 Concerns and Recommendation Status
Engineering Department Self-Assessment Completed 1997 - 1998
Condition Reports Requiring Root Cause Analysis (June 1997 - June 1998)
Open/Closed Operability Evaluations (June 1997 - 1998)
Open/Closed Corrective action Document(June 1997 - 1998)