April 29, 2003

Mr. Douglas E. Cooper
Site Vice President
Palisades Nuclear Plant
Nuclear Management Company, LLC
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR GENERATING PLANT
NRC INTEGRATED INSPECTION REPORT 50-255/03-02

Dear Mr. Cooper:

On March 31, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Palisades Nuclear Generating Plant. The enclosed report documents the inspection findings which were discussed on April 11, 2003, with yourself and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, three findings of very low safety significance (Green) were identified which involved violations of NRC requirements. However, because of their very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest these Non-Cited Violations, you should provide a response with a basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 801 Warrenville Road, Lisle, IL 60532-4351; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Palisades facility.

Since the terrorist attacks on September 11, 2001, the NRC has issued two Orders (dated February 25, 2002, and January 7, 2003) and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance access authorization. The NRC also issued Temporary Instruction 2515/148 on August 28, 2002, that provided guidance to inspectors to audit and inspect licensee implementation of the interim compensatory measures (ICMs) required by the

February 25th Order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) ‘02, and the remaining inspections are scheduled for completion in CY ‘03. Additionally, table-top security drills were conducted at several licensees to evaluate the impact of expanded adversary characteristics and the ICMs on licensee protection and mitigative strategies. Information gained and discrepancies identified during the audits and drills were reviewed and dispositioned by the Office of Nuclear Security and Incident Response. For CY ‘03, the NRC will continue to monitor overall safeguards and security controls, conduct inspections, and resume force-on-force exercises at selected power plants. Should threat conditions change, the NRC may issue additional Orders, advisories, and temporary instructions to ensure adequate safety is being maintained at all commercial power reactors.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Eric Duncan, Chief
Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 50-255/03-02

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	50-255/03-02(DRP)
Licensee:	Nuclear Management Company, LLC
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
Dates:	December 29, 2002, through March 31, 2003
Inspectors:	J. Lennartz, Senior Resident Inspector R. Krsek, Resident Inspector R. Alexander, Radiation Specialist T. Madeda, Physical Security Inspector, RIII C. Phillips, Senior Operations Engineer, RIII
Approved by:	Eric Duncan Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000255/03-02; Nuclear Management Company, LLC; 12/29/2002 - 03/31/2003; Palisades Nuclear Generating Plant; Equipment Alignment; Surveillance Testing; Event Followup.

This report covers a 3-month period of baseline resident inspections, a physical protection inspection, and a radiation protection inspection. The inspections were conducted by resident inspectors and a regional based physical security inspector. Three Green findings with associated Non-Cited Violations (NCVs) were identified during the inspection. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process," (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding for the failure to implement adequate corrective actions to prevent recurrence of issues associated with the construction of seismic scaffolding near safety-related systems.

  This finding was more than minor because if left uncorrected it would become a more significant safety concern in that inadequately constructed scaffold could affect the availability of mitigating systems during a seismic event. The finding was of very low safety significance because the finding did not screen as potentially risk significant due to a seismic initiating event and did not involve the total loss of any safety function that contributes to core damage accident sequences initiated by seismic events. The inspectors also determined that this finding represented continued human performance deficiencies in the construction of seismic scaffolding near safety-related systems. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified. (Section 1R04.1)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding for the failure to ensure that testing of the fuel handling area ventilation system was performed in accordance with test procedures which incorporated the appropriate requirements and acceptance limits specified in Technical Specification 5.5.10, "Ventilation Filter Testing Program."

  This finding was more than minor because if left uncorrected it would become a more significant safety concern in that the radiological barrier function provided by the fuel handling area ventilation system was degraded and was not being tested adequately. The finding was of very low safety significance because the finding represented a degradation of only the radiological barrier function provided for the spent fuel pool. The inspectors also determined that this finding was a result of human performance deficiencies related to developing and implementing the Technical Specification surveillance. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” was identified. (Section 1R22.1)

- Green. The inspectors determined that a self-revealed finding was associated with the failure to restore an inoperable channel of containment hydrogen monitoring within the allowed outage times specified in Technical Specification Action Statements 3.3.7.A and 3.3.7.D.

  The finding was more than minor because the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events was affected. The finding was determined to be of very low safety significance after a Region III Senior Reactor Analyst, in conjunction with the inspectors, performed a SDP Phase 3 assessment. Utilizing NUREG-1675, “Basis Document for Large Early Release Frequency Significance Determination Process,” the analyst determined that the significance threshold for large early release frequency of 100 volume percent per day leak rate from containment would not be exceeded. The inspectors also noted that this finding was attributable to a latent human performance deficiency which occurred during the April 2001 refueling outage, but was self-revealed in December 2002. A Non-Cited Violation of Technical Specification Section 3.3.7 was identified. (Section 4OA3.1)

B. Licensee Identified Findings

One violation of very low safety significance, which was identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. The violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

<u>REPORT DETAILS</u>

A list of documents reviewed within each inspection area is included at the end of the report.

<u>Summary of Plant Status</u>

The plant was at full power for the majority of the inspection period with the following two exceptions:

- During an Unusual Event for an unexplained lowering level in the service water pump bays on February 16, 2003, operators reduced power to 89 percent to stabilize the balance of plant equipment. The lowering level in the service water pump bays was subsequently determined to have been caused by ice forming on the intake structure in Lake Michigan which reduced flow to the pump bays. On February 21, 2003, power was again raised to full power after licensee personnel determined that no ice remained and that no major damage had occurred to the intake structure.

- On March 16, 2003, the plant was shut down to begin a scheduled refueling outage which was ongoing when the inspection period ended.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 <u>Equipment Alignment</u>

.1 <u>Inadequate Corrective Actions for Seismic Scaffolding Built Near Safety-Related Equipment</u> (71111.04S)

a. <u>Inspection Scope</u>

During walkdowns of the auxiliary feedwater system, the inspectors assessed the condition of seismic scaffolding built near safety-related equipment. The inspectors verified the scaffolds were built in accordance with the licensee's procedures and reviewed the licensee's corrective actions taken for scaffolding issues previously identified by the NRC.

b. <u>Findings</u>

<u>Introduction</u>

The inspectors identified a finding of very low safety significance (Green) that is being treated as a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to implement adequate corrective actions to prevent recurrence for issues associated with the construction of seismic scaffolding near safety-related systems.

Description

On February 27, 2003, the inspectors identified that scaffolding erected near safety-related equipment in the west engineering safeguards room did not meet the minimum spacing requirements from the safety-related backup suction piping to Auxiliary Feedwater Pump P-8C. The inspectors also noted that the scaffolding did not have the required engineering evaluation and approval to be built in this manner. Step 5.5.3a of Procedure MSM-M-43, “Scaffolding,” required, in part, that a minimum separation of 1-inch in all directions shall be maintained between braced scaffolding and safety-related equipment. Step 5.5.7 of Procedure MSM-M-43 required, in part, that design engineering provide justification and approval for deviating from the procedure requirements.

The licensee rebuilt the scaffold and initiated Condition Report CAP033667, “Seismic Scaffold Does Not Meet Installed Plant Equipment Separation Requirement.” The licensee’s immediate corrective actions included a temporary suspension of scaffold erection activities, stand-down and retraining of scaffold builders and supervisors, and an extent of condition walkdown to identify other potentially deficient scaffold installations. During the extent of condition walkdown, the licensee identified and corrected several additional seismic scaffolds which were not built in accordance with Procedure MSM-M-43.

On March 3, 2003, the inspectors again identified additional seismic scaffolding erected near safety-related equipment in the component cooling water heat exchanger room which did not meet the requirements of Procedure MSM-M-43. The licensee rebuilt the scaffold, initiated a Condition Report, and again took additional immediate corrective actions.

The inspectors reviewed NRC-identified scaffold control issues which had occurred in the past 2 years to determine if there had been prior opportunities to address seismic scaffolding construction deficiencies. In February and March 2001, just prior to the start of the 2001 Refueling Outage, the NRC issued a Green finding and associated Non-Cited Violation (NCV) 50-255/01-06-02, for several examples of the licensee’s failure to adequately implement procedural requirements for the control of seismic scaffolding in the vicinity of safety-related equipment. The licensee’s root cause analysis for this condition adverse to quality determined that the site lacked a programmatic method to control scaffold design, erection, inspection and approval. The licensee completed all corrective actions to prevent recurrence for this finding on October 15, 2002.

On December 17, 2002, the NRC documented a Green finding and associated NCV 50-255/02-10-01, for several examples of the licensee’s failure to adequately implement procedural requirements for the control of seismic scaffolding in the vicinity of safety-related equipment. Based on the identification of repetitive failures in the past 2 years to adequately control seismic scaffolding construction the inspectors concluded that previous remedial and corrective actions to prevent recurrence were not effective.

The inspectors determined that the failure to implement adequate corrective actions to prevent recurrence of issues associated with the construction of seismic scaffolding near safety-related systems was a licensee performance deficiency warranting a significance evaluation.

Analysis

The inspectors determined that this finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," issued on February 21, 2003, in that the finding would become a more significant safety concern if left uncorrected.

Specifically, scaffolding installed in the vicinity of safety-related equipment could fail during a seismic event and result in damage to mitigating equipment. Consequently, continued human performance deficiencies in the construction of scaffolding near safety-related systems could affect the availability of mitigating systems that respond during seismic events, and if left uncorrected would become a more significant safety concern.

The inspectors evaluated the finding using Manual Chapter 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding:

- was not a design or qualification deficiency;
- did not represent an actual loss of safety function of a system;
- did not represent an actual loss of a safety function of a single train for greater than Technical Specification outage time;
- did not represent an actual loss of a safety function of one or more Non-Technical Specification trains of equipment;
- did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event;
- did not involve the total loss of any safety function that contributed to core damage accident sequences initiated by seismic events; and
- did not involve the loss or degradation of equipment or function designed to mitigate a seismic initiating event.

Therefore, the finding was determined to be of very low safety significance (Green).

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that in the case of significant conditions adverse to quality, measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to this, the licensee's corrective actions to preclude repetitive human performance deficiencies during the construction of seismic scaffolding near safety-related systems have not been effective as evidenced by the following:

- NCV 50-255/01-06-02 issued on April 19, 2001, for several examples of the failure to adequately implement procedural requirements for the control of seismic scaffolding in the vicinity of safety-related equipment;

- NCV 50-255/02-10-01 issued on December 17, 2002, for the failure to adequately implement procedural requirements for the control of scaffolding in the vicinity of safety-related equipment; and
- continued examples of the failure to adequately implement requirements for the construction of seismic scaffolding near safety-related equipment in February and March 2003.

This violation is associated with an inspector identified finding that is characterized by the significance determination process as having very low safety significance (Green) and is being treated as a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 50-255/03-02-01)

This issue was entered in the licensee's corrective action program as Condition Reports CAP033667, "Seismic Scaffold Does Not Meet Installed Plant Equipment Separation Requirement," and CAP033744, "Seismic Scaffold That Did Not Appear to Meet Equipment Separation Requirement."

.2 Semiannual Equipment Alignment Walkdowns (71111.04S)

a. Inspection Scope

The inspectors walked down the Auxiliary Feedwater System utilizing piping and instrumentation diagrams, system operating procedures and system checklists to verify that accessible system components were correctly aligned. The inspectors also reviewed active maintenance work requests; active design and engineering issues, including known operator workarounds and temporary modifications; to verify that the equipment's safety function was not adversely impacted.

b. Findings

No findings of significance were identified.

.3 Quarterly Equipment Alignment Walkdowns (71111.04)

a. Inspection Scope

The inspectors performed partial equipment alignment walkdowns on the following three systems:

- Emergency Diesel Generator 1-2;
- Service Water Traveling Screen Back Wash and Warm Water Recirculation Systems; and
- Low Pressure Safety Injection Pump P-67A.

The inspectors walked down the Emergency Diesel Generator and Low Pressure Safety Injection Pump P-67A to verify proper system lineup while redundant plant equipment was out of service. After the Unusual Event that was declared on February 16, 2003, for the problems associated with the service water intake crib the inspectors walked down the Service Water Traveling Screen Systems. The inspectors verified that power was available, that accessible equipment and components were appropriately aligned, and that no discrepancies existed which would impact the systems' function.

Further, the inspectors reviewed condition reports related to equipment alignment issues to verify that the problems were appropriately characterized, and that planned and completed corrective actions were reasonable.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors toured the following six areas in which a fire could affect safety-related equipment:

- Component Cooling Water Room (Fire Area 16);
- Control Room Complex (Fire Area 1);
- Intake Structure (Fire Area 9);
- Emergency Diesel Generator 1-2 Fuel Oil Day Tank (Fire Area 8);
- Engineered Safeguards Panel Area (Fire Area 15); and
- Containment (Fire Area 14).

The inspectors assessed the material condition of the passive fire protection features and verified that transient combustibles and ignition sources were appropriately controlled. Also, the inspectors reviewed documentation for randomly selected completed surveillances to verify that the sprinkler fire suppression system, smoke detection system, and manual fire fighting equipment for these areas were available.

The inspectors also verified that the fire protection equipment that was installed and available in the fire areas corresponded with the equipment which was referenced in the applicable portions of the Updated Final Safety Analysis Report, Section 9.6, "Fire Protection." Finally, the inspectors verified that compensatory actions were being implemented, as required, for designated fire areas where compensatory actions were needed.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors observed heat exchanger performance testing for the following heat exchangers:

- Component Cooling Water Heat Exchanger E-54B; and
- Containment Air Cooler VHX-3.

The inspectors verified the following items during the inspections:

- Tests conformed with the Licensee's Generic Letter 89-13 Program for Heat Exchanger Inspections;
- Inspection results were appropriately categorized against acceptance criteria and results were acceptable;
- Frequency of inspection was sufficient to detect degradation; and
- Conditions adverse to quality identified during the inspections were appropriately documented in the licensee's corrective action system.

Portions of the heat exchanger tube inspections were observed by the inspectors. In addition, the inspectors observed the cleaning of heat exchanger tubes and verified that the methods used to inspect and clean the heat exchanger were adequate. Additionally, the as-found results of the inspection and testing were verified to be appropriately dispositioned before the system was returned to service.

b. Findings

No findings of significance were identified

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The inspectors observed licensed operator performance during annual operating requalification examinations on January 24, 2003. The inspectors observed portions of operator performance during job performance measures in the simulator for one Senior Reactor Operator and three Reactor Operators. The inspectors assessed the licensed operator's ability to complete the following tasks:

- fill a safety injection tank;
- alternate pressurizer pressure controls; and
- start and load an emergency diesel generator.

The inspectors verified that the operators were able to complete the tasks in accordance with applicable plant procedures and that the success criteria as established in the job performance measures were satisfied.

The inspectors observed the licensee evaluators to ensure that no inappropriate cues were provided by the evaluators while the licensed operators were completing the tasks. The inspectors also reviewed the documented operator evaluations to assess the licensee evaluators' ability to identify licensed operator performance weaknesses.

In addition, the inspectors verified that condition reports written regarding licensed operator requalification training were entered into the licensee's corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13Q)

a. Inspection Scope

The inspectors reviewed Operator's Risk Reports, Shift Supervisor logs, and daily maintenance schedules to verify that equipment necessary to minimize plant risk was operable or available as required during planned and emergent maintenance activities. The inspectors also conducted plant tours to verify that equipment necessary to minimize risk was available for use. The following six activities were reviewed:

- Planned maintenance on Control Room Heating, Ventilation and Air Conditioning Charcoal Filter Testing (RT-85), Steam Generator Pressure Instrumentation Calibrations (RI-5), and West Engineered Safeguards Room Cooler Fan testing during the week of January 6, 2003;
- Planned outage on Switchyard 345 Kilovolt (KV) Rear-Bus, Emergency Diesel Generator 1-1 Testing, and Traveling Screen F-4C during the week of February 2, 2003;
- Planned troubleshooting on the Auxiliary Feedwater Actuation System Automatic Test Circuitry on February 11, 2003;
- Scheduled maintenance activities from February 16-19, 2003, during the Unusual Event for frazil ice on the intake crib which caused reduced flow from the ultimate heat sink;
- Scheduled maintenance activities on March 18, 2003, during the declared Alert due to a fire in the protected area which resulted in the loss of all charging pumps and on March 19-23, 2003, during the first primary coolant system reduced inventory period in the scheduled refueling outage; and
- Scheduled maintenance activities on March 25-27, 2003, during the declared Alert due to a loss of offsite power.

The inspectors discussed plant configuration control for the maintenance activities with operations, maintenance and work control center staff to verify that work activities were appropriately controlled.

In addition, the inspectors reviewed select condition reports to verify that problems identified during the work activities were appropriately characterized and entered into the licensee's corrective action program.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Non-Routine Plant Evolutions and Events (71111.14)

.1 Operator Response to Unexplained Lowering Level In Service Water Bay

a. Inspection Scope

The inspectors observed operator performance during an unexplained lowering level in the service water bay on February 16, 2003. This event occurred while the plant was at full power and the inspectors responded to the site after licensee personnel declared an Unusual Event. The inspectors assessed the operator's response to verify that procedural guidance contained in the Alarm Response, Off-Normal, System Operating and General Operating Procedures was appropriate and adhered to during this event.

The inspectors also reviewed the control room logs and event notifications to the local and state municipalities on the declared Unusual Event to verify the event was accurately described. In addition, the inspectors reviewed the resultant condition reports which were initiated to verify that these issues were entered into the corrective action program with the appropriate characterization and significance.

Licensee personnel subsequently determined that the lowering level in the service water bay was due to frazil ice buildup on the intake crib. This issue is discussed further in Section 4OA3.4 of this report.

b. Findings

No findings of significance were identified.

.2 Operator Response to Loss of All Charging Pumps

a. Inspection Scope

The inspectors reviewed actions taken by the control room operators in response to the unexpected loss of all charging pumps during solid plant operations on March 18, 2003, while the plant was in Mode 5, "Cold Shutdown." The inspectors assessed the operator's response to determine if the actions taken were as required by plant procedures and to determine if any human performance deficiencies contributed to the unplanned event. This issue resulted in a declared Alert on March 18, 2003, which is discussed further in Section 4OA3.5 of this report.

b. Findings

The loss of all charging pumps event was the subject of an NRC special inspection and any findings related to operator performance during the event will be documented in Inspection Report 50-255/03-05(DRP).

.3 Operator Response to Loss of Offsite Power With Resultant Loss of Shutdown Cooling

a. Inspection Scope

The inspectors observed the control room operators' response following the unexpected loss of offsite power which resulted in the loss of shutdown cooling on March 25, 2003. The inspectors verified that control room operators actively used and accurately implemented Off-Normal Procedure 17, "Loss of Shutdown Cooling," and Off-Normal Procedure 2.1, "Loss of AC [Alternating Current] Power," as required to restore the shutdown cooling system and mitigate the event in a timely manner. The inspectors assessed the operator's response to determine if any human performance deficiencies contributed to the unplanned event. This issue resulted in a declared Alert on March 25, 2003, which is discussed further in Section 4OA3.6 of this report.

b. Findings

The loss of offsite power event was the subject of an NRC special inspection and any findings regarding operator performance while responding to the event will be documented in Inspector Report 50-255/03-05(DRP).

.4 Restoration of Offsite Power Sources to Safety-Related Busses

a. Inspection Scope

On March 27, 2003, the inspectors observed the control room operators during a planned evolution to align offsite power to the safety-related 2400 Volt Busses 1C and 1D and then unload and secure the emergency diesel generators. The planned evolution was performed after offsite power to the startup transformers was restored following the loss of all offsite power event that occurred on March 25, 2003. The inspectors verified that control room operators actively used and accurately implemented Standard Operating Procedure 22, "Emergency Diesel Generators," and Off-Normal Procedure 2.1, "Loss of AC Power," as required.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15Q)

a. Inspection Scope

The inspectors reviewed five operability assessments as documented in the associated condition reports for the following risk significant plant equipment and analyses:

- Service Water System;
- Diesel Generator 1-2 Room Ventilation Dampers;
- Computer Code Errors Discovered in the Accident Analysis;
- Component Cooling Water System; and

- Component Cooling Water Heat Exchanger Service Water Outlet Control Valve CV-0826.

The inspectors interviewed the cognizant engineers and reviewed the supporting documents to assess the adequacy of the operability assessments for the current plant mode. The inspectors also reviewed the applicable sections of the Technical Specifications, Updated Final Safety Analysis Report, and Design Basis Documents to verify that the operability assessments were technically adequate and that the components remained available, such that no unrecognized increase in plant risk had occurred.

The inspectors reviewed select condition reports to verify that identified problems regarding operability evaluations were entered into the licensee’s corrective action program with the appropriate significance characterization. The inspectors also verified that identified corrective actions were reasonable and had been implemented in a manner commensurate with safety.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the engineering analyses, safety analyses, modification documents and design change information associated with the following permanent modification to the Low Pressure Safety Injection Pump P-67B:

- Install Casing Vent Valves on Low Pressure Safety Injection Pumps P-67A and P-67B.

The inspectors discussed the modifications with the appropriate licensee personnel. In addition, the inspectors reviewed the applicable sections of the Updated Final Safety Analysis Report to verify that the modifications would not adversely impact the system’s safety functions.

Further, the inspectors reviewed condition reports associated with installation of the modification to verify that identified problems associated with the modifications were appropriately characterized and entered into the licensee’s corrective action program.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed portions of post maintenance testing and reviewed documented testing activities following scheduled maintenance to determine whether the tests were performed as written. The inspectors also verified that applicable testing prerequisites were met prior to the start of the tests and that the effect of testing on plant conditions was adequately addressed by control room staff. The following five post maintenance test activities were reviewed:

- High Pressure Safety injection Pump P-66A;
- Emergency Diesel Generator 1-2;
- Reactor Protection System and Auxiliary Feedwater Actuation System Setpoint Adjustments;
- Service Water Bay Low Level Alarm Setpoint Change; and
- Low Pressure Safety Injection Pump P-67B.

The inspectors reviewed post maintenance testing criteria to verify that the test criteria was appropriate with respect to the scope of work performed and that the acceptance criteria were clear.

In addition, the inspectors reviewed the completed tests and procedures to verify that the tests adequately verified system operability. Documented test data was reviewed to verify that the data was complete and that the equipment met the procedure acceptance criteria, which demonstrated that the equipment was able to perform the intended safety functions.

Further, the inspectors reviewed condition reports regarding post maintenance testing activities to verify that identified problems were appropriately characterized.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Review of Outage Plan and Monitoring Of Shutdown Activities

a. Inspection Scope

The inspectors reviewed the results from the Probabilistic Safety Assessment Group's review of the 2003 refueling outage schedule and various 3-day and 2-week look ahead assessments. The assessments were conducted to verify that plant equipment required by General Operating Procedure (GOP) 14, "Shutdown Cooling Operations" was not adversely impacted by the scheduled activities, and that plant risk was appropriately considered and minimized during the scheduled outage activities. The inspectors reviewed the licensee's responses to Generic Letter (GL) 88-17, "Loss of Decay Heat

Removal," and plant procedures to verify that previous commitments were in place and adequately addressed the recommendations referenced in GL 88-17.

The inspectors observed portions of the plant shutdown and subsequent cooldown at the start of the refueling outage to verify that the evolutions were completed in accordance with plant procedures. The inspectors also reviewed primary coolant system temperature data to verify that Technical Specification plant cooldown limits were adhered to and condition reports to verify that identified problems were entered into the licensee's corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

.2 Licensee Control Of Outage Activities

a. Inspection Scope

The inspectors assessed the following aspects of the licensee's outage activities:

- Equipment Configuration Management: The inspectors verified that equipment designated in GOP-14, "Shutdown Cooling Operations," was maintained available as required to minimize plant risk;

- Reactor Coolant System Temperature and Level Instrumentation: The inspectors verified that reactor coolant system temperature, level and pressure indication were available and actively being used to accurately monitor plant conditions;

- Electrical Power Availability: The inspectors verified that the configuration of the electrical system was maintained to ensure equipment necessary to minimize plant risk remained operable;

- Decay Heat Removal System Monitoring: The inspectors monitored Shutdown Cooling System parameters to verify the system was operating properly;

- Spent Fuel Pool Cooling System Operation: The inspectors verified that methods to recover spent fuel pool cooling and inventory existed and that equipment necessary for cooling was available and not adversely affected by ongoing outage activities;

- Reactor Coolant System Inventory Control: The inspectors verified that plant equipment needed for primary coolant system inventory control was appropriately maintained during periods of higher risk such as during mid-loop operations; and

- Reactivity Control: The inspectors verified that the licensee identified and implemented the appropriate administrative controls on potential boron dilution paths.

b. Findings

No findings of significance were identified.

.3 Reduced Inventory and Mid-Loop Conditions

a. Inspection Scope

The inspectors observed the control room operators during the primary coolant system drain down to reduced inventory to verify that the operators maintained positive control of the primary coolant system level. The inspectors also verified that the configuration of plant equipment was in accordance with GOP 14, "Shutdown Cooling Operations," during reduced inventory and mid-loop primary coolant system conditions. In addition, the inspectors verified that the licensee's procedures were appropriate and implemented as prescribed for the following activities:

- Containment closure capability was in place for the mitigation of radioactive releases, which included appropriate staging of personnel and equipment, and current lists available in the control room of inoperable containment penetrations and of cables through the equipment hatch;

- At least two independent, continuous indications of primary coolant system temperature and level were available; and

- At least two additional means of adding inventory to the primary coolant system were available, in addition to the residual heat removal system.

The inspectors also verified that Off-Normal Procedures were available which addressed reduced inventory operation and that contingency plans existed to re-power vital electrical busses if the primary source of electrical power was lost.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Fuel Handling Ventilation System Testing

a. Inspection Scope

The inspectors observed portions of the Fuel Handling Ventilation System testing to verify that the testing was conducted in accordance with prescribed procedures.
The inspectors reviewed the licensee and vendor test procedures to verify that testing was done in accordance with Regulatory Guide 1.52, Revision 2 and the American

Society of Mechanical Engineers (ASME) Standard N-510-1989 as required by Technical Specifications. The inspectors also reviewed the documented test data for the Technical Specification surveillance test procedures and the associated basis documents to verify that testing acceptance criteria were satisfied.

In addition, the inspectors reviewed applicable portions of Technical Specifications, the Updated Final Safety Analysis Report, and Design Basis Documents to verify that the surveillance tests adequately demonstrated that system components could perform designated safety functions.

Further, the inspectors reviewed selected condition reports regarding the fuel handling ventilation surveillance testing activities to verify that conditions adverse to quality were appropriately entered into the licensee's corrective action program and that identified corrective actions were reasonable and had been implemented in a manner commensurate with significance.

b. Findings

Introduction

The inspectors identified a Green finding that is being treated as a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," for the failure to ensure that testing of the fuel handling area ventilation system was performed in accordance with test procedures which incorporated the appropriate requirements and acceptance limits specified in Technical Specification 5.5.10, "Ventilation Filter Testing Program."

Description

The inspectors reviewed and observed the licensee perform Procedure RT-85C, "Technical Specification Surveillance Procedure - Fuel Handling Area Ventilation System Filter Testing," on March 3, 2003. The purpose of the surveillance test was to demonstrate operability of the Fuel Handling Ventilation System in accordance with the Surveillance Requirements of Technical Specification 3.7.12, "Fuel Handling Area Ventilation System," and the Ventilation Filter Testing Program specified in Section 5.5.10 of the Technical Specifications. The inspectors noted that the surveillance test was performed with the use of licensee system engineers concurrent with a contractor and contractor specific procedures for certain portions of the test.

The inspectors identified numerous issues associated with the testing of the Fuel Handling Ventilation System which included: (1) human performance issues by the licensee and contractor personnel associated with performing the test in accordance with the test procedures; (2) test procedures which did not incorporate all the requirements and acceptance limits contained in Regulatory Guide 1.52, Revision 2 and the ASME Standard N-510-1989, as required; (3) lack of contractor oversight by licensee personnel during portions of the test to assure that the testing was performed as required; and (4) inadequate material condition of specific portions of the Fuel Handling Ventilation System. Consequently, the inspectors concluded that the surveillances performed on the fuel handling ventilation system were inadequate. Some specific issues the inspectors identified included the following:

- Several instances where procedure steps were not implemented as prescribed and were performed out of sequence for the 'Continuous Use' procedures;
- Sampling of the filter testing 'challenge gas' downstream of the charcoal and high efficiency particulate air (HEPA) filters was performed in a manner that air outside the filter housings diluted the downstream filter test results;
- Upon completion of the required visual inspection of the Fuel Handling Ventilation System by the licensee and contractor as satisfactory with minor exceptions, the inspectors identified the following issues:
  - a previously unidentified through-wall tear on the expansion joint for the suction of Ventilation Fan V-8B;
  - a tear and hole on the east side of the exhaust ductwork from Ventilation Fan V-8B; and
  - a hole on the southwest corner of the exhaust ductwork from Ventilation Fan V-8B.

  The holes and tears identified by the inspectors constituted bypass paths around the Fuel Handling Ventilation System to the atmosphere.
- Contractor's procedure utilized to perform the HEPA and charcoal filter efficiency tests was written to ASME Standard N-510-1975 as opposed to ASME Standard N-510-1989 which was required by Technical Specifications; and
- No evidence that an air-aerosol mixing uniformity test had been performed as required by ASME Standard N-510-1989 as a prerequisite for performing HEPA and charcoal filter efficiency tests. In addition, licensee personnel could not identify an exemption from the Office of Nuclear Reactor Regulation for not having to perform the test.

Licensee personnel subsequently initiated condition reports for the inspector identified issues which were entered into the licensee's corrective action program. In addition, condition reports were generated for additional deficiencies that were identified by licensee personnel who reviewed the circumstances surrounding the inspector identified issues associated with this testing. As a result, licensee personnel subsequently revised the test procedures and re-performed the tests, including the air-aerosol mixing uniformity test, to demonstrate that the Fuel Handling Ventilation System was operable for refueling operations.

The inspectors determined that the failure to ensure that testing was performed for the Fuel Handling Ventilation System in accordance with written test procedures which incorporated the applicable requirements and acceptance limits was a licensee performance deficiency warranting a significance evaluation.

<u>Analysis</u>

The inspectors determined that this finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," issued on February 21, 2003, because the finding affected the barrier integrity cornerstone and if left uncorrected would become a more significant safety concern.

Specifically, human performance deficiencies identified by the inspectors resulted in the failure to ensure that the fuel handling area ventilation system testing was performed in accordance with test procedures which incorporated the appropriate requirements and acceptance limits. Consequently, the radiological barrier function provided by the fuel handling area ventilation system was degraded and was not being tested adequately, and if left uncorrected would become a more significant safety concern.

The inspectors evaluated the finding using Manual Chapter 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening. The inspectors determined that although the barrier integrity cornerstone was adversely impacted, the finding represented a degradation of only the radiological barrier function provided for the spent fuel pool.

Therefore, the finding was determined to be of very low safety significance (Green).

Enforcement

10 CFR 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to this, on March 3, 2003, the licensee failed to assure that testing was performed in accordance with the written test procedures for the Fuel Handling Ventilation System. In addition, the licensee failed to assure that the written test procedures incorporated the requirements and acceptance limits as required by Technical Specification 5.5.10, "Ventilation Filter Testing Program."

This violation is associated with an inspector identified finding that is characterized by the significance determination process as having very low safety significance (Green) and is being treated as a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV50-255/03-02-02)

This issue was entered in the licensee's corrective action program as Condition Reports CAP033742, CAP033749, CAP033764, CAP033788, CAP033829, CAP033895, CAP033907, and CAP033910.

.2 Other Routine Surveillance Tests

a. Inspection Scope

The inspectors reviewed seven surveillance testing activities conducted on the following risk-significant plant equipment:

- Emergency Diesel Generator 1-1;
- Reactor Protection System Steam Generator Pressure Channel D Calibration;
- Reactor Protection System Pressurizer Pressure Channel D Calibration;
- Emergency Air Lock Local Leak Rate Test;
- Auxiliary Feedwater Actuation System;

- Control Room Heating, Ventilation and Air Conditioning System; and
- Safety Injection Tank Outlet Check Valve.

The inspectors observed portions of the testing in the plant to verify that testing was conducted in accordance with prescribed procedures. The inspectors also reviewed the documented test data for the Technical Specification Surveillance Test procedures and the associated basis documents to verify that testing acceptance criteria were satisfied.

In addition, the inspectors reviewed applicable portions of Technical Specifications, the Updated Final Safety Analysis Report and Design Basis Documents to verify that the surveillance tests adequately demonstrated that system components could perform designated safety functions.

Further, the inspectors reviewed selected condition reports regarding surveillance testing activities to verify that identified problems were entered into the corrective action program with the appropriate significance characterization and that the identified corrective actions were reasonable and had been implemented as scheduled.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the documentation and associated 10 CFR 50.59 evaluation for the following temporary plant modification:

- 2003-003, Steam Generator Narrow Range Level Setpoint Changes.

The inspectors verified that the temporary modification did not adversely impact other safety-related equipment and that the modification was being controlled in accordance with Administrative Procedure 9.31, “Temporary Modification Control,” requirements.

In addition, the inspectors reviewed selected condition reports regarding temporary modifications to verify that identified problems were appropriately characterized.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns, Radiological Boundary Verification, and Radiation Work Permit (RWP) Reviews

a. Inspection Scope

The inspectors reviewed the station's implementation of physical and administrative controls over access to radiologically restricted areas (RRAs), including worker adherence to these controls, by reviewing station procedures, RWPs, electronic dosimetry alarm set points, and walking down radiologically significant areas (radiation areas, high radiation areas (HRAs), and locked HRAs) of the station. Specifically, areas in the Reactor Building and the Auxiliary Building were observed to verify these areas were posted and controlled in accordance with 10 CFR Part 20, licensee procedures, and Technical Specifications.

Additionally, the inspectors reviewed the licensee's assessment of an airborne radioactivity area created around the containment cavity after flood-up, which had the potential to result in doses greater than 50 millirem committed effective dose equivalent, to evaluate the effectiveness of engineering controls (High-Efficiency Particulate Air (HEPA) ventilation units) and respiratory protection used to mitigate worker internal dose.

b. Findings

No findings of significance were identified.

.2 High Risk Significant, High Dose Rate (HDR)-Locked HRA and Very HRA Controls

a. Inspection Scope

The inspectors reviewed the station's implementation of physical and administrative controls over access to HDR-locked HRAs and Very HRAs, including a discussion of these controls with Radiation Protection (RP) supervisors and lead RP technicians, to verify that processes and procedures (including any recent changes) implementing these controls provided an appropriate level of worker protection. The inspectors conducted walkdowns of all accessible HDR-locked HRA boundaries to verify adequate posting and control of all entrances into these areas. Additionally, the inspectors reviewed selected plant survey maps to confirm that no Very HRAs existed in the current plant configuration as discussed with the RP staff.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed Condition Reports (CRs) completed in conjunction with the refueling outage which focused on access control to radiologically significant areas, radiation worker practices, and RP technician practices. The inspectors reviewed these documents to assess the licensee's ability to identify repetitive problems, contributing causes, the extent of conditions, and implement corrective actions intended to achieve lasting results.

Additionally, the inspectors reviewed the licensee's root cause evaluation related to a November 2002 event in which two workers received unintended occupational exposures of 134 millirem and 174 millirem, respectively, following an at-power containment entry to repair a defective weld. The licensee reported the event as an Occupational Radiation Safety Performance Indicator Unintended Exposure Occurrence in the 4th Quarter 2002 and the inspectors reviewed the evaluation to determine if there was an overexposure or substantial potential for an overexposure related to the event.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Radiological Work/ALARA Planning

a. Inspection Scope

The inspectors reviewed the station's procedures for radiological work/ALARA planning and scheduling, and evaluated the dose projection methodologies and practices implemented for the refueling outage, to verify that sound technical bases for outage dose estimates existed. Specifically, the inspector reviewed six radiologically significant Radiation Work Permit (RWP)/ALARA planning packages to verify that adequate person-hour estimates, job history files, lessons learned, industry experiences, and the use of mockups (where applicable) were utilized in the ALARA planning process and to confirm that these elements were integrated into the associated RWPs. The RWP/ALARA planning packages included:

- Disassemble and Move Reactor Head to Stand (RWP P03-5102);
- Reactor Head Reassembly and Refueling Close-Out Activities (RWP P03-5108);
- Scaffold Work in Containment (RWP P03-5306);
- Install and Remove Nozzle Dams in [Steam Generators] E-50A/B (RWP P03-5150);
- Removal/Install ICI [In-Core Instrumentation] Flanges and Associated Tasks (RWP P03-5111); and
- NDE [Non-Destructive Evaluation] Bare Metal Inspection on N-50 Reactor Head (RWP P03-5503).

The inspectors also attended the mid-outage Station ALARA Committee Meeting and reviewed RP/Operations-coordinated dose rate reduction activities (e.g., shield package installation, timing of forced oxidation) to further assess inter-departmental coordination and ownership in the radiological work/ALARA planning and scheduling processes.

b. Findings

No findings of significance were identified.

.2 Job Site Inspections and ALARA Controls

a. Inspection Scope

The inspectors observed work activities in the RRA that were performed in radiation areas, HRAs, and locked HRAs to evaluate the use of ALARA controls. Specifically, the inspectors reviewed radiological surveys, attended pre-job radiological briefings, and assessed job site ALARA controls, at least in part, for the following work activities:

- Removal of Reactor Head Inspection Platforms (RWP P03-5100 - Westinghouse Crane and Rigging Activities);
- Steam Generator Eddy Current Testing (ECT) (RWP P03-5152 - Installation of ROSA [Remotely Operated Service Arm], ECT and Tube Plugging);
- Steam Generator Secondary Side Sludge Lance Operations (RWP P03-5155 - Secondary Side Steam Generator Inspection); and
- Perform Under Reactor Head Inspections/Measurements (RWP P03-5503 - NDE Bare Metal Inspection on N-50 Reactor Head).

Worker instruction requirements, including protective clothing, engineering controls to minimize dose exposures, the use of predetermined low dose waiting areas, as well as the on-the-job supervision by the work crew leaders and RP technicians, were observed to determine if the licensee had maintained the radiological exposure for these work activities ALARA. Enhanced job controls including RP technician use of electronic teledosimetry and cameras was also evaluated to assess the licensee's ability to maintain real time doses ALARA in the field.

b. Findings

No findings of significance were identified.

.3 Radiation Worker Performance

a. Inspection Scope

The inspectors observed radiation workers performing the activities described in Section 2OS2.2 and evaluated their awareness of radiological conditions, personal electronic dosimetry alarm set points, and their implementation of applicable radiological controls.

b. Findings

No findings of significance were identified.

.4 Verification of Dose Estimates, Dose Trending, and Dose Tracking Systems

a. Inspection Scope

The inspectors reviewed the licensee's total outage dose estimates, selected individual job dose estimates and the related dose trending for the refueling outage. The ALARA In-Progress reviews for RWP Nos. P03-5102 and P03-5150 were examined to evaluate the licensee's ability to assess the effectiveness of the ALARA plans in a timely manner and institute changes in the plan or its execution, if warranted. The licensee's dose tracking system was also reviewed to determine if the level of dose tracking detail, dose report timeliness, and report distribution were sufficient to support the control of collective and individual dose.

b. Findings

No findings of significance were identified.

.5 Declared Pregnant Worker Program

a. Inspection Scope

The inspectors reviewed the controls implemented by the licensee for workers who voluntarily entered the licensee's fetal protection program. The inspectors assessed the licensee's adherence to the requirements contained in 10 CFR 20.1208 and station procedures by reviewing the licensee's tracking and evaluation of the dose to the workers' embryos/fetuses. Specifically, the inspectors examined the licensee's program to ensure that any declared pregnant workers' monthly and cumulative exposure for the gestation period were controlled so as not to exceed regulatory limits.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed licensee CRs completed in conjunction with the refueling outage which focused on ALARA planning and controls. The inspectors reviewed these documents to assess the licensee's ability to identify repetitive problems, contributing causes, the extent of conditions, and develop corrective actions intended to achieve lasting results.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection (PP)

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspectors reviewed Revision 47 to the Palisades Nuclear Plant Security Plan to verify that changes did not decrease the effectiveness of the submitted document. The referenced revision was submitted in accordance with 10 CFR 50.54(p) by licensee letter dated January 10, 2003.

1. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors verified that the data submitted by the licensee was accurate and complete for the following two Performance Indicators:

- Emergency Diesel Generator Unavailability; and
- Safety System Functional Failures.

The inspectors reviewed control room logs, licensee monthly operating reports, and the licensee’s corrective action system to verify that the licensee had accurately reported the emergency diesel generator unavailability and safety system functional failures performance indicators for January 2002 through December 2002.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Right Channel Hydrogen Monitor Line Caps Found Removed

a. Inspection Scope

The inspectors reviewed the root cause evaluation associated with the following Condition Report:

- CAP0322269, "Right Channel Monitor Line Caps Found Removed During Technical Specification Surveillance Test RT-71P."

The inspectors verified that the following attributes were adequately addressed in the licensee's evaluation and associated corrective actions:

- Consideration of extent of condition, generic implications, common cause and previous occurrences;
- Classification and prioritization of problem resolution was commensurate with safety significance;
- Root and contributing causes were identified;
- Corrective actions were appropriately focused to correct the problem and were implemented in a timely manner commensurate with the safety significance; and
- Implementation of longer term corrective actions appear appropriate and adequate compensatory actions were in place to minimize a problem until the permanent action was completed.

The inspectors also discussed the corrective actions and associated evaluations with applicable site personnel including the condition report evaluators and system engineers.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report (LER) 02-003, "Inoperable Containment Hydrogen Monitors"

a. Inspection Scope

The inspectors reviewed Licensee Event Report 02-003, "Inoperable Containment Hydrogen Monitors," to verify that the event was accurately described, to determine if any violations of NRC requirements had occurred and to assess the appropriateness of identified corrective actions.

b. Findings

Introduction

The inspectors determined that a self-revealed Green finding was associated with a Non-Cited Violation of Technical Specification Section 3.3.7 for the failure to restore an inoperable channel of containment hydrogen monitoring within the allowed outage times contained in Technical Specification Action Statements 3.3.7.A and 3.3.7.D.

Description

On December 21, 2002, during the performance of a surveillance test, the licensee discovered that the pipe caps for two test taps on the right channel containment hydrogen monitor instrument lines were not installed. The uninstalled pipe caps prevented the surveillance from being completed satisfactorily which prompted licensee personnel to identify the problem.

The containment hydrogen monitor is utilized to sample the containment atmosphere for hydrogen concentrations following a design basis accident to determine if additional actions are needed to abate high hydrogen concentrations. The two test taps were located between the outermost containment isolation valve and the next control valve on the hydrogen monitor instrument line. Therefore, if the right channel hydrogen monitor was placed in service following an accident then the following would have occurred:

- The first test tap without a pipe cap would result in a hydrogen sample being returned to the auxiliary building instead of being returned to containment;
- The second test tap without a pipe cap could open a path from the auxiliary building atmosphere to the hydrogen monitor sample pump such that when containment was at or below auxiliary building pressure, the hydrogen sample would be diluted and the hydrogen monitor would display a value less than the actual hydrogen concentration in containment; and
- Following an accident, if containment pressure was greater than auxiliary building pressure then both taps could be pressurized by containment atmosphere resulting in a release to the auxiliary building.

Licensee personnel completed a root cause evaluation for this event and concluded that the pipe caps were not placed back on the test taps after the Containment Integrated Leak Rate Test, Test RT-36, was last performed in the April 2001 refueling outage. Licensee personnel also concluded that the failure to reinstall the test tap pipe caps was attributable to human performance failures involving inadequacies in procedure compliance and independent verification policies when completing prescribed procedure actions. The inspectors determined that the conclusions were reasonable.

The inspectors determined that the failure to ensure that the right channel hydrogen monitoring system was aligned properly after the system was restored from testing during the 2001 refueling outage was a self-revealed licensee performance deficiency which warranted a significance evaluation.

Analysis

The inspectors determined that this finding was greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” issued on February 21, 2003. The finding dealt with maintaining the functionality of containment and was related to the barrier integrity cornerstone attribute regarding configuration control. The cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events was affected.

Specifically, the human performance deficiencies which resulted in the failure to ensure that the pipe caps were installed after testing was completed in April 2001 resulted in the right channel hydrogen monitor being inoperable. Consequently, the hydrogen monitor was incapable of measuring a representative hydrogen concentration in the containment atmosphere following an accident. In addition, the uninstalled pipe caps also represented a potential radiological release path if the hydrogen monitor had been placed in service following an accident.

The inspectors evaluated the finding using Manual Chapter 0609, “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening. The inspectors determined that the finding represented an actual open pathway in the physical integrity of reactor containment which required additional analysis using Manual Chapter 0609, Appendix H, “Containment Integrity SDP.” Utilizing Appendix H, the inspectors determined the finding was potentially risk significant.

A Region III Senior Reactor Analyst in conjunction with the inspectors performed a Phase 3 assessment. Utilizing NUREG-1675, “Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP),” the analyst determined that a containment leak rate of 100 volume percent per day constitutes the threshold for LERF significance. The 100 volume percent per day leakage rate is approximately equivalent to an opening size in the containment of 2.5 to 3 inches in diameter for Pressurized Water Reactors with large dry containments, thus the significance of this finding was found to be very low as the actual combined open pathways from both test taps was less than 0.5 inches in diameter.

Therefore, the finding was determined to be of very low safety significance (Green).

Enforcement

Technical Specification 3.3.7 requires, in part, that two containment hydrogen monitor channels be operable in Modes 1, 2 and 3. Technical Specification Action Statement 3.3.7.A requires, in part, that when one or more functions with one required containment hydrogen monitor channel is inoperable then the required channel be restored to an operable status within 30 days or a report to the NRC be initiated in accordance with Technical Specification 5.6.6. In addition, Technical Specification Action Statement 3.3.7.D requires, in part, that when two containment hydrogen monitor channels are inoperable that one channel is restored to operable status within 72 hours or be in hot standby within the next 6 hours and in hot shutdown within the following 36 hours.

Contrary to the above, from May 2001 through June 2001 and January 2002 through December 2002, while the plant was in Mode 1, the right channel containment hydrogen monitor was inoperable because the pipe caps for two test taps were not installed. In addition, in November 2002, while the plant was in Mode 1 with the right channel containment hydrogen monitor inoperable, the left channel containment hydrogen monitor was also inoperable for approximately 200 hours because of scheduled maintenance.

This violation is associated with a self-revealed finding that is characterized by the significance determination process as having very low safety significance (Green) and is being treated as a Non-Cited Violation of Technical Specification 3.3.7, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 50-255/03-02-03)

This issue was entered in the licensee's corrective action program as Condition Report CAP0322269, "Right Channel Monitor Line Caps Found Removed During Technical Specification Surveillance Test RT-71P." This LER is closed.

.2 <u>(Closed) LER 02-002, "Automatic Reactor Trip and Safety System Actuation"</u>

a. <u>Inspection Scope</u>

The inspectors reviewed LER 02-002 to verify that the event was accurately described, determine if any violations of NRC requirements had occurred and to assess the appropriateness of identified corrective actions.

This issue was previously documented in Inspection Report 50-255/02-09 as a self-revealed Green finding and associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to rigorously evaluate industry operating experience information which resulted in inadequate preventive maintenance activities being developed for the 345 KV transmission lines that connect the plant and switchyard.

The inspectors did not identify any new information in the LER which was of concern and concluded that the completed and planned corrective actions were reasonable. Therefore, this LER is closed.

b. <u>Findings</u>

No findings of significance were identified.

.3 <u>(Closed) LER 03-001, "Inoperable Steam Generator Low-Level Channels"</u>

a. <u>Inspection Scope</u>

The inspectors reviewed LER 03-001 to verify that the event was accurately described, determine if any violations of NRC requirements had occurred and to assess the appropriateness of identified corrective actions.

On January 15, 2003, at 8:15 p.m. licensee personnel identified that all four steam generator reactor protection system low-level trip setpoints on both steam generators were set below the allowable value specified in Technical Specification 3.3.1, “Reactor Protective System Instrumentation.” Consequently, all steam generator low-level trip instrument channels were declared inoperable and Technical Specification 3.0.3 was entered which required action be initiated within 1 hour to shut down the plant to Mode 3, “Hot Standby,” within 7 hours.

Licensee personnel requested and received enforcement discretion from the NRC to extend the completion times in Technical Specification 3.0.3 by an additional 36 hours in order to conduct online repairs. Licensee personnel subsequently adjusted the trip setpoints to comply with Technical Specification requirements and the steam generator low-level reactor protection system trip functions were declared operable within the extended time allowed by the enforcement discretion. The enforcement discretion related to this event is discussed in Section 4OA5.3 of this report.

Licensee personnel evaluated this event and subsequently determined that the low-level trip setpoints had been set incorrectly since 1998. The event was caused by a vendor calculation error that was not previously identified and resulted in a biasing correction factor being applied to the level transmitters in the non-conservative direction.

This finding was more than minor because it affected the mitigating systems cornerstone objective by impacting the reactor protection system’s capability to respond to a steam generator low-level initiating event to prevent undesirable consequences. However, while the steam generator low-level setpoints were below the allowable values specified in Technical Specifications, all the low-level setpoints were greater than the analytical values specified in the plant safety analysis. Therefore, plant safety would have been maintained during a steam generator low level initiating event.

The finding affected the mitigating systems cornerstone and was determined to be of very low safety significance (Green) using NRC Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” because the finding:

- was not a design or qualification deficiency;
- did not represent the loss of the reactor protection system’s safety function; and
- did not screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating event.

This licensee identified finding involved a violation of Technical Specification 3.3.1, “Reactor Protective System Instrumentation,” and the related enforcement aspects are discussed in Section 4OA7 of this report. This LER is closed.

b. Findings

No findings of significance were identified.

.4 Declared Unusual Event Due to Unexplained Service Water Bay Lowering Level

a. Inspection Scope

The inspectors reviewed the circumstances associated with the Unusual Event on February 16, 2003, which was declared due to the unexplained lowering level in the service water bay. The resident inspectors provided 24-hour site coverage after the Unusual Event was declared to monitor the licensee's event response until the Unusual Event was terminated on February 19, 2003.

The inspectors reviewed the circumstances surrounding this event to assess whether any licensee performance issues existed which contributed to the initiation of the event. Licensee operator response during the event was assessed under "Personnel Performance Related to Non-Routine Plant Evolutions and Events," in Section 1R14.1 of this report.

b. Findings

Introduction

No findings of significance or licensee performance deficiencies were identified during the inspectors' review of the events which occurred. However, the inspectors did note that NUREG-0820, "Integrated Plant Safety Assessment - Systematic Evaluation Program - Palisades Plant," refers to the ability of the plant to utilize a warm water recirculation pump as an alternate supply for the service water pumps if catastrophic damage were to occur to the intake pipe or other components that completely block intake flow. However, the warm water recirculation pump currently does not have the ability to perform this function; therefore, the inspectors opened an Unresolved Item to track whether or not the licensee is in nonconformance of commitments made in the Systematic Evaluation Program.

Description

On February 16, 2003, at approximately 1:50 a.m. the Control Room received a service water bay low level alarm. The operators determined that there was reduced flow from the service water intake structure in Lake Michigan as opposed to an equipment problem in the plant. At 02:53 a.m. the Site Emergency Director declared an Unusual Event based on the Miscellaneous Emergency Action Level, due to the lowering service water bay levels. Operators responded by taking the manual actions specified in Alarm Response Procedure 7 and Off-Normal Procedure 6.1, "Loss of Service Water." Operator actions included securing the dilution water pumps, which were the largest service water load, lowering plant power level from 100 percent to approximately 90 percent and initiation of warm water recirculation to the service water bay from the make-up basin through the use of Pump P-5. Actions taken by the operators successfully stabilized level in the service water bay.

On February 16, 2003, divers performed inspections which identified that at least two sides of the intake crib appeared to be encased in frazil ice. Adverse weather conditions prevented the divers from inspecting all four sides of the intake crib.

Licensee personnel also learned that municipal water plants on the eastern shore of Lake Michigan experienced the same phenomenon of frazil ice forming on the intake cribs. The Unusual Event was terminated on February 19, 2003, after inspections were completed of the intake crib and intake piping which revealed that no frazil ice remained and that the intake crib and piping was not damaged.

The inspectors reviewed industry operating experience and concluded that a performance issue did not exist with regard to this event. The licensee's procedures contained the appropriate actions to abate the formation of frazil ice on the intake structure, mainly by reducing the largest service water loads and reducing plant power. However, the inspectors did note that NRC NUREG-0820, "Systematic Evaluation Program for the Palisades Nuclear Plant," contained commitments the licensee made to the NRC during the Integrated Assessment Program which were referenced in Updated Final Safety Analysis Report (UFSAR) Section 1.8.1, "Systematic Evaluation Program," and Table 1.3.

Specifically, Section 4.7.1, "Cooling Water System Structures Inspection," of NUREG-0820, referenced in UFSAR Table 1.3, states in part that the NRC staff determined that inspection of cooling water systems and structures as referenced in Regulatory Guide 1.127 did not have to be made, in part, because "An alternate water supply for the service water pumps is available independent of the normal intake path. This alternate supply is piped to the intake structure from the warm water recirculation pump (can be powered from emergency diesel generator) and takes a suction from the mixing basin or from the lake at the discharge structure."

However, the ability for the warm water recirculation pump to take a suction from the lake was not currently available due to the suction pipe being buried in sand. Also, the ability to take a suction from the mixing basin has not been available since the year 2000 due to the low levels in Lake Michigan.

The licensee initiated Condition Report CAP033802, "Warm Water Recirculation Pump P-5, Potential Commitment Nonconformance," to evaluate this issue. The inspectors opened an Unresolved Item to track the licensee's evaluation of this potential nonconformance of a commitment from the Systematic Evaluation Program.
(URI 50-255/03-02-04)

.5 <u>Declared Alert Due to a Fire With The Potential To Affect Safety-Related Equipment</u>

a. <u>Inspection Scope</u>

The inspectors responded to the plant after an Alert was declared at 8:47 p.m. on March 18, 2003. The plant was in Mode 5, Cold Shutdown, when a fire alarm activated in the cable spreading room. Fire brigade personnel responded and reported a significant amount of smoke in the cable spreading room which prompted the Alert declaration due to a fire with the potential to affect safety-related equipment. The inspectors monitored licensee actions during the event and verified that the shutdown cooling system remained in service to remove reactor decay heat. The event was terminated at 11:26 p.m. on March 18, 2003, after the licensee's preliminary investigation determined that the likely cause of the smoke was the circuit breaker for

Charging Pump P-55A which was found damaged. This event was the subject of an NRC special inspection and the event details were documented in Inspection Report 50-255/03-05(DRP).

b. Findings

Findings associated with this event will be documented in NRC Special Inspection Report 50-255/03-05(DRP).

.6 Declared Alert Due to Loss of Offsite Power

a. Inspection Scope

On March 25, 2003, an Alert was declared at 11:21 a.m. after offsite power to the plant was lost unexpectedly which resulted in a loss of shutdown cooling to the reactor. The inspectors responded to the control room and observed the operator's response to the event to ensure that appropriate plant procedures were utilized in an accurate and timely manner. The inspectors verified that Off-Normal Procedure 17, "Loss of Shutdown Cooling," General Operating Procedure 14, "Shutdown Cooling Operations," and Off-Normal Procedure 2.1, "Loss of AC Power," were implemented as required to restore shutdown cooling and mitigate the event. The inspectors also verified that the Emergency Plan was implemented in an accurate and timely manner.

The inspectors walked down the control room panels to monitor key plant parameters including the primary coolant system temperature heatup rate. The inspectors also verified that the emergency diesel generators were operating properly in order to provide power to plant equipment necessary to re-establish shutdown cooling to the reactor.

The Alert emergency was downgraded to an Unusual Event after shutdown cooling was restored to the reactor with the shutdown cooling pump being powered from the emergency diesel generator. The inspectors provided 24-hour site coverage to monitor licensee actions until the Unusual Event was terminated on March 27, 2003, after offsite power to the plant was restored and the emergency diesel generators were secured. This event was the subject of an NRC special inspection and the event details will be documented in Inspection Report 50-255/03-05(DRP).

b. Findings

Findings associated with this event will be documented in NRC Special Inspection Report 50-255/03-05(DRP).

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 50-255/02-03-02: "Potential 125VDC Single-Failure Impact on Containment Spray"

In Inspection Report 50-255/02-03, Section 1R21.b.2, NRC inspectors documented a potential single-failure scenario whereby the containment spray system would be nonfunctional during a loss-of-coolant accident. At the end of the inspection, this issue was characterized as a finding that had a credible impact on safety, in that the plant may have been operated outside of the design basis. The issue was characterized as unresolved pending an Office of Nuclear Reactor Regulation (NRR) licensing basis determination.

To resolve this issue, discussions were held with NRR to review the Palisades licensing basis and determine whether the plant was licensed with breaker and fuse failures considered as passive. Participants in those discussions were staff from NRR projects and the electrical branch, and Region III staff. As a result of the discussions, NRR confirmed that breakers and fuses which did not have to change state to accomplish the intended safety function, as those involved in the scenario, were considered passive devices.

As discussed in the inspection report, NRR was requested to determine whether a backfit, in accordance with 10 CFR 50.109, to conform with present regulatory and industry guidance was appropriate. It was determined, that because of the extremely low probability of a large break loss-of-coolant accident coincident with the occurrence of the proposed single failure, a backfit could not be justified. This URI is closed.

.2 (Closed) Violation (VIO) 50-255/01-08-01, Smoke Detectors Inadequate - Northwest Portion of Cable Spreading Room

During a triennial fire protection inspection completed in September 2001, the NRC identified a finding of "to be determined" significance and associated Apparent Violation (AV) 50-255/01-08-01 regarding inadequate smoke detectors in the northwest portion of the cable spreading room.

The NRC subsequently characterized the finding as White (i.e. an issue with low to moderate safety significance, which may require additional NRC inspections) in a letter dated October 26, 2001, from the NRC to Mr. Douglas E. Cooper, Site Vice President, Palisades Nuclear Plant, Nuclear Management Company, LLC. A Notice of Violation was also issued with the White finding. Nuclear Management Company did not contest the violation or the characterization (White) of the risk significance for the finding.

On July 25, 2002, the NRC completed a supplemental inspection in accordance with Inspection Procedure 95001 which assessed the adequacy of the corrective actions for the White finding and associated violation. The inspection was documented in Inspection Report 50-255/02-08(DRS).

During the supplemental inspection, the inspectors concluded that the licensee had developed a comprehensive corrective action plan to address the issue as well as any other historical NFPA (National Fire Protection Association) code conformance issues. The inspectors also concluded that adequate measures were in place to prevent a similar occurrence from recurring. Therefore, this violation is closed.

.3 (Closed) Unresolved Item 50-255/03-02-05: "Review of Notice of Enforcement Discretion (NOED) 03-3-001 For Nuclear Management Company LLC Regarding Palisades"

The inspectors reviewed the circumstances associated with issuing NOED 03-3-001 and the basis for the NOED request to determine if a failure to comply with regulatory requirements contributed to the need for enforcement discretion. The inspectors also verified that licensee personnel complied with the compensatory actions noted in the NOED.

On January 15, 2003, at 8:15 p.m. licensee personnel identified that all four steam generator reactor protection system low-level trip setpoints on both steam generators were set below the allowable value specified in Technical Specification 3.3.1, “Reactor Protective System Instrumentation.” Consequently, all steam generator low-level trip instrument channels were declared inoperable and Technical Specification 3.0.3 was entered which required that action be initiated within 1 hour to shut down the plant and that the plant be in Mode 3, Hot Standby, within 7 hours. Licensee personnel requested enforcement discretion to extend the completion times in Technical Specification 3.0.3 by an additional 36 hours to avoid a plant shutdown and conduct repairs online.

The NRC verbally granted NOED 03-3-001 at 12:07 a.m. on January 16, 2003. Licensee personnel subsequently adjusted the trip setpoints to comply with Technical Specification requirements and the steam generator low-level reactor protection system trip functions were declared operable at 8:00 p.m. on January 16, 2003.

No findings of significance were identified during the inspectors’ review of the basis of the NOED request and the licensee's implementation of compensatory actions required by the NOED. Therefore, this URI is closed.

This issue was determined to be a licensee identified finding which involved a violation of Technical Specification 3.3.1, “Reactor Protective System Instrumentation.” The event is discussed further in Section 4OA3.3 of this report and the related enforcement aspects for the violation are discussed in Section 4OA7 of this report.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. D. Cooper and other members of licensee management on April 11, 2003. Licensee personnel acknowledged the findings presented. The inspectors asked licensee personnel whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meeting

Interim Exit was conducted for:

- Physical Protection Inspection with Mr. B. Rowland on January 31, 2003.
- Radiation Protection Inspection with Mr. D. Cooper on April 2, 2003.

4OA7 Licensee Identified Violations

The following finding of very low safety significance (Green) was identified by the licensee and was a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation.

- Technical Specification 3.3.1, “Reactor Protective System Instrumentation,” required, in part, that four reactor protective system trip units including the associated steam generator low-level instrument channels be operable in Mode 1. Contrary to this, on January 15, 2003, with the plant in Mode 1, all steam generator low-level instrument channels on both steam generators were determined to be inoperable. This event was documented in the licensee’s corrective action program as CAP032931, “Level Transmitter Static Pressure Effect Not Incorporated Correctly.” This finding was of very low safety significance because the finding did not represent the loss of the reactor protection system’s safety function.

KEY POINTS OF CONTACT

Licensee

D. Cooper, Site Vice President
P. Harden, Director, Engineering
G.W. Hettel, Manager, Maintenance and Construction
D. G. Malone, Supervisor, Regulatory Assurance
D. J. Malone, General Plant Manager
C. Moeller, ALARA Supervisor
G. Packard, Operations Manager
R. Remus, Assistant Plant Manager
B. Rowland, Security Manager
P. Russell, Manager Performance Improvement

NRC

J. Eads, Project Manager, NRR
E. Forrest, NRR
J. Hayes, NRR
J. Creed, Branch Chief, Safeguards
H. Walker, NRR

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-255/03-02-01	NCV	Inadequate Corrective Actions to Address Scaffolding Control Problems
50-255/03-02-02	NCV	Inadequate Testing of the Fuel Handling Area Ventilation System
50-255/03-02-03	NCV	Failure to Restore an Inoperable Channel of Hydrogen Monitoring
50-255/03-02-04	URI	Nonconformance with Systematic Evaluation Program Commitment (Section 4OA3.4)
50-255/03-02-05	URI	Review of Notice of Enforcement Discretion (NOED) 03-3-001

Closed

50-255/03-02-01	NCV	Inadequate Corrective Actions to Address Scaffolding Control Problems
50-255/03-02-02	NCV	Inadequate Testing of the Fuel Handling Area Ventilation System
50-255/03-02-03	NCV	Failure to Restore an Inoperable Channel of Hydrogen Monitoring
50-255/03-02-02	URI	Potential 125VDC Single-Failure Impact on Containment Spray
50-255/01-08-01	VIO	Smoke Detectors Inadequate - Northwest Portion of Cable Spreading Room
50-255/02-003	LER	Inoperable Containment Hydrogen Monitors
50-255/02-002	LER	Automatic Reactor Trip and Safety System Actuation
50-255/03-001	LER	Inoperable Steam Generator Low-Level Channels
50-255/03-02-05	URI	Review of Notice of Enforcement Discretion (NOED) 03-3-001

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
AV	Apparent Violation
CFR	Code of Federal Regulations
CR	Condition Report
CY	Calendar Year
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECT	Eddy Current Testing
GL	Generic Letter
HDR	High Dose Rate
HEPA	High Efficiency Particulate Air
HRA	High Radiation Area
HSAS	Homeland Security Advisory System
ICI	In-Core Instrumentation
ICM	Interim Compensatory Measures
IMC	Inspection Manual Chapter
LER	Licensee Event Report
LERF	Large Early Release Frequency
NCV	Non-Cited Violation
NDE	Non-Destructive Evaluation
NFPA	National Fire Protection Association
NMC	Nuclear Management Company
NOED	Notice of Enforcement Discretion
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OHS	Office of Homeland Security
RIS	Regulatory Information Summary
ROSA	Remotely Operated Service Arm
RRA	Radiologically Restricted Area
RWP	Radiation Work Permit
SDP	Significance Determination Process
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

Plant Procedures and Miscellaneous Documents

SOP-12	System Operating Procedure - Auxiliary Feedwater	Revision 14
DBD-1.03	Design Basis Document - Auxiliary Feedwater System	Revision 6
M-207, Sheet 2	Piping and Instrument Diagram - Auxiliary Feedwater	Revision 32
M-17, Sheet 41	Auxiliary Feedwater Mini Flow Valves	Revision B
	Open Work Order/Work Requests on the Auxiliary Feedwater System	

Condition Reports Reviewed to Assess Characterization of Identified Problems

CAP033714	MV-ES502 Outlet is not Capped as Shown on PID - 204, Sheet 1B
CAP033541	Technical Issue Regarding the Removal of Auxiliary Feedwater Pumps Start-up Strainers
CAP033346	Discrepancies Noted in Safeguard Rooms by NRC
CAP033503	Licensing Basis Requirements of High Energy Line Break of Auxiliary Feedwater Outside Containment
CAP033667	Seismic Scaffold Does Not Meet Installed Plant Equipment Separation Requirements
CAP033627	Piping and Instrument Diagram Updated Prior to Associated Procedure Change
CAP033630	Auxiliary Feedwater Pumps P-8A and P-8B Cooling Water Outlet Valves not Locked
CAP033744	Seismic Scaffold Found that Did Not Appear to Meet Equipment Separation Requirement

Condition Reports Reviewed to Assess Corrective Actions

CAP030979	Inconsistency Noted Between Configuration of Auxiliary Feedwater and Emergency Core Cooling System Pump Cooling Water Valves

CAP031074	Emergency Feed Supply Line to AFW Pumps Subjected to Sand Accumulation	

1R05 Fire Protection

Plant Procedures

FP-MS-1	Fire Protection Check Sheet - Monthly Inspection and Testing of Fire Doors for Fire Areas 1, 15, 16	Revision 2
FPSP-SI-1	Data Sheet for Alarm Bells and Ionization Smoke Detectors for Fire Areas 1, 15, 16	Revision 2
FPSP-RP-11	Fire Barrier Penetration Seal/Conduit Seal Inspection Form for Fire Areas 1, 8, 15, and 16	Revision 4
FPSP-WP-1	Safety-Related Fire Door Data Sheet Fire Areas 1, 8 and 15	Revision 1
FPSP-SO-2	Safety-Related Fire Door Data Sheet for Fire Areas 1, and 15	Revision 0
ONP-12	Off-Normal Procedure - Acts of Nature	Revision 17
AP-4.02	Administrative Procedure - Control of Equipment	Revision 18
ONP-25.1	Off-Normal Procedure - Fire Which Threatens Safety-Related Equipment	Revision 12
ONP25.2	Off-Normal Procedure - Alternate Safe Shutdown Procedure	Revision 18
FP-MS-1	Fire Protection Check Sheet Monthly Inspection and Testing of Fire Doors Fire Area 1, 15	Revision 2
FPSP-RI-2	Ionization Smoke Detector Locations, Containment Building, Attachment 2	Revision 1
FPSP-RP-12	Fire Rated Assembly/Fire Protection Assembly Checkoff/Comment Sheet	Revision 2
FPSP-RO-8	Containment Building Fire Hose Replacement, Nozzle Inspection and Station Valve Check	Revision 1
FPSP-RO-9	Fire Sprinkler System Inspection	Revision 0

Miscellaneous Documents

EA-PSSA-00-001	Palisades Plant Post Fire Safe Shutdown Summary Report, for Fire Areas 1, 8, 14, 15, and 16	Revision 1

Palisades Plant Fire Hazards Analysis	Analysis for Fire Areas 1, 8, 14, 15, and 16	Revision 4
Palisades Plant Fire Hazards Analysis	Analysis for Fire Areas 1, 8, 14, 15, and 16	Revision 4
BTP ASB 9.5-1	U.S. NRC Branch Technical Position 9.5-1 - Guidelines for Fire Protection for Nuclear Power Plants	Revision 1
	Consumer Power Company - List of Changes and Response to Appendix A to Branch Technical Position APCSB 9.5-1 and Regulatory Guides 1.78 and 1.101	Revision 2 August 24, 1996
FSAR 9.6	Updated Final Safety Analysis Report, Section 9.6 - Fire Protection	Revision 23
EA-APR-98-008, Section 4.4	Analysis of Specific Barrier Segments; "Fire Barrier Segment 147C/149F (Auxiliary Building @ EL 613'6")	

<u>1R07</u> <u>Heat Sink Performance</u>

T-390	Special Test - Single Tube Testing of the Component Cooling Water Heat Exchangers completed March 21, 2003	Revision 0

<u>1R11</u> <u>Licensed Operator Requalification</u>

SIS-JPM-06	Job Performance Measure, "Fill an SI Tank"	Revision 0
PPCS-JPM-01	Job Performance Measure, "Alternate Pressurizer Pressure Channels"	Revision 0
EDG-JPM-01A	Job Performance Measure, "Start and Load a D/G in Parallel"	Revision 0
AP 4.05, Attachment 1	Completed "Operator Performance Evaluations," for 1 SRO and 3 RO's	January 24, 2003

<u>Condition Reports Reviewed to Assess Problem Identification Characterization</u>

CAP033270	Review Expectations for Time to Classify with SEDs and EOF Directors

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

Miscellaneous Documents

	Equipment and System Operational Guidance Recommendation #117, “Intake Crib (Ultimate Heat Sink) Operable but Degraded”	February 17, 2003
	Operations Log Entries for January 7-12, 2003; February 6-7, 11 and 17-21, 2003; March 18-23 and 25-27, 2003	
	Scheduled Maintenance Activities for January 7-12, 2003; February 6-7, 11 and 17-21, 2003; March 18-23 and 25-27, 2003	
	Operators Risk Reports for January 7-12, 2003; February 6-7, 11, 17-21, 2003	
	Shift Activities Sheets, Operations, March 18 - 23 and 25-27, 2003	
GOP-14, Attachment 16	Shutdown Operation Equipment Sheets, “Shutdown Safety Risk Assessment,” March 18-23, and 25-27, 2003	

Condition Reports Reviewed to Assess Characterization of Identified Problems

CAP033470	Bolt Found Missing on Warm Water Recirculation Pump P-5 Protective Screen
CAP034302	Workers Entered Protected Train Barriers without SRO Knowledge
CAP033807	Procedure Revision Not Incorporated Into Ongoing Evaluation
CAP034530	Discrepancies in Diesel Generator Operating Procedure SOP-22

1R14 Non-Routine Plant Evolutions and Events

Plant Procedures

GOP-8	General Operating Procedure - Power Reduction and Plant Shutdown to Mode 2 or Mode 3 $\geq$ 525° Fahrenheit	Revision 18
ARP-7	Alarm Response Procedure - Auxiliary Systems Scheme EK-11 (C-13)	Revision 63
ONP 6.1	Off-Normal Procedure - Loss of Service Water	Revision 10
SOP 15	Service Water System	Revision 25

SOP-22	Emergency Diesel Generators	Revision 34
ONP-2.1	Loss of AC Power	Revision 12
ONP-17	Loss of Shutdown Cooling	Revision 28

Miscellaneous Documents

WO 24113520	Breaker 52-1205: Install Original Breaker When it Arrives from Vendor
WO 24912222	Breaker 52-1205: Remove 52-1205 Breaker, Install Spare

Condition Reports Reviewed to Assess Characterization of Identified Problems

CAP033437	Received EK-1129 Service Water Bay Low Level Alarm
CAP033438	Stop Log Gasket Failure at Mixing Basin
CAP034629	Temporary Modification Not Evaluated for Full Duration of Expected Implementation

1R15 Operability Evaluations

CAP032499	Damper D-22, Diesel Generator 1-2 Room Exhaust Damper is Broken on North Side
CAP032787	Code Error Found in Contempt -LT/28
CAP033644	Component Cooling Water Heat Exchanger Service Water Outlet Control Valve CV-0826 Inoperable
CAP032556	Potential Non-Conservative Minimum Service Water Temperature Used in the Accident Analysis

Miscellaneous Documents

Table 14.17.1-4	Safety Analysis, Chapter 14 Table, “Fan Cooler Capacity Used in the LBLOCA Analysis	Revision 23

Condition Reports Reviewed to Assess Characterization of Identified Problems

CAP033644	Component Cooling Water Heat Exchanger Service Water Outlet Valve CV-0826 Inoperable
CAP033610	Gravity Damper D-29 May Hang Open when Wind is From the North

Condition Reports Reviewed to Assess Corrective Actions

C-PAL-97-1363	Component Cooling Water System Not Analyzed For Present LOCA Containment Analysis	

1R17 Permanent Plant Modifications

EAR 2001-0601	Install Casing Vent Valves on Low Pressure Safety Injection Pumps P-67A and P-67B	
FDC-24210903-1	Field Design Change for EAR 2001-0601	
8-SK-EAR-2001-0601	Drawing - Pump Casing Vent Valve	Revision B
WO 24210903	Install LPSI Pump Casing Vent Valve	

Condition Reports Reviewed to Assess Characterization of Identified Problems

CAP033868	Work Not Released Prior to Commencing Work
CAP033869	LPSI Pump P-67B - As Found Anomalies Discovered During Power End / Seal Replacement
CAP033870	LPSI Pump P-67B Impeller Retaining Nut Found Loose
CAP033668	Discrepancy Between LPSI Pump P-67A and B Vendor Drawing and Maintenance Procedure Drawing
CAP033882	Evidence of Overheating on 52-1101 Circuit Breaker Rear Bus Stabs
CAP033878	152-111 “LPSI P-67B” Anomalous Behavior
CAP033903	Differential Pressure Trend Increasing for LPSI P-67B
CAP033905	Problems Encountered with Vent Line Addition to P-67B Casing
CAP033906	Possible Excessive Play in P-67B Bearing Housing
CAP033943	Final Field Dimensions on P-67B’s Pump Casing Vent Not Acceptable for Growth
CAP033451	LPSI Pump P-67B Case Vent Valve Pipe May Crack if Installed per Current Design

1R19 Post Maintenance Testing

Plant Procedures

QO-20, Attachment 1	Inservice Test Procedure - Low Pressure Safety Injection Pumps, Low Pressure Safety Injection Pump Test Evaluation; documented test data completed on March 17, 2003	Revision 12
T-303	Special Test - Emergency Diesel Generator 1-2 Overspeed Trip Setpoint Verification completed January 24, 2003	Revision 4
MO-7A-2	Emergency Diesel Generator 1-2 completed January 25, 2003	Revision 54
QI-39	Auxiliary Feedwater Actuation System Logic Test	Revision 0
ARP-7	Annunciator Number 29, "Service Water Pump Bay Lo-Level	
QO-19	Inservice Test Procedure - HPSI Pumps and ESS Check Valve Operability Test completed on December 28, 2002	Revision 22

Work Orders

24320249	Work Order - K-6B, EDG 1-2; Inspect and Adjust Fuel Rack
24320159	LS-0752D, Auxiliary Feedwater Pump P-8B Automatic Start
24320154	LS-0451C, Auxiliary Feedwater Pump P-8A Automatic Start
24320144	LA-0751A, RPS Channel A S/G 1 Lo Level Bistable Trip Unit
24320149	LA-0752B, RPS Channel B S/G 2 Lo Level Bistable Trip Unit
24320562	LIA-1338, Level Indicating Alarm for Service Water Bay Level
24214528	P-66A, High Pressure Safety Injection Pump

Miscellaneous Documents

Condition Reports Reviewed to Assess Characterization of Identified Problems

CAP034409	LPSI P-67B Vibration in the Alert Range of QO-20

CAP034038	LPSI P-67B Pump Motor Unable to be Aligned Within Specifications	
CAP033099	K-6B Failed to Stop after Overspeed Trip Test	

1R20 Refueling and Other Outage Activities

Plant Procedures

GOP-14	Shutdown Cooling Operations	Revisions 14, 15, 17
ONP-17	Loss of Shutdown Cooling	Revision 28
SOP-1	Primary Coolant System	
SOP-3	Safety Injection and Shutdown Cooling System	
GOP-14, Attachment 16	Shutdown Operation Equipment Sheets	
WI-PCS-M-06	Work Instruction - NSSS Walkdown	Revision 0

Miscellaneous Documents

	EOOS Review of REFOUT-3 Schedule for GOP 14 Compliance	February 26, 2003
	EOOS Review of REFOUT-3 Schedule for GOP 14 Compliance	March 16, 2003
	GOP-14 EOOS compliance review - 2 week look ahead for 3/16/03 - 3/29/03	March 18, 2003
	Plant computer data for primary coolant system cooldown rate and primary coolant system loop temperatures	March 15-17, 2003
	Palisades Daily Refout Status sheets and Operations Shift Briefing packages	
GL 88-17	Generic Letter, Loss of Decay Heat Removal	October 17, 1988
	Licensee 60 Day response to GL 88-17	January 3, 1989
	Licensee 90 Day response to GL 88-17	January 31, 1989

Condition Reports Reviewed to Assess Characterization of Identified Problems

CAP034061	Gray Duct Tape Found on Quench Tank Vent Line	
CAP034334	Boron Accumulation on HPSI Subcooling Control Valve CV-3071	

1R22 Surveillance Testing

Completed Technical Specification Surveillance Tests

RO-138B	Emergency Escape Air Lock Interlock Verification	Completed January 23, 2003
RT-85D	Inplace HEPA and Charcoal Filter Testing Control Room Ventilation “B” Train	Completed February 14 and 17, 2003
RT-85D	Inplace HEPA and Charcoal Filter Testing Control Room Ventilation “A” Train	Completed January 11 and February 4, 2003
RT-85C	Inplace HEPA and Charcoal Filter Testing - VF 66	Completed March 3, 4, 15 and 16th
RO-105	Full Flow Test for SIT Check Valves and PCS Loop Check Valves completed March 27, 2003; in addition to previously completed tests	Revision 7
RO-32-50	LLRT - Local Leak Rate Test Procedure for Escape Air Lock completed January 29, 2003	Revision 0
RE-131	Diesel Generator 1-1 Load Reject	Revision 2
T-302	Emergency Diesel Generator 1-1 Overspeed Trip Setpoint Verification	Revision 5
RI-3D	Pressurizer Pressure Channel D Calibration	Revision 0
RI-3C	Pressurizer Pressure Channel C Calibration	Revision 0
RI-5D	Steam Generator Pressure Channel D Calibration	Revision 0
QI-39	Auxiliary Feedwater Actuation System Logic Test completed February 18, 2003	Revision 0

Miscellaneous Documents

ASME N510-1989	Testing of Nuclear Air Treatment Systems
ANSI N510-1975	Testing of Nuclear Air-Cleaning Systems
PROC EQS-FLT03	In-Place Testing of Palisades Nuclear Plant HEPA Filter Systems
PROC EQS-FLT04	In-Place Testing of Palisades Nuclear Plant Charcoal Absorber Filter Systems

RG 1.52	NRC Regulatory Guide - Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System	Revision 2
WO 24320373	Work Order - MZ-50 Lights do Not Work	
WO 24211054	Work Order - MZ-50 - Inner Door Occasionally Sticks Closed	
Reg Guide 1.9, Section 2.2	Selection, Design, and Qualification of Diesel-Generator Units as Standby (Onsite) Electric Power Systems at Nuclear Power Plants, Test Descriptions	Revision 3, July 1993

Condition Reports Reviewed To Assess Problem Identification Characterization

CAP033742	Spent Fuel Pool Exhaust Fan V-8B Ductwork is Separated/Torn (VF-66 Inoperable)	‘
CAP033749	Procedure Administrative Weaknesses Found During Surveillance Peer Checking	
CAP033764	Laboratory Services Test Procedures Refer to Incorrect ASME Std for HVAC Test	
CAP033788	RT-85D Visual Inspections Not Performed in Accordance with TS 5.5.10 Sequence	
CAP033792	Damage Found to V-8A’s Discharge Duct	
CAP033829	Spent Fuel Pool Exhaust Fan V-8A was Found to have Low Air-Flow per TSST RT-85C	
CAP033895	Spent Fuel Pool Ventilation Testing not in Conformance with ASME Standard	
CAP033907	Incomplete Set of Visual Inspections Prescribed by RT-85D and RT-85C on HVAC	
CAP033908	RT-85D Improvement Needed to Check Required Flow Rates	
CAP033910	Incorrect Interpretation in RT-85C and RT-85D Basis Document	
CAP033911	Downstream HEPA Filter not tested in RT-85D for Control Room Ventilation Filter	
CAP033337	EDG 1-1 Right Fuel Oil Inlet Header Has Small Leak Near 1R Cylinder	

CAP033333	K-6A, Cylinder 9L is Making Different Noise Than Other Cylinders	
CAP032881	Setpoint Found Out of As-Found Tolerance During RI-3B	
CAP032759	PPC Indicated Reactor Power Unexpectedly Affected During RI-5B S/G Press Chan Calib	

Condition Reports Reviewed to Assess Condition Evaluation and Corrective Actions

CPAL0101259	Discontinuity Observed in PPC Plot for Safety Injection Tank T-82D Level Transmitter	

1R23 Temporary Plant Modifications

Plant Procedures

RI-4A	Technical Specification Surveillance Procedure - Steam Generator Level Channel A Calibration	Revision 0
RI-4B	Technical Specification Surveillance Procedure - Steam Generator Level Channel B Calibration	Revision 0
RI-4C	Technical Specification Surveillance Procedure - Steam Generator Level Channel C Calibration	Revision 0
RI-4D	Technical Specification Surveillance Procedure - Steam Generator Level Channel D Calibration	Revision 0

Temporary Modification Packages

TM-2003-003	Change the Setpoints and Associated Tolerances Listed in RI-4A, B, C & D for the Low Steam Generator Trip and AFAS Trip to the Values shown on the Attached Marked-Up Pages	

Condition Reports Reviewed To Assess Problem Identification Characterization

CAP032931	Level Transmitter Static Pressure Effect not Incorporated Correctly	

2OS1 Access Control to Radiologically Significant Areas

CAP 032361	Root Cause Evaluation for CAP032361 - Discrepant Dosimetry Results	January 27, 2003
CAP 032678	Locked High Radiation Area (LHRA) Locker Key Found Unattended in the RP Office	December 28, 2002

CAP 034220	Adverse Trend in the Number of Electronic Dosimeter Alarms during the Outage	March 19, 2003
CAP 034408	Worker Violated Locked High Radiation Barricade	March 23, 2003
CAP 034460	HEPA Unit Found Unplugged	March 24, 2003
HP 2.5	Palisades Nuclear Plant Health Physics Procedure - High Radiation Area Entry and Control	Revision 20
HP 2.18	Palisades Nuclear Plant Health Physics Procedure - Personnel Contamination	Revision 16
HP 2.20	Palisades Nuclear Plant Health Physics Procedure - Radiation Safety Area Posting	Revision 15
HP 2.29	Palisades Nuclear Plant Health Physics Procedure - Special Monitoring	Revision 9
HP 2.33	Palisades Nuclear Plant Health Physics Procedure - Dose Investigation and Assessment	Revision 12
HP 2.33	Palisades Nuclear Plant Health Physics Procedure - Dose Investigation and Assessment	Revision 12
HP 8.2	Palisades Nuclear Plant Health Physics Procedure - Whole Body Count Evaluation	Revision 11
HP 11.2	Palisades Nuclear Plant Health Physics Procedure - Control and Use of Portable Ventilation	Revision 3
HP 11.4	Palisades Nuclear Plant Health Physics Procedure - Evaluating Control of Airborne Radioactivity and Respiratory Protection	Revision 4
HP 2.19	Palisades Nuclear Plant Health Physics Procedure - Airborne Radioactivity Sampling	Revision 20
RWP P03-0007	Planning Inspections and Walkdowns	Revision 0
RWP P03-0501	Minor Activities in HRA	Revision 0
RWP P03-5004	Walkdowns and Inspections in Containment	Revision 3

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls

	ALARA Committee Meeting Minutes	March 27, 2003
	ALARA In-Progress Review for RWP P03-5102, Revision 1	March 23, 2003

	ALARA In-Progress Review for RWP P03-5150, Revision 0	March 24, 2003
AP 7.02	Palisades Nuclear Plant Administrative Procedure - ALARA Program	Revision 11
AP 7.04	Palisades Nuclear Plant Administrative Procedure - Radiation Dosimetry	Revision 19
AP 7.14	Palisades Nuclear Plant Administrative Procedure - Temporary Shielding Program	Revision 9
CAP 034283	Three Individuals Receive Unnecessary Dose as Work Had Already Been Completed	March 21, 2003
CAP 034284	Primary Coolant Activity Cleanup Did Not Achieve Preoutage Agreed Upon Value	March 21, 2003
CAP 034336	Increased Personnel Dose Experienced Due to Wrong Steam Generator Parts and Tool	March 22, 2003
HP 11.1	Palisades Nuclear Plant Health Physics Procedure - Processing Radiation Work Permits and ALARA Reviews	Revision 15
RWP P03-5100	Westinghouse Crane and Rigging Activities	Revision 1
RWP P03-5152	Installation and Removal of ROSA, ECT, and Tube Plugging	Revision 5
RWP P03-5155	Secondary Side Steam Generator Inspection	Revision 7
RWP/ALARA Plan P03-5102	Disassemble and Move Reactor Head to Stand	Revisions 0 - 1
RWP/ALARA Plan P03-5108	Reactor Head Reassembly and Refueling Close-Out Activities	Revision 0
RWP/ALARA Plan P03-5111	Removal/Install ICI Flanges and Associated Tasks	Revisions 0 - 1
RWP/ALARA Plan P03-5150	Install and Remove Nozzle Dams in E-50A/B	Revision 0
RWP/ALARA Plan P03-5306	Scaffold Work in Containment	Revisions 0 - 5
RWP/ALARA Plan P03-5503	NDE Bare Metal Inspection on N-50 Reactor Head	Revisions 0 - 1

3PP4 Physical Protection-Security Plan Change

	Palisades Nuclear Plant Security Plan	Revision 47, Dated January 10, 2003

4OA1 Performance Indicator Verification

	Licensee Performance Indicator Data for Safety System Functional Failures and Emergency Diesel Generator Unavailability	January through December 2002

4OA2 Identification and Resolution of Problems

Condition Reports Reviewed to Assess Condition Evaluation and Corrective Actions

CAP032269	Right Channel Monitor Line Caps Found Removed During RT-71P

Condition Reports Reviewed to Assess Characterization of Identified Problems

CAP033879	NRC Annual Assessment Letter Notes CAP Cross-Cutting Issue Still Open
CAP032785	Minor Error Noted in Condition Report Evaluation CPAL02-01930

4OA3 Event Follow-up

Plant Procedures

GOP-14	Shutdown Cooling Operations	Revision 17
ONP-2.1	Loss of AC Power	Revision 12
ONP-17	Loss of Shutdown Cooling	Revision 28
EI-1	Emergency Classifications and Actions	Revision 41

Miscellaneous Documents

EA-CND-03-01	Past Operability Determination for Two Removed Instrument Tube Caps on the Hydrogen Monitoring System	Revision 0
LER 02-003	Licensee Event Report, “Inoperable Containment Hydrogen Monitors”	February 19, 2003
LER 02-002	Licensee Event Report, “Automatic Reactor Trip and Safety System Actuation”	January 21, 2003
LER 03-001	Licensee Event Report, “Inoperable Steam Generator Low-Level Channels”	March 10, 2003
	Technical Specification 3.3.1, Reactor Protective System Instrumentation	Amendment 189

Condition Reports Reviewed To Assess Characterization of Identified Problems

CAP032931	Level Transmitter Static Pressure Effect not Incorporated Correctly	
CAP033022	Revision Required to January 17, 2003 Notice of Enforcement Discretion (NOED)	

Condition Reports Reviewed to Assess Root Cause Analysis and Corrective Actions

CAP033879	NRC Annual Assessment Letter Notes CAP Cross-Cutting Issue Still Open	
CAP032269	Root Cause Evaluation RCE 000314 for CAP032269 Right Channel Monitor Line Caps Found Removed During RT-71P	

4OA5 Other

Miscellaneous Documents

	Request for Enforcement Discretion - Steam Generator Low-Level Setpoints	January 17, 2003
	Request for Enforcement Discretion - Steam Generator Low-Level Setpoints, Revision 1	January 22, 2003
NOED 03-3-001	Notice of Enforcement Discretion For Nuclear Management Company LLC Regarding Palisades	January 22, 2003
	Technical Specification 3.0.3, Limiting Condition for Operation Applicability	Amendment 189

Condition Reports Reviewed To Assess Problem Identification Characterization

CAP032943	Delay in Initiating Action Request for Steam Generator Level Instrumentation	