May 1, 2000

Mr. Thomas J. Palmisano
Site Vice President and General Manager
Palisades Nuclear Generating Plant
Consumers Energy Company
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES INSPECTION REPORT 50-255/2000002(DRP)

Dear Mr. Palmisano:

On April 01, 2000, the NRC completed an inspection conducted at your Palisades Nuclear Generating Plant. The enclosed report presents the results of that inspection.

The inspection covered a 6-week period. Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of a selective examination of procedures and representative records, interviews with personnel, and observations of activities in progress.

Based on the results of this inspection, the NRC has determined that one violation of NRC requirements occurred. The violation involved three examples of previously identified problems where, for significant conditions adverse to quality, your staff failed to take measures to determine the cause and initiate corrective actions to prevent repetition. Consequently, past corrective action did not prevent repetitive problems in October 1999. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or the severity level of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, the enclosure, and your response, if you choose to provide one, will be placed in the NRC Public Electronic Reading Room (PERR) link at the NRC homepage, http://www.nrc.gov/NRC/ADAMS/index.html
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Sincerely,

Original signed by
Michael J. Jordan, Chief

Michael J. Jordan, Chief
Reactor Projects Branch 3

Docket No. 50-255
License No. DPR-20

Enclosure: Inspection Report 50-255/2000002(DRP)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	50-255/2000002(DRP)
Licensee:	Consumers Energy Company 212 West Michigan Avenue Jackson, MI 49201
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
Dates:	March 9 through April 1, 2000
Inspectors:	J. Lennartz, Senior Resident Inspector R. Krsek, Resident Inspector
Approved by:	Michael J. Jordan, Chief Reactor Projects Branch 3 Division of Reactor Projects

EXECUTIVE SUMMARY

Palisades Nuclear Generating Plant
NRC Inspection Report 50-255/2000002(DRP)

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 3.5-week period of resident inspection activities from March 9 through March 31, 2000.

Engineering

- Root cause evaluations that licensee personnel conducted regarding the equipment failures that were identified in October 1999 were thorough and comprehensive. Also, the equipment problems were appropriately evaluated with respect to the Maintenance Rule. Documented corrective actions were reasonable. (Section E7.1)

- Licensee personnel failed to identify the cause of equipment problems during past evaluations. Consequently, past corrective actions did not prevent repetitive equipment problems regarding degraded bearings in control rod drive mechanisms, failed containment air cooler service water outlet check valves, and cracked control rod drive seal housings. The self revealing equipment problems were three examples of failure to identify the cause to prevent repetitive significant conditions adverse to quality and was treated as a Non-Cited Violation. (Section E7.1)

Report Details

Summary of Plant Status

The plant was at full power when the inspection period started. Plant power was decreased to approximately 59 percent on March 11, 2000, because of an emergent equipment problem regarding a failed seal on a main feedwater pump. Necessary repairs were completed and plant power was escalated to full power which was achieved on March 15. The plant remained at full power for the remainder of the inspection period.

I. Operations

O8 Miscellaneous Operations Issues

O8.1 (Closed) Licensee Event Report (LER) 50-255/98-006: "Manual Operator Actions Not Adequately Addressed In Operating Procedures."

This licensee identified and corrected this issue which was discussed in NRC Inspection Report 50-255/98002(DRP) and resulted in a Non-Cited Violation. The inspectors did not identify any new issues regarding this event report. This item is closed.

O8.2 (Closed) Supplemental LER 50-255/98-006-01: "Manual Operator Actions Not Adequately Addressed In Operating Procedures."

This licensee identified and corrected this issue which was discussed in detail in NRC Inspection Report 50-255/98002(DRP) and resulted in a Non-Cited Violation. This supplemental report contained the evaluation for a pending action that was identified in LER 50-255/98-006. Specifically, a simultaneous small break loss of coolant accident and loss of offsite power with a single failure such as loss of an emergency diesel generator was evaluated. Licensee personnel subsequently concluded that manual actions directed by current procedural guidance was adequate to ensure the availability of high pressure air. The inspectors reviewed the procedure guidance and did not identify any significant concerns.

In addition, a modification was subsequently performed that repowered the high pressure air compressors from the same electrical division as the emergency core cooling train it supported. The inspectors noted that the modification effectively eliminated a required manual operator action to crosstie the high pressure air receiver tanks during loss of offsite power with a failure of a single emergency diesel generator event.

The inspectors did not identify any new issues associated with this event report. This item is closed.

O8.3 (Closed) LER 50-255/00-002: "Non-conformance With Technical Specification (TS) Requirements For Auxiliary Feedwater Pump P-8B."

This issue was discussed in detail in NRC Inspection Report 50-255/2000001 and was being tracked as Unresolved Item 50-255/2000001-02 which was closed in Section E8 of this report. The inspectors did not identify any new issues associated with this event report. This item is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Surveillance Test Observations (61726)

The inspectors reviewed the applicable sections of the TSs and the Final Safety Analysis Report. Also, the inspectors reviewed the completed documentation and observed the performance of selected portions of the following surveillance test.

- RO-127 Auxiliary Feedwater System, 18-Month Test Procedure

The inspectors did not identify any significant issues regarding the performance of Surveillance Test RO-127 and concluded that the surveillance test was completed in accordance with plant procedures.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) Inspector Followup Item (IFI) 50-255/96017-05(DRS): "Review of Licensee's Methodology For Analyzing Cable."

The inspectors questioned the adequacy of the licensee's analysis of cable tray fill and cable ampacities to verify power cables would not excessively overheat and exceed electrical code requirements specified in the Final Safety Analysis Report. The licensee's analysis utilized the Harshe-Black diversity based ampacity method that had not been approved by NRC. Therefore, this issue was opened pending Office of Nuclear Reactor Regulation (NRR) review of the licensee's methodology for analyzing cable.

The NRC staff, with assistance from its contractor, Sandia National Laboratories, completed a review of the licensee's cable ampacity methodology and found it to be acceptable. The Office of Nuclear Reactor Regulation issued a letter dated March 1, 2000, to the licensee titled "Palisades Plant - Cable Ampacity Adjustment Methodology," that presented the review results. Enclosed with the letter was the Staff Evaluation Report and the associated Letter Report titled, "A Review of the Harshe-Black Diversity Based Ampacity Method as Published and as Applied at the Palisades Nuclear Plant," dated December 19, 1997.

On the basis of the Staff Evaluation Report, the NRC staff concluded the following:

- the relevant concerns associated with the Harshe-Black ampacity adjustment methodology had been resolved;

- the licensee provided an adequate technical basis to assure the modified Harshe-Black methodology was acceptable for use at Palisades; and,

- there were no outstanding safety concerns regarding the cable ampacity methodology used at Palisades.

Therefore, this item is closed.

III. Engineering

E7 Quality Assurance In Engineering Activities

E7.1 Root Cause Evaluations For Equipment Failures

a. Inspection Scope (37551, 92903)

The inspectors reviewed root cause evaluations and associated event reports regarding equipment problems which were documented in the following Condition Reports (CPAL) and LER:

- CPAL9901817 “Control Rod Failure to Trip (CRD-14)”
- CPAL9902561 “Reactor Head Cooling Ductwork Louvers Found Closed”
- CPAL9902295 “Control Rod Drive Mechanism Housing Crack and Indication”
- CPAL9801988 “Boric Acid Leakage on Control Rod Drive No. 2”
- CPAL9902268 “Failure of Containment Air Cooler Service Water Outlet Check Valves CK-SW407, CK-SW408, and CK-SW409"
- LER 98-014 “Control Rod Drive Seal Housing Leak”
- LER 99-004 “Control Rod Drive Seal Housing Leaks and Crack Indications”
- LER 99-003 “Reduction in Service Water Flow Through Containment Air Coolers VHX-1 and VHX-2"
- LER 99-003-01 “Reduction in Service Water Flow Through Containment Air Coolers VHX-1 and VHX-2"

The inspectors also reviewed applicable plant drawings, the vendor manual for the control rod drive mechanisms as well as the following documents:

- Engineering Manual - 25, “Maintenance Rule Program,” Revision 2
- Technical Specifications 3.4, Amendment No. 172
- Final Safety Analysis Report Chapter 14, “Safety Analysis,” Revision 21
- Technical Specification 3.4, Amendment No. 172;
- Standard Operating Procedure - 5, “Containment Air Cooling and Hydrogen Recombining System,” Revision 16

b. Observations and Findings

The inspectors noted that the licensee's root cause evaluations were thorough and comprehensive regarding the following equipment problems:

- failure of Control Rod No. 14 to insert into the core on October 16, 1999, because of degraded bearings in the control rod drive mechanism clutch assembly (CPAL9901817);
- ventilation louvers on Reactor Head Cooling System found closed in November 1999 (CPAL9902561);
- leaks from seal housings for control rod drive mechanisms No. 10, and No. 44 because of through-wall cracks in October 1999 (CPAL9902295, LER 99-004); and
- failure of containment air cooler service water outlet check valves CK-SW407, CK-SW408, and CK-SW409, in October 1999 (CPAL9902268, LER 99-003 and 99-003-01)

Also, the inspectors noted that in each evaluation, the documented corrective actions were reasonable and that licensee personnel evaluated each equipment problem with respect to the Maintenance Rule. The inspectors did not identify any concerns regarding licensee personnel conclusions regarding the Maintenance Rule evaluations.

However, the inspectors reviewed the licensee's root cause evaluations regarding past equipment problems that were similar to the issues identified in 1999 and determined that a violation of NRC requirements occurred. Specifically, 10 CFR 50, Appendix B, Criterion 16, states, in part, that for significant conditions adverse to quality, measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

The inspectors determined that the three equipment problems that occurred in October 1999 were significant conditions adverse to quality in which licensee personnel had failed to identify the cause of the condition during previous root cause evaluations. Consequently, past corrective actions did not prevent the equipment problems in October 1999 regarding degraded bearings in control rod drive mechanisms, failed containment air cooler service water outlet check valves, and cracked control rod drive seal housings which had all previously occurred.

Therefore, the self revealing equipment problems that occurred in 1999 were three examples of a violation of 10 CFR 50, Appendix B, Criterion 16 requirements. This Severity Level IV Violation is being treated as a Non-Cited Violation (NCV 50-255/2000002-01), consistent, with Section VII.B.1.a of the NRC Enforcement Policy. Details of the three examples where licensee personnel failed to identify the cause to prevent repetitive equipment problems are described below.

b.1 Failure of Control Rod No. 14

During plant shutdown for refueling in October 1999, Control Rod Drive No. 14 failed to insert into the core via gravity, as designed, when a manual reactor trip signal was initiated. The control rod was instead driven into the core by an automatic rundown feature. Subsequent troubleshooting determined that the failure to trip was caused by a failed bearing in the control rod drive mechanism internal clutch assembly. The details surrounding this issue and the repairs made to the drive mechanisms are discussed in detail in NRC Inspection Reports 50-255/99011(DRP) and 50-255/99012(DRP), Sections E2.1 and M1.3, respectively.

During the evaluation regarding the failure of Control Rod No. 14, licensee personnel identified that the non-safety related Reactor Head Cooling System was misaligned in that 95 percent of the ventilation louvers were closed. Licensee personnel could not definitively determine when the louvers were closed. However, review of available data suggested that the Reactor Head Cooling System had been in this configuration for at least five fuel cycles. Licensee personnel concluded that the system did not provide adequate cooling to the control rod drives mechanisms in the as found configuration.

Therefore, the control rod drive mechanisms operated in temperatures in excess of the 130°F design temperature for continuous operation for a minimum of five fuel cycles. Consequently, this condition thermally affected the control rod drive mechanism bearings and shortened the expected service life to the point of grease breakdown and subsequent bearing failure. Licensee personnel documented this issue in Condition Report CPAL9902561, “Reactor Head Cooling Ductwork Louvers Found Closed,” which was entered into the corrective action program.

The inspectors also reviewed the licensee’s evaluation, documented in Condition Report CPAL9901817, and noted that similar bearing degradation had been identified in the past. Specifically, licensee personnel had disassembled and inspected seven control rod drive mechanisms in November 1996, which revealed degraded bearings (dry grease and corrosion products present). External, and in some cases, internal clutch bearings were degraded and were subsequently replaced. Six additional drive packages were inspected after degraded bearings were discovered. The internal clutch bearing on Drive Mechanism No. 12 was found to be rough and was subsequently replaced.

Licensee personnel concluded in 1996 that the inspections that were conducted provided adequate assurance that significant bearing degradation in other drive mechanisms was unlikely. In addition, licensee personnel were confident that acceptable rod drop times and torque traces performed every refueling outage combined with a rebuild of 5 to 10 drive packages each outage would identify any additional bearing degradation issues.

However, the bearing that seized and prevented Control Rod No. 14 to fall into the core on October 16, 1999, served that same function as the degraded bearing that was found on Control Rod No. 12 in 1996. The inspectors determined that the evaluations completed in1996 did not thoroughly evaluate the causes for the degraded bearings in the drive mechanisms.

Therefore, the inspectors determined that a violation of NRC requirements occurred. Specifically, 10 CFR 50, Appendix B, Criterion 16, states, in part, that for significant conditions adverse to quality, measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

After licensee personnel discovered degraded control rod drive mechanism bearings in November 1996, the measures taken did not identify that the Reactor Head Cooling System ventilation louvers were closed which was a cause of bearing degradation.

Also, adequate preventative maintenance practices were not implemented to preclude bearing degradation.

Consequently, the corrective actions taken in 1996 did not preclude repetition as evidenced by the failed bearing that prevented Control Rod No. 14 to fall into the core via gravity, as designed, in October 1999. This Severity Level IV Violation was self revealing and is being treated as a Non-Cited Violation (NCV 50-255/2000002-01a), consistent, with Section VII.B.1.a of the NRC Enforcement Policy. Theses issue were entered into the licensee's corrective action program as Condition Reports CPAL9901817 and CPAL9902561.

Licensee personnel concluded that the safety significance was minimal which was considered reasonable. All other full length control rods fulfilled the intended safety function of inserting into the core and the failure of Control Rod No. 14 was bounded by analyzed accidents in the Final Safety Analysis Report. Also, one control rod that could not be tripped was a permissible operating condition as defined by TS 3.10.4.b.

b.2 Failure of Containment Air Cooler Service Water Check Valves

Licensee personnel performed Special Test T-218, "Service Water Pumps P-7A, P-7B, and P-7C Performance Test By Flow To Containment," on October 29, 1999, during a refueling outage which revealed a reduction in service water flow through the containment air coolers. The flow rate was approximately 360 gallons per minute less than the flow obtained during similar testing in the 1998 refueling outage. This issue was discussed in detail in NRC Inspection Report 50-255/99012(DRP).

Licensee personnel identified, during visual inspections on October 31, 1999, with the plant in refueling shutdown, that the valve disc was disconnected from the valve disc swing arm on check valves CK-SW407, CK-SW-408, and CK-SW409. Also, CK-SW407 and CK-SW408 discs were wedged in the valve outlet port while CK-SW409 disc was lying on the valve bottom. Consequently, service water flow through Containment Air Coolers VHX-1 and VHX-2 was restricted.

Licensee personnel concluded that containment air cooler system operation and the lack of preventative maintenance were root causes. The system was operated such that the flow rate through the check valves was not enough to maintain the check valve in full open position. Consequently, the valve disc fluttered that caused accelerated wear and there was no preventative maintenance plan that would have identified the accelerated wear before the valves failed.

However, the inspectors reviewed the root cause evaluation as documented in Condition Report CPAL9902268 as well as the associated LER 99-003 and 99-003-01 and noted that similar equipment failures had occurred in the past.

The containment air cooler service water outlet check valves were replaced several times in the past for conditions similar to those identified in October 1999. Specifically, valve CK-SW407 had been replaced in 1990 and 1995; valve CK-SW408 had been replaced in 1990 and 1996; and valve CK-SW409 had been replaced in 1990.

Therefore, the inspectors determined that a violation of NRC requirements occurred. Specifically, 10 CFR 50, Appendix B, Criterion 16, states, in part, that for significant conditions adverse to quality, measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

Licensee personnel's past evaluations failed to identify that system operation caused valve flutter which resulted in excessive wear and subsequent failure of containment air cooler service water outlet check valves CK-SW407, CK-SW408 and CK-SW409. Consequently, the corrective actions did not preclude repetition as evidenced by the failure of check valves CK-SW407, CK-SW408, and CK-SW409 which were identified in October 1999. This Severity Level IV Violation was self revealing and is being treated as a Non-Cited Violation, consistent, with Section VII.B.1.a of the NRC Enforcement Policy (NCV 50-255/2000002-01b). This issue was entered into the licensee's corrective action program as Condition Report CPAL9902268.

Licensee personnel concluded that this issue had minimal safety significance based on the following:

- The indicated flow through VHX-1 and VHX-2 during non-intrusive testing on October 29, 1999, provided evidence that the containment air coolers retained a substantial portion of design capability.

- Based on the sensitivity studies, that the restricted flow would not have prevented the containment air coolers from performing the safety function of maintaining containment pressure below design following a main steam line break accident.

- Either of two trains of containment cooling equipment were required to mitigate a main steam line break accident. The failed service water check valves did not affect the equipment train that consisted of two containment spray pumps which was available to mitigate a main steam line break accident.

Licensee personnel's conclusion was considered reasonable.

b.3 Leaks From Seal Housings On Control Rod Drive Mechanisms

Licensee personnel identified leaks from the seal housings for Control Rod Drive Mechanisms No. 10, No. 26, and No. 44 on October 16, 1999, when the plant was shutdown for a refueling outage. This issue was discussed in NRC Inspection Report 50-255/99012(DRP), Section E8.2.

The three control rod drive seal housings were removed from the reactor head and subjected to nondestructive visual and dye penetrant examinations. The examinations determined that all three housings had crack indications in the vicinity of the "J" welds which attached the seal housing tube to the autoclave flange. Further testing revealed that seal housings for Control Rod Drives No. 10 and No. 44 had through-wall cracks. This issue was appropriately reported to the NRC as LER 99-004, "Control Rod Drive Seal Housing Leaks and Crack Indications," on December 1, 1999.

Licensee personnel concluded that the seal housing cracking identified in October 1999 was caused by use of a material susceptible to transgranular stress corrosion cracking. Also, the use of induction heating for post weld heat treatment during seal housing manufacturing was inadequate which contributed to the seal housing cracking. The use of induction heating did not completely relieve the residual stresses in the vicinity of the "J" weld. Instead, the post weld heat treatment induced tensile stresses which was one factor that was required for transgranular stress corrosion cracking to occur in the austenitic stainless steel Type 304 and Type 347 seal housings that were installed.

However, the inspectors noted that a similar equipment problem occurred previously in December 1998. Licensee personnel identified a leak from the seal housing on Control Rod Drive Mechanism No. 2. Further investigation revealed that the leak was due to transgranular stress corrosion cracking which resulted in a through-wall crack. This issue was appropriately reported to the NRC in LER 98-014, "Control Rod Drive Seal Housing Leak," on January 26, 1999.

The inspectors reviewed the root cause evaluation for the leak from seal housing No. 2 as documented in Condition Report CPAL9801998. During the evaluation, licensee personnel removed and inspected two additional seal housings to address any generic concerns. Seal housings for Control Rod Drive Mechanisms No. 10 and No. 23 were removed and inspected on January 1, 1999, and no indications were found on either housing. Therefore, licensee personnel concluded in January 1999 that the leak from seal housing No. 2 was an isolated event which was caused by excessive rework during the manufacturing process.

Also, the inspectors noted that licensee personnel reviewed manufacturing records during the evaluation for the leak that occurred in 1998 from seal housing No. 2. The manufacturing records revealed that all 45 seal housings were given a post weld heat treatment using induction heating. However, licensee personnel did not identify, at that time, that the use of induction heating induced stresses in the seal housings. Consequently, licensee personnel missed an opportunity to identify a generic cause of cracking in control rod drive seal housings during the evaluation conducted in 1998.

Therefore, the inspectors determined that a violation of NRC requirements occurred. Specifically, 10 CFR 50, Appendix B, Criterion 16, states, in part, that for significant conditions adverse to quality measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

Licensee personnel's evaluation for the leak on seal housing No. 2 in 1998 failed to identify that the use of induction heating as a post weld heat treatment during the manufacturing process induced tensile stresses that caused seal housing cracking. Consequently, the corrective actions taken in 1998 did not preclude repetition as evidenced by leaks from seal housings for Control Rod Drive Mechanisms No. 10, No. 26, and No. 44 that were identified on October 16, 1999. This Severity Level IV Violation was self revealing and is being treated as a Non-Cited Violation (NCV 50-255/2000002-01c), consistent, with Section VII.B.1.a of the NRC Enforcement Policy. These issues were entered into the licensee's corrective action program as Condition Reports CPAL9902295 and CPAL9801988.

Licensee personnel identified the seal housing cracking while the plant was shutdown and therefore, there were no actual adverse safety consequences. Also, licensee personnel's evaluation determined that total failure of a seal housing from transgranular stress corrosion cracking prior to detection was not a credible event. However, if a complete failure would occur, the resultant small break loss of coolant accident was an analyzed event that the plant was designed for. Therefore, licensee personnel concluded that the seal housing cracking had minimal safety significance which was considered reasonable.

c. Conclusions

The inspectors concluded that the root cause evaluations that licensee personnel conducted regarding the equipment failures that were identified in October 1999 were thorough and comprehensive. Also, the equipment problems were appropriately evaluated with respect to the Maintenance Rule. Documented corrective actions were reasonable.

However, the inspectors determined that licensee personnel failed to identify the cause of equipment problems during past evaluations. Consequently, past corrective actions did not prevent three repetitive significant conditions adverse to quality equipment problems. This issue was treated as a Non-Cited Violation.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Unresolved Item 50-255/2000001-02: "Resolution of Notice of Enforcement Discretion Issues Associated With the Auxiliary Feedwater Pump P-8B Alternate Steam Supply Line."

This unresolved item was discussed in detail in NRC Inspection Report 50-255/2000002(DRP). On February 5, 2000, with the plant in hot shutdown, the alternate steam supply line to Turbine Driven Auxiliary Feedwater Pump P-8B ruptured. The licensee subsequently replaced the damage section of piping. However, the remainder of piping that was buried underground could not be accessed. Consequently, the integrity of the remainder of the underground piping could not be quantitatively proven. Therefore, the licensee considered the alternate steam supply line inoperable on February 13, 2000.

However, TS Surveillance Requirement 4.9.a.2 specifically required testing of the alternate steam supply line. Consequently, the alternate steam supply line was a required feature to support operability of Pump P-8B. Therefore, the licensee requested and was granted Notice of Enforcement Discretion 00-6-002 on February 16, 2000 from NRC, Office of Nuclear Reactor Regulation, to permit plant startup and subsequent operation until a TS change request could be processed.

The item was opened pending NRC Office of Nuclear Reactor Regulation review of TS change request that the licensee submitted on February 18, 2000, as required by Notice of Enforcement Discretion 00-6-002. Specifically, the request was submitted to remove the testing pertaining to the alternate steam supply line for P-8B as required by current TS 4.9a.2, "Auxiliary Feedwater System Tests Surveillance Requirements - Auxiliary Feedwater Pumps." Related changes to improved TS 3.7.5, "Auxiliary Feedwater System," as issued on November 30, 1999, but not yet implemented were also requested.

The office of Nuclear Reactor Regulation subsequently issued the TS change, in Amendment No. 190 to Facility Operating License No. DPR-20 for the Palisades plant in a letter dated March 14, 2000. This item is closed.

E8.2 (Closed) Inspector Follow-up Item (IFI) 50-255/99009-01: "Boric Acid Leak Inspection Program Ineffectiveness."

This item was documented in detail in NRC Inspection Report 50-255/99009(DRP). The inspectors did not identify any violations of regulatory requirements pertaining to this item. Also, the item was entered into the licensee's corrective action program as Condition Report 99-1237, "NRC Identified Ownership / Effectiveness Concerns With Boric Acid Program." The corrective actions documented in the condition report appeared reasonable. Therefore, this item is closed.

E8.3 (Closed) Violation 50-255/98012- 01: "Failure To Implement Testing of Molded Case Circuit Breakers In a Timely Manner."

This issue was discussed in detail in NRC Inspection Report 50-255/98012(DRS) and was also the subject of LER 96-013, "DC Breaker Failure During Testing For As-Found Trip Setting," and an associated supplement. The NRC concluded, as documented in NRC Inspection Report 50-255/98012, that the corrective actions taken and planned to correct the violation, and to prevent recurrence, were adequately addressed in the LER. Also, this issue was entered into the licensee's corrective action program as Condition Report CPAL961453. Therefore, this item is closed.

E8.4 (Closed) LER 50-255/99-003: "Reduction in Service Water Flow Through Containment Air Coolers VHX-1 and VHX-2."

This self revealing and corrected event was discussed in detail in NRC Inspection Report 50-255/99012(DRP) and remained open pending an analysis of the event considering the impact of reduced service water flow through two containment air coolers. That analysis was subsequently completed and reported in Supplemental LER 50-255/99-003-01 as discussed in Section E8.5 of this report. The inspectors did not identify any new issues regarding this event report. This item is closed.

E8.5 (Closed) Supplemental LER 50-255/99-003-01: "Reduction in Service Water Flow Through Containment Air Coolers VHX-1 and VHX-2."

This self revealing event was discussed in detail in NRC Inspection Report 50-255/99012(DRP) and in Section E7.1 of this report. This supplemental report contained the analysis for a pending action that was identified in LER 50-255/98-003. Specifically, the impact of reduced flow through two containment air coolers was analyzed. This self revealing and corrected event resulted in a Non-Cited Violation (50-255/2000002-01b). The inspectors did not identify any new issues regarding this event. This item is closed.

E8.6 (Closed) LER 50-255/98-014: "Control Rod Drive Seal Housing Leak."

This self revealing and corrected event was discussed in detail in Section E7.1 of this report and resulted in a Non-Cited Violation (50-255/2000002-01c). This item is closed.

E8.7 (Closed) LER 50-255/99-004: "Control Rod Drive Seal Housing Leaks and Crack Indications."

This self revealing and corrected event was discussed in detail in Section E7.1 of this report and resulted in a Non-Cited Violation (50-255/2000002-01c). This item is closed.

E8.8 (Closed) Revised LER 50-255/98-011-01: "Inadequate Lube Oil Collection System For the Primary Coolant Pumps."

This licensee identified event was discussed in detail in NRC Inspection Report 50-255/98022(DRP) and resulted in a Non-Cited Violation 50-255/98022-05. This revised event report revealed that the primary coolant pump oil collection tanks' useable volume was less than that as documented in LER 50-255/98-011. However, the differences were insignificant and did not change the event's safety significance which was low.

The licensee submitted an exemption from the 10 CFR 50 Appendix R requirements to the NRC Office of Nuclear Reactor Regulation in a letter dated August 13, 1999, for the existing collections tank design. The licensee subsequently received an exemption from the requirements of 10 CFR Part 50, Appendix R, Section III.O, in a letter dated March 31, 2000, from the NRC Office of Nuclear Reactor Regulation. Also, this issue was documented in the licensee's corrective action program as Condition Report CPAL981962. This item is closed.

E8.9 (Closed) Inspector Followup Item 50-255/99011-02: "Root Cause Evaluation Associated With the Failure of Control Rod Drive Mechanism 14."

This item was discussed in detail in Section E7.1 of this report and resulted in a Non-Cited Violation (50-255/2000002-01a). This item is closed.

E8.10 The Severity Level IV violation listed below were issued in Notices of Violation prior to the March 11, 1999, implementation of the NRC's new policy for treatment of Severity Level IV violations (Appendix C of the Enforcement Policy). Because these violations would have been treated as Non-Cited Violations in accordance with Appendix C, they are being closed out in this report. The violation was as follows:

- (Closed) Violation (VIO) 50-255/98003-09: "Overcurrent Relays for Supply Breakers 152-105 and 152-106 to Bus 1C had not been Calibrated and Tested as Required by the Surveillance Test Program."

This violation was documented in the licensee's corrective action system as Condition Report CPAL9701568, and the corrective actions taken appeared adequate. This item is closed.

E8.11 (Closed) Inspector Followup Item (IFI) 94014-62: "Weakness in the Implementation of Program to Control Electrical Load Growth."

This item pertained to a program weakness and was entered into the licensee's corrective action program as Action Item Record (AIR) Number A-NL-95-091 which can be used to track and assess corrective actions. This item is closed.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on April 05, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. R. Boss, Operations Manager
D. E. Cooper, General Manager, Plant Operations
P. D. Fitton, System Engineering Manager
G. A. Katt, System Engineering
K. M. Haas, Director, Engineering
D. G. Malone, Acting Director, Licensing
R. L. Massa, Shift Operations Supervisor
T. J. Palmisano, Site Vice President

NRC

D. Hood, Project Manager, NRR

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92901:	Follow-up Operations
IP 92902:	Follow-up Maintenance
IP 92903:	Follow-up Engineering
IP 92700:	LER Follow-up

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-255/2000002-01	NCV	Three examples of Failure to Identify the Cause of a Significant Condition Adverse to Quality to Prevent Repetition

Closed

50-255/2000002-01	NCV	Three examples of Failure to Identify the Cause of a Significant Condition Adverse to Quality to Prevent Repetition
50-255/98-006	LER	Manual Operator Actions Not Adequately Addressed In Operating Procedures
50-255/98-006-01	LER	Manual Operator Actions Not Adequately Addressed In Operating Procedures
50-255/00-002	LER	Non-conformance With TS Requirements For Auxiliary Feedwater Pump P-8B
50-255/96017-05	IFI	Review of Licensee's Methodology For Analyzing Cable
50-255/2000001-02	URI	Resolution of Notice of Enforcement Discretion Issues Associated With the Auxiliary Feedwater Pump P-8B Alternate Steam Supply Line
50-255/99009-01	IFI	Boric Acid Leak Inspection Program Ineffectiveness
50-255/98012- 01	VIO	Failure To Implement Testing of Molded Case Circuit Breakers In a Timely Manner
50-255/99-003	LER	Reduction in Service Water Flow Through Containment Air Coolers VHX-1 and VHX-2
50-255/99-003-01	LER	Reduction in Service Water Flow Through Containment Air Coolers VHX-1 and VHX-2
50-255/98-014	LER	Control Rod Drive Seal Housing Leak
50-255/99-004	LER	Control Rod Drive Seal Housing Leaks and Crack Indications
50-255/98-011-01	LER	Inadequate Lube Oil Collection System For the Primary Coolant Pumps
50-255/99011-02	IFI	Root Cause Evaluation Associated With the Failure of Control Rod Drive Mechanism 14

50-255/98003-09	VIO	Overcurrent Relays for Supply Breakers 152-105 and 152-106 to Bus 1C had not been Calibrated and Tested as Required by the Surveillance Test Program
50-255/94014-62	IFI	Weakness in the Implementation of Program to Control Electrical Load Growth

Discussed

None