April 19, 2001

EA-01-088

Mr. John Paul Cowan
Site Vice President
Palisades Nuclear Generating Plant
Consumers Energy Company
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR GENERATING PLANT - NRC INSPECTION REPORT 50-255/01-06(DRP)

Dear Mr. Cowan:

On March 31, 2001, the NRC completed an inspection at your Palisades Nuclear Generating Plant. The enclosed report documents the inspection findings which were discussed on April 9, 2001, with you and members of your staff.

The inspection examined activities conducted under your license as they relate to reactor 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, an Apparent Violation was identified and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is available on the NRC's website at www.nrc.gov/OE. We identified the apparent violation for the failure to provide the Commission complete and accurate information in your letter dated February 16, 2000, requesting enforcement discretion, and your letter dated February 18, 2000, requesting a change to your Technical Specifications. Specifically, information regarding the license basis for the underground (backup) steam supply line to your turbine driven auxiliary feedwater pump was not provided to the NRC in a complete and accurate manner. Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

An open predecisional enforcement conference to discuss this apparent violation has been scheduled for Tuesday, May 8, 2001, at 9:00 a.m. (CT), at the NRC Region III office in Lisle, Illinois. The decision to hold a predecisional enforcement conference does not mean that the NRC has determined that a violation has occurred or that enforcement action will be taken. This conference is being held to obtain information to assist the NRC in making an enforcement decision. This may include information to determine whether a violation occurred, information to determine the significance of a violation, information related to the identification of a violation, and information related to any corrective actions taken or planned. The conference will provide an opportunity for you to provide your perspective on these matters and any other information that you believe the NRC should take into consideration in making an enforcement decision.

You will be advised by separate correspondence of the results of our deliberations on this matter. No response regarding the apparent violation is required at this time.

In addition, the inspectors identified two issues of very low safety significance (Green), both of which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Palisades Nuclear Generating Plant.

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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Geoffrey E. Grant, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 50-255/01-06(DRP)

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REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	50-255/01-06(DRP)
Licensee:	Consumers Energy Company
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
Dates:	February 11 through March 31, 2001
Inspectors:	J. Lennartz, Senior Resident Inspector R. Krsek, Resident Inspector R. Walton, Reactor Engineer D. Chyu, Reactor Engineer
Approved by:	Anton Vegel, Chief Reactor Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000255-01-06 on 02/11 - 03/31/2001, Consumers Energy Company, Palisades Nuclear Generating Plant. Resident Inspectors Report. Fire Protection, Refueling and Outage Activities, and Temporary Modifications.

The baseline inspection was conducted by resident and region based inspectors. The inspection identified one non-color finding, which was determined to be an apparent violation. In addition, the inspection identified two Green findings, both of which were non-cited violations. The significance of most findings are indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html. Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- The inspectors identified an apparent violation regarding the licensee's failure to provide the NRC complete and accurate information in a letter dated February 16, 2000, requesting enforcement discretion, and in a letter dated February 18, 2000, requesting a Technical Specification change. Specifically, complete and accurate information concerning the license basis specified in the Post-Fire Safe Shutdown Analysis regarding the underground (backup) steam supply line to the turbine driven auxiliary feedwater pump was not provided.

  This finding potentially impacted the NRC's ability to perform its regulatory function. Since this finding cannot be processed through the Significance Determination Process, the apparent violation will be processed using the traditional enforcement process. (Section 1R05)

- Green. The inspectors identified a non-cited violation with three examples for the failure to construct seismically qualified scaffolds and storage racks near safety-related equipment.

  The finding was determined to be of very low safety significance (Green) by the significance determination process. Although the non-seismically qualified scaffold and storage racks could have credibly affected the operability, availability or function of components in mitigating systems during a seismic event, no seismic event occurred. Also, the as-found condition of the scaffolds did not impair the operation of the mitigating system components with the plant at power. (Section 1R20)

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for a temporary modification design change. Specifically, the underground (backup) steam supply to the Turbine Driven Auxiliary Feedwater Pump P-8B was isolated and removed from service when Temporary Modification 2000-06 was implemented. Consequently, the design change did not identify, evaluate and reconcile that Auxiliary Feedwater Pump P-8B was not available to remove decay heat using the underground (backup) steam supply line as credited in the Post-Fire Safe Shutdown Analysis for a fire in the Turbine Building.

The finding was determined to be of very low safety significance (Green) by the significance determination process. Isolating the underground (backup) steam supply line affected the operability, availability and function of Turbine Driven Auxiliary Feedwater Pump P-8B for a Turbine Building fire. However, the inspectors and an NRC Senior Reactor Analyst reviewed a quantitative Probabilistic Safety Assessment that was completed by licensee personnel and concluded that this issue was of very low safety significance. In addition, the licensee’s evaluation concluded that the operators would have been able to safely shut down the plant with the existing plant procedures using an additional auxiliary feedwater pump that would have been available for decay heat removal. (Section 1R23)

Report Details

Summary of Plant Status

The plant was operated at full power until March 30, 2001, when the plant was taken off-line for a scheduled refueling outage.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed partial walkdowns of the Emergency Diesel Generator 1-2, and High Pressure Safety Injection Pump P-66B to verify proper system lineup. The inspection verified that power was available, that accessible equipment was appropriately aligned, and that no discrepancies existed which would impact the system's function.

The inspection incorporated discussions and system walkdowns with operations and engineering personnel. The inspectors also reviewed applicable portions of the Technical Specifications and reviewed the following documents:

- Standard Operating Procedure 22, "Emergency Diesel Generators," Revision 30;
- Standard Operating Procedure 22, Attachment 8, "Checklist 22.1, Diesel Generators System Checklist," Revision 30;
- Final Safety Analysis Report Section 8.4, "Emergency Power Sources";
- System Operating Procedure 3, Checklist 3.1, "Engineered Safeguards System Checklist (Shutdown Cooling in Service)," Revision 45;
- System Operating Procedure 3, Checklist 3.2, "Engineered Safeguards System Checklist (To Secure Shutdown Cooling)," Revision 45;
- System Operating Procedure 3, Checklist 3.9, "Engineered Safeguards Administrative Control Verification," Revision 45;
- Monthly Operating Procedure 29, "Engineered Safety System Alignment," Revision 30;
- Piping and Instrument Diagram M204, Sheets 1, 1A, and 1B Revisions 70, 25, and 30 respectively;
- Engineering Analysis EA-PLTB-01, "Evaluation of the Containment Sump Outlet Gate Valves CV-3029 and CV-3030 for Susceptibility to Pressure Locking and Thermal Binding," Revision 1; and
- Completed Technical Specification Surveillance Test Procedure Quarterly Operational 2, "Recirculation Actuation System," Revision 29, dated November 20, 1999.

b. Issues and Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Turbine Building (Fire Area 23) Post-Fire Safe Shutdown Analysis (PSSA)

a. Inspection Scope

The inspectors reviewed applicable portions of the following documents:

- Fire Protection Program Report;
- Palisades Plant Post-Fire Safe Shutdown Analysis, dated April 16, 1996;
- Engineering Analysis EA-PSSA-00-001, "Palisades Plant Post-Fire Safe Shutdown Analysis," Revision 0, dated March 3, 2000;
- Design Basis Document 1.03, "Auxiliary Feedwater System," Revision 5;
- Palisades Plant Fire Hazards Analysis Report, Revision 4, dated July 2, 1998;
- Engineering Analysis EA-APR-95-007, "10 CFR 50, Appendix R, Fire Safe Shutdown Analysis," Revision 1;
- Consumers Energy Correspondence to the U. S. NRC, dated February 16, 2000, "Docket 50-255 - License DPR-20 - Palisades Plant - Notice of Enforcement Discretion - Auxiliary Feedwater";
- Final Safety Analysis Report, Section 9.6, "Fire Protection," Revision 22;
- Consumers Energy Correspondence to the U. S. NRC, dated February 18, 2000, "Docket 50-255 - License DPR-20 - Palisades Plant - Technical Specifications Change Request - Auxiliary Feedwater"; and
- U. S. NRC Correspondence to Consumers Energy, dated March 14, 2000, "Palisades Plant - Issuance of Amendment Re: Backup Steam Supply for Turbine-Driven Auxiliary Feedwater Pump P-8B (TAC No. MA8247)," and the associated Safety Evaluation Report.

In addition, the inspectors reviewed the following condition reports to verify that identified problems regarding the Appendix R Analyses for Fire Area 23 were entered into the corrective action program with the appropriate characterization and significance:

- CPAL0100259, "Removal of Auxiliary Feedwater Control Valve CV-0522A Supply to Auxiliary Feedwater Pump P-8B was not Adequately Reviewed Against Appendix R Analyses";
- CPAL0100797, "Appendix R Program Deficiencies"; and
- CPAL9500276, "Erroneous Information Submitted to the NRC."

b. Issues and Findings

The inspectors identified an apparent violation of 10 CFR 50.9 regarding the licensee's failure to provide the NRC complete and accurate information in all material respects. Specifically, the licensee failed to provide the Commission complete and accurate information in a letter dated February 16, 2000, requesting enforcement discretion, and in a letter dated February 18, 2000, requesting a Technical Specification change.

The underground (backup) steam supply line through Control Valve CV-0522A to Turbine Driven Auxiliary Feedwater Pump P-8B failed in February 2000 (reference NRC Inspection Report 50-255/00-01(DRP) Sections E2.2 and M2.1). Consequently, licensee personnel submitted a Request for Enforcement Discretion and a Technical Specification change request to the NRC dated February 16 and 18, 2000, respectively, to remove the line from service via a temporary modification. (See Section 1R23 of this Report for additional information regarding the temporary modification and associated technical issues.)

The licensee's Request for Enforcement Discretion, Section 3, "Safety Basis Supporting the Request," and the Technical Specification Change Request, Section D, "Design Basis Event Considerations," stated the following, in part:

> "For the Post-Fire Safe Shutdown Analysis [Auxiliary Feedwater Pump] P-8C is the preferred source of feedwater for this event. Only one case exists where use of the underground steam supply through [Control Valve] CV-0522A was considered as available to help the plant in achieving cold shutdown. This case is associated with a fire in the Southwest Cable Penetration Room, which will affect controls for P-8A, and CV-0522B. The analysis of record notes that either P-8C or CV-0522A would be available for supplying auxiliary feedwater to the steam generators but only requires one source of auxiliary feedwater. P-8C is the preferred source of feedwater for this event. In addition, the manual handwheel on [Control Valve] CV-0522B is available for use in providing a steam supply to [Auxiliary Feedwater Pump] P-8B during this event."

Based on the information obtained from the licensee's correspondence, the Office of Nuclear Reactor Regulation concluded in the Safety Evaluation Report related to the Technical Specification Change Request, Amendment 190, (TAC No. MA8247) the following, in part:

> "The licensee indicated that the only instance for which the backup steam supply for P-8B has been credited is for a postulated fire in the southwest cable penetration room, which could affect CV-0522B and the controls for P-8A. The licensee's analysis of record relies upon either P-8C or CV-0522A to be available for satisfying the AFW [Auxiliary Feedwater] function. With the elimination of the underground (backup) steam supply line for P-8B (which also eliminates CV-0522A), the licensee would now rely upon use of the manual handwheel for CV-0522B for providing steam to P-8B during this event. However, P-8C is unaffected and would continue to be the preferred source of AFW [Auxiliary Feedwater]. The NRC staff considers this to be an acceptable approach for assuring availability of AFW [Auxiliary Feedwater] during a fire in the southwest cable penetration room."

The NRC granted the Enforcement Discretion and approved the Technical Specification change request based on the belief that only one case existed where use of the underground (backup) steam supply line was considered to help the plant in achieving cold shutdown in the Post Fire Safe Shutdown Analysis.

However, in January of 2001, the inspectors identified that Fire Protection Program Report, Section IV, credited the underground (backup) steam supply line and Control Valve CV-0522A for a fire in the turbine building. Specifically, the Post-Fire Safe Shutdown Analysis (PSSA) for Fire Area 23, "Turbine Building," dated April 16, 1996, and

Engineering Analysis EA-PSSA-00-001, “Palisades Plant Post-Fire Safe Shutdown Analysis,” Revision 0, specified an initial reliance on Control Valve CV-0522A and Auxiliary Feedwater Pump P-8B following a turbine building fire. Consequently, the turbine building fire area was an area in addition to the SW penetration room in the licensee’s fire protection program where control valve CV-0522A and the underground (backup) steam supply line were considered available in helping the plant achieve safe shutdown.

Facility Operating License No. DPR-20, License Condition 2.C.(3), states, in part, that Consumers Energy Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility.

Final Safety Analysis Report, Section 9.6, “Fire Protection,” specifically, Section 9.6.1.1, “Other FSAR Sections Related to Fire Protection,” stated the following, in part:

> “The fire area safe shutdown analysis, including details of components lost, and components relied upon for safe shutdown following a fire in any fire area of the plant, is presented in the Post-Fire Safe Shutdown Analysis (PSSA), which is included in the Fire Protection Program Report described below.”

The Fire Protection Program Report, Section IV, “Post-Fire Safe Shutdown Analysis (PSSA),” dated April 16, 1996, Section 3.23, “Fire Area 23 - Turbine Building,” under decay heat removal -hot standby stated the following, in part:

> “AFW [Auxiliary Feedwater] pumps P-8B and P-8C both remain available for a fire in this area. Initially, P-8B is credited via local manual operation of the underground steam line supply valve CV-0522A using its handwheel, located in Room 238....Pump P-8C remains available from the control room as a second source of feedwater....AFW [Auxiliary Feedwater] Pump P-8C cannot be relied upon as the primary AFW [Auxiliary Feedwater] source because the steam generator safety relief valve setpoint is higher than the shutoff head of this pump. Therefore this pump will not be able to provide feedwater to the steam generators until the secondary side pressure has been reduced below approximately 900 psig.”

In addition, the inspectors noted that Engineering Analysis EA-APR-95-007, “10 CFR 50 Appendix R Fire Safe Shutdown Analysis,” Revision 1, dated January 28, 1999, Appendix D, “Fire Safe Shutdown Compliance Assessments,” for Fire Area 23 also considered Control Valve CV-0522A as available to help the plant in achieving cold shutdown for a fire in the Turbine Building.

Therefore, the licensee’s submittals in February 2000, for the Notice of Enforcement Discretion and Technical Specification Change Request failed to identify that Control Valve CV-0522A and the underground (backup) steam supply was credited in the fire protection program for a fire in the Turbine Building. Consequently, the submittals were inaccurate and incomplete.

This finding potentially impacted the NRC's ability to perform its regulatory function. Since this finding cannot be processed through the Significance Determination Process, the apparent violation will be processed using the traditional enforcement process.

10 CFR 50.9 (a), "Completeness and Accuracy of Information," requires, in part, that information provided to the Commission by the licensee shall be complete and accurate in all material respects. However, the licensee's letters submitted for a Notice of Enforcement Discretion dated February 16, 2000, and a Technical Specification Change Request dated February 18, 2000, stated the following, in part: "Only one case exists where use of the underground steam supply through [Control Valve] CV-0522A was considered as available [in the Post-Fire Safe Shutdown Analysis] to help the plant in achieving cold shutdown. This case is a fire in the Southwest Cable Penetration Room...."

In January 2001, the inspectors identified that Section IV of the Fire Protection Program Report, specifically, "Post-Fire Safe Shutdown Analysis," Section 3.23, "Fire Area 23 - Turbine Building," initially credited the underground (backup) steam supply through Control Valve CV-0522A to Auxiliary Feedwater Pump P-8B as the source of feedwater available for decay heat removal-hot standby for a fire in the Turbine Building. Consequently, the licensee failed to provide the Commission complete and accurate information regarding the underground (backup) steam supply line in that the underground (backup) steam supply line was considered as available for a postulated fire in the turbine building in addition to the SW cable penetration room. Also, the information was material, in that, the NRC relied on the information to grant the requested Notice of Enforcement Discretion and the exigent Technical Specification Change in February 2000, to remove the requirements for the underground (backup) steam supply from the Technical Specifications. The failure to provide complete and accurate information is an Apparent Violation of 10 CFR 50.9. This issue was captured in the licensee's corrective action program as Condition Report CPAL0100259.
(EEI 50-255/01-06-01)

The licensee's evaluation of this issue, which was completed in April 2001, concluded that plant equipment was available and procedures were in place to safely shutdown the plant in the event of a fire in the turbine building. Specifically, Auxiliary Feedwater Pump P-8C would have performed the required decay heat removal functions and in place off-normal and emergency operating procedures provided adequate guidance to the operators to safely shut down the plant.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed requalification testing during simulator scenario evaluations for licensed operators, to assess the licensed operator's ability to mitigate an Excess Steam Demand Event and a Steam Generator Tube Rupture with multiple equipment malfunctions. The following attributes of operator performance were assessed:

- crew communications and event diagnosis;
- ability to take timely and appropriate mitigation actions;

- correct use of emergency operating procedures;
- ability to operate the controls in a timely and appropriate manner;
- effective command and control by the senior reactor operator; and
- emergency plan implementation by the shift supervisor.

Also, the inspectors observed the post-scenario critique to assess the licensee evaluator's ability to identify operator performance deficiencies. The inspectors reviewed the following documents:

- Palisades Nuclear Training Procedure 7, Attachment 5, "Simulator Performance Evaluations," Revision 6;
- Emergency Operation Procedure 1.0, "Standard Post Trip Actions," Revision 10;
- Emergency Operating Procedure 5.0, "Steam Generator Tube Rupture Recovery," Revision 12; and
- Emergency Operating Procedure 6.0, "Excess Steam Demand Event," Revision 12.

In addition, the inspectors reviewed the following condition report to verify that the identified problem was initially characterized and evaluated appropriately with respect to licensed operator requalifications:

- CPAL0100693, "Qualification Requirements Not Clearly Defined for Simulator Evaluation Team."

b. Issues and Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors evaluated the licensee's effectiveness regarding Maintenance Rule, 10 CFR 50.65, implementation for components in systems ranked in the high safety significant category. The inspectors reviewed recent maintenance rule evaluations to assess the categorization of specific issues related to the following systems:

- Low Pressure Safety Injection System;
- Normal Shutdown and Design Basis Accident Sequencer; and
- Main Steam and Atmospheric Steam Dump Valve System.

The inspectors also reviewed and evaluated the applicable performance criteria, risk rankings and scoping criteria for appropriateness. In addition, the inspectors interviewed the licensee's maintenance rule coordinator and evaluated the monitoring and trending of performance data with the responsible system engineer when applicable. The inspectors reviewed the following maintenance rule program and supporting documentation:

- NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2;

- Engineering Manual Procedure EM-25, “Maintenance Rule Program,” Revision 3;
- Design Basis Document 2.01, “Low Pressure Safety Injection System,” Revision 6;
- Engineering Procedure EGAD-EP-10, Attachment 2, “Maintenance Rule Scoping Document - Low Pressure Safety Injection System,” Revision 2;
- Final Safety Analysis Report Section 6.1, “Safety Injection System”;
- System Health Assessment Report - 3rd/ 4th Quarter 2000, Engineered Safeguards Systems;
- Emergency Operating Procedure 1.0, “Standard Post-Trip Actions Basis,” Revision 8;
- Emergency Operating Procedure 5.0, “Steam Generator Tube Rupture Basis,” Revision 9; and
- Emergency Operating Procedure 6.0, “Excess Steam Demand Event Basis,” Revision 9.

In addition, the inspectors reviewed the following condition reports to verify that identified problems were appropriately characterized and evaluated with respect to the maintenance rule:

<u>Low Pressure Safety Injection System</u>

- CPAL0003197, “Auxiliary Contact Configuration Discrepancy (Low Pressure Safety Injection Pump P-67A)”;
- CPAL0100182, “Low Pressure Safety Injection Pump As-Found Alignment Out of Specification”; and
- CPAL0100189, “Low Pressure Safety Injection Pump P-67A Increased Vibrations Observed During Quarterly Operations Technical Specification Surveillance Test QO-20A.”

<u>Design Basis Accident Sequencer</u>

- CPAL0001352, “Updated Motor Acceleration Times Not included In DBA Sequencer Timing Study.”

<u>Main Steam and Atmospheric Steam Dump Valve System</u>

- CPAL0001051, “A Steam Generator Atmospheric Dump Valves Steaming Harder than B Steam Generator Atmospheric Dump Valves Following Plant Trip”;
- CPAL0002063, “Turbine Bypass Valve CV-0511 did not Fully Close During Plant Cooldown”;
- CPAL0002071, “CV-0501 Does Not Close Completely During QO-37, Main Steam Isolation and Bypass Valve Testing”; and
- CPAL0002558, “Lowering Trend on “A” Steam Generator Steam Flow Indication (FT-0702).”

b. <u>Issues and Findings</u>

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed equipment out-of-service risk assessments for planned and emergent maintenance activities and reviewed Administrative Procedure 4.02, “Control of Equipment,” Revision 17. The inspectors discussed the risk evaluations and plant configuration control for the maintenance activities with operations, maintenance and work control center personnel to evaluate whether the necessary steps were taken to control the work activities. The inspectors reviewed the following documents for planned maintenance activities:

- Operator’s Risk Reports and Shift Supervisor log entries for February 12 through February 16, 2001, for work performed on the Reactor Protection System “C” and “D” Channels Thermal Margin Monitors Current Isolators;
- Operator’s Risk Reports and Shift Supervisor log entries for March 5 through March 9, 2001, for work performed on Emergency Diesel Generator 1-1 and Technical Specification Surveillance Tests performed on the Reactor Protection System Matrix Relays and the Reactor Protection System Trip Units; and
- Operator’s Risk Reports and Shift Supervisor log entries for March 13 through March 16, 2001, for work performed on the Service Water intake structure.

In addition, the inspectors reviewed the following condition reports to verify that identified problems were appropriately characterized and evaluated with respect to maintenance risk assessments and emergent work evaluations:

- CPAL0100708, “Out of Tolerance Data During the Performance of QI-2A (Thermal Margin/ Low Pressure Trip Units Surveillance Procedure)”;
- CPAL0100716, “Service Water Pump P-7B Basket Strainer High Differential Pressure”; and
- CPAL0100727, “Boric Acid Gravity Feed Path Declared Inoperable.”

b. Issues and Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspectors assessed operator performance in response to a lowering level in the service water bay on February 17, 2001, due to frazzle ice on the traveling screens. The inspectors verified that the operator’s responded appropriately in accordance with Off Normal Operating Procedure 6.1, “Loss of Service Water,” and various annunciator response procedures. Also, the inspectors verified compliance with Technical Specifications. The inspectors reviewed the following documents:

- Off Normal Procedure 6.1, “Loss of Service Water,” Revision 10

- Standard Operating Procedure 11, Attachment 2, “Backflushing Condensate; Pump P-2A (P-2B) Thrust Bearing Cooling Coil,” Revision 11;
- Standard Operating Procedure 15, “Service Water System,” Revision 18;
- Standard Operating Procedure 14, “Circulating Water and Chlorination Systems,” Section 7.13.1, “Warm Water Recirculation For Screen Icing,” Revision 32;
- Annunciator Response Procedures EK0156, “Condensate Pump P2A Hi Temp or O.L.,” Revision 50; EK1144, “Serv Water Pump P7C Basket Str Hi DP,” EK1129, “Service Water Pump Bay LO Level,” and EK1124, “Traveling Screen HI DP (Reflash),” Revision 61;
- Work Order 24110520, “Traveling Water Screen F-4C Broken Shear Pin Due To Icing”;
- Shift Supervisor Log Book entries for February 17 through February 18, 2001; and
- Equipment and System Operational Guidance Recommendation, Item 70A, “Frazzle Ice.”

The inspectors also reviewed the following documents to verify that corrective actions taken to address industry experience information and past condition reports related to frazzle ice issues were reasonable:

- CPAL0100025, “Traveling Screen High Differential Pressure Alarms”;
- CPAL970210, “Service Water Pump Bay Level Decrease Without Alarm”;
- Significant Event Report 96-008, “Extreme Environmental Conditions Result In a Reactor Scram With Failure of Five Control Rods To Fully Insert”;
- NRC Information Notice 98-02, “Nuclear Power Plant Cold Weather Problems and Protective Measures”; and
- NRC Information Notice 96-036, “Degradation Of Cooling Water Systems Due To Icing.”

Further, the inspectors reviewed the following condition report to verify that this issue was entered into the corrective action program with the appropriate characterization and significance:

- CPAL0100545, “Intake Bay Ice Results In Traveling Screen F-4C Failure and Entering of ONP-6.1, “Loss of Service Water.”

b. Issues and Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability assessment as documented in the following condition report regarding Auxiliary Feedwater Pump P-8B:

- CPAL0100259, "Removal of Auxiliary Feedwater Control Valve CV-0522A Supply to Auxiliary Feedwater Pump P-8B was not Adequately Reviewed Against Appendix R Analyses."

The inspectors interviewed the applicable engineers, and reviewed the following supporting documents to assess the adequacy of the operability assessment:

- Action Item Request A-NL-82-100, "Use of the Air Hogging Ejector for Decay Heat Removal";
- Design Basis Document 1.03, "Auxiliary Feedwater System," Revision 5; and
- Final Safety Analysis Report Section 9.7, "Auxiliary Feedwater Systems."

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Further, the inspectors reviewed the following condition report to verify that identified problems associated with the operability evaluation were appropriately characterized and entered into the licensee's corrective action program:

- CPAL0100531, "10 CFR 50 Appendix R Analysis Basis Does Not Adequately Document a Turbine Building Fire Safe Shutdown Path"; and
- CPAL0100592, "Appendix R Analysis Does Not Adequately Document an Auxiliary Feedwater Suction Source For The Turbine Building Fire."

b. Issues and Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors evaluated the cumulative effect of identified operator workarounds to assess any potential effects on the functionality of mitigating systems. Also, the inspectors walked down a random sample of identified actions in abnormal and emergency operating procedures to assess whether the operators' ability to implement the procedures in a timely manner was affected. The inspectors reviewed the following identified Operator Workarounds:

- 01-01OWA, "Component Cooling Water Containment Isolation Valves";
- 01-02OWA, "Safety Injection Refueling Water Tank Recirculation Valves";
- 01-06OWA, "High Pressure Safety Injection Pumps";
- 01-11OWA, "High Pressure Safety Injection/Low Pressure Safety Injection Motor Operator Valves"; and
- 01-13OWA, "Engineered Safeguards System Sump Pumps."

In addition, the inspectors reviewed the following documents:

- Administrative Procedure 4.12, “Operator Work Around Program,” Revision 0;
- Palisades Nuclear Plant Action Plan 2210-009, “Operator Work Around Program,” Revision 0;
- Palisades Operator Challenges and Operator Workarounds Lists;
- Operations Equipment Status List and Control Room Deficiencies List;
- Compensatory Actions for Degraded Equipment List;
- Completed Administrative Procedure 4.0, Attachment 2, “Senior Reactor Operator Shift Turnover Sheets”;
- Probabilistic Safety Assessment Report 1, “Post Accident Operator Actions with Importance Measures Table”;
- Emergency Operating Procedure 4.0, “Loss Of Coolant Accident Recovery,” Revision 12, and associated basis document;
- Emergency Operating Procedure 4.0 Attachment 1, “Loss Of Coolant Accident Recovery Safety Function Status Check Sheet,” Revision 12, and associated basis document;
- Emergency Operating Procedure Supplement 6, “Checksheet for Containment Isolation and Component Cooling Water Restoration,” Revision 0; and
- Emergency Operating Procedure Supplement 4, “High Pressure Safety Injection and Low Pressure Safety Injection Flow Curves,” Revision 5.

b. Issues and Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed a modification to replace the shaft seal on Low Pressure Safety Injection Pump P-67A. The inspectors verified that the modification did not degrade the design bases, the licensing bases, and performance capability of this risk significant safety system component. The inspectors also interviewed applicable engineering personnel, and toured the East Engineering Safeguards room with the system engineer.

The inspectors reviewed the following documents:

- Engineering Action Request 99-0238, “Identify and Evaluate the Use of an Improved Mechanical Shaft Seal”;
- 10CFR50.59 Safety Review, SDR-00-1146, “Replacement of Mechanical Seal for P67-A and P-67B”;
- Final Safety Analysis Report Change Request 2023; and
- System Operating Procedure 3, “Safety Injection and Shutdown Cooling System,” Revision 45.

In addition, the inspectors reviewed the following piping and instrument diagrams:

- M1-GB, Sheet 804, Chesterton Seal Installation;
- M1-GB, Sheet 163-2, Low Pressure Safety Injection Pump Seal Piping;
- M1-GB, Sheet 164-3, Low Pressure Safety Injection Pump Manifold Piping;
- M1-GB, Sheet 803, Heliflow Heat Exchanger - Operating and Maintenance Instructions; and
- M1-GB, Sheet 163- Rev. B., Low Pressure Safety Injection Pump Seal Piping.

b. Issues and Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors observed portions of post maintenance testing and reviewed documented testing activities following scheduled maintenance to verify that the tests were performed as written, and that applicable testing prerequisites were met prior to the start of the test. The inspectors reviewed post maintenance test activities for the following components:

- Emergency Diesel Generator 1-1;
- Containment Spray Pump P-54C;
- High Pressure Safety Injection Pump P-66A; and
- Auxiliary Feedwater Pump P-8C.

The inspectors reviewed post maintenance testing criteria specified in the following preventive Work Orders regarding Emergency Diesel Generator 1-1:

- 24014994, “K-6A Diesel Generator Check and Collector Ring Preventive Maintenance”; and
- 24110396, “K-6A Lube Oil Leak in Cold Lube Oil Return.”

The inspectors reviewed post maintenance testing criteria specified in the following Work Orders regarding Containment Spray Pump P-54C:

- 24014051, “Preventive Maintenance on Breaker 152-114 (Containment Spray Pump P-54C Motor)”; and
- 24014052, “Thermalscan Equipment Pump P-54C and Breaker 152-114.”

The inspectors reviewed post maintenance testing criteria specified in the following Work Orders regarding High Pressure Safety Injection Pump P-66A:

- 24014226, “Preventive Maintenance on Breaker 152-207 (High Pressure Safety Injection Pump P-66A Motor)”; and
- 24014227, “Thermalscan Equipment Pump P-66A and Breaker 152-207.”

The inspectors reviewed post maintenance testing criteria specified in the following preventative and corrective Work Orders regarding Auxiliary Feedwater Pumps P-8A and P-8C:

- 24013809, "Thermalscan Equipment Pump P-8A and Breaker 152-104"; and
- 24013525, "Preventive Maintenance on Breaker 152-209 (Auxiliary Feedwater Pump P-8C)."

In addition, the inspectors reviewed the completed test procedures to verify that the tests were adequate for the scope of work performed and to verify that acceptance criteria were clear. The inspectors reviewed documented test data to verify that the data was complete and that the equipment met the procedure acceptance criteria in the following Technical Specification Surveillance Procedures:

- MO-7A-1, "In-Service Test Procedure - Emergency Diesel Generator 1-1 (K-6A)," Revision 54 and associated bases document;
- QO-16C, "In-Service Test Procedure - Containment Spray Pumps," Revision 17 and associated basis document;
- QO-19A, "In-Service Test Procedure - High Pressure Safety Injection Pumps and Engineered Safeguards System Check Valve Operability Test," Revision 21 and associated basis document; and
- QO-21, "In-Service Test Procedure - Auxiliary Feedwater Pumps," Revision 20 and associated basis document.

Further, the inspectors reviewed the following condition reports to verify that identified problems regarding post maintenance testing activities were appropriately characterized and entered into the licensee's corrective action program:

- CPAL0100872, "Preventive Maintenance on Breaker 152-207 (High Pressure Safety Injection Pump P-66A) Identifies Component Discrepancies";
- CPAL0100875, "High Pressure Safety Injection Pump P-66A Second Start";
- CPAL0100971, "Revision to QO-21 (Inservice Test Procedure - Auxiliary Feedwater Pumps) Leads to Confusion Regarding Acceptance Criteria"; and
- CPAL0100974, "Photo-Optic Sensor used for QO-21B, Auxiliary Feedwater Pump P-8B Testing."

b. Issues and Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

.1 Pre-Outage Scaffolding Construction

a. Inspection Scope

The inspectors conducted plant walkdowns prior to the 2001 refueling outage to verify scaffolding that was built to support the outage did not impact safety related equipment and did not impact the operator's ability to perform in-plant tasks. Also, the inspectors reviewed the following plant procedures regarding scaffolding and storage rack construction:

- Permanent Maintenance Procedure MSM-M-43, “Scaffolding,” Revision 4; and
- Administrative Procedure 1.01, Attachment 2, “Guidelines for Placement of Unrestrained Items in Areas Containing Operable Safety-Related Equipment.”

Further, the inspectors reviewed the following condition reports to verify that identified problems regarding scaffolding activities were appropriately characterized and entered into the licensee’s corrective action program:

- CPAL0100652, “Scaffold Adjacent to Control Valve CV-522B Was Not Requested For Safety Applications”;
- CPAL0100695, “Scaffold Materials in Scaffold Racks Found Unsecured”;
- CPAL0100747, “Previously Identified Deficient Green Tag Scaffolds Not Seismically Built”;
- CPAL0100721, “Green Tagged Scaffolds with Procedural Discrepancies”; and
- CPAL0100866, “Scaffold on Low Pressure Turbine "B" Shell Violates Procedure.”

b. Issues and Findings

The inspectors identified a green finding and associated non-cited violation regarding the failure to construct seismically qualified scaffolding and storage racks near operable safety-related equipment.

The inspectors identified, prior to the start of the Refueling Outage, the following examples in which scaffolds and storage racks constructed near operable safety-related equipment were not seismically qualified as required by plant procedures:

- On February 27, 2001, the inspectors identified that scaffolding built adjacent to the Auxiliary Feedwater Pump P-8B steam supply piping and associated Control Valve CV-522B was not rigidly secured and consequently not seismically qualified;

- On March 2, 2001, the inspectors identified that a scaffold storage rack in the East Engineered Safeguards Room was contacting the safety-related Low Pressure Safety Injection Pump P-67A suction piping; and

- On March 2, 2001, the inspectors identified that scaffold materials on the storage racks near safety-related equipment in the East Engineered Safeguards and Component Cooling Water Pump Rooms were not rigidly secured. Consequently, the material stored on the racks could potentially impact and effect the operable safety-related equipment during a seismic event.

Licensee personnel subsequently verified that the as-built scaffold and storage rack, and materials stored on the storage racks were not seismically qualified as required by Plant Procedure MSM-M-43, “Scaffolding,” and Administrative Procedure 1.0.1, Attachment 2, “Guidelines for Placement of Unrestrained Items in Areas Containing Operable

Safety-Related Equipment." The licensee immediately modified the scaffolding, storage rack, and materials stored on the storage racks to comply with the seismic requirements. In addition, the licensee initiated Condition Reports CPAL0100652 and CPAL0100695. The inspectors also noted that the requirements for storage racks were not succinctly addressed in Administrative Procedure 1.01 which may have contributed to the problem.

Subsequently, licensee maintenance and operations personnel initiated an extent of condition review of all scaffold and storage racks which had been built. Licensee personnel identified two additional scaffolds constructed near operable safety-related equipment which did not satisfy the seismic requirements. Licensee personnel immediately corrected the problem and initiated Condition Report CPAL0100747.

The inspectors concluded that these issues had a credible impact on safety, in that, during a seismic event, the as found condition of the scaffold and storage racks could have credibly affected the operability, availability or function of components in mitigating systems.

The finding was determined to be of very low safety significance (Green) by the significance determination process. Although the non-seismically qualified scaffold and storage racks could have credibly affected components in mitigating systems during a seismic event, no seismic event had occurred. Also, the as-found condition of the scaffolds and storage racks did not impair operation of the mitigating system components with the plant at power.

Technical Specification 5.4.1, "Procedures," states, in part, that written procedures shall be implemented covering the applicable procedures recommended in Regulatory Guide 1.33. Regulatory Guide 1.33, Section 1, "Administrative Procedures," states, in part, that procedures to address equipment control are required. Permanent Maintenance Procedure MSM-M-43 and Administrative Procedure 1.01 required, in part, that scaffolds, storage racks and stored materials shall be built and maintained to seismic requirements to prevent damage to operable safety-related equipment in the event of an earthquake.

However, the inspectors identified three examples where the as-found condition of scaffolds, storage racks and stored materials did not satisfy the seismic requirements contained in the procedures. In accordance, with Section VI.A.1 of the NRC Enforcement Policy, these examples of procedure violations are being treated as a Non-Cited Violation. These issues were entered into the licensee's corrective action program as Condition Reports CPAL0100652 and CPAL0100695. (NCV 50-255/01-06-02)

.2 Review of Refueling Outage 2001 Plan

a. Inspection Scope

The inspectors reviewed the results from the Probabilistic Safety Assessment Group's review of the 2001 refueling outage schedule that was conducted to verify compliance with General Operating Procedure 14, "Shutdown Cooling Operations." Also, the inspectors reviewed the licensee's outage risk control plan to verify that risk was appropriately considered. In addition, the inspectors verified that procedures existed to respond to a loss of Shutdown Cooling. The inspectors reviewed the following documents:

- EOOS (Equipment Out Of Service) Review of REFOUT-1 Schedule For General Operating Procedure 14 Compliance;
- General Operating Procedure 14, "Shutdown Cooling Operations," Revision 13; and
- Off Normal Procedure 17, "Loss of Shutdown Cooling," Revision 26.

In addition, the inspectors reviewed the licensee's responses to Generic Letter 88-17, "Loss of Decay Heat Removal," to verify that any commitments made were in place and adequate as documented in the following:

- Palisades Plant 90 Day Response, dated January 31, 1989;
- Palisades Plant 60 Day Response, dated January 3, 1989;
- Palisades Plant Response To Programmed Enhancements Update, dated February 27, 1991; and
- Palisades Plant Response Change In Commitment, dated August 27, 1992.

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed portions of surveillance testing activities conducted on the following risk-significant plant equipment to verify that testing was conducted in accordance with prescribed procedures:

- Low Pressure Safety Injection Pump P-67B and Check Valves;
- High Pressure Safety Injection Pump P-66A and Check Valves; and
- Shutdown Cooling Control Valves.

The inspectors also reviewed documented test data for the following Technical Specification Surveillance Test procedures and the associated basis documents to verify that testing acceptance criteria were satisfied:

- QO-20B, "In-Service Test Procedure - Low Pressure Safety Injection Pumps," Revision 12;
- QO-19B, "In-Service Test Procedure - High Pressure Safety Injection Pumps," Revision 21; and
- QO-42, "Section XI Testing of Shutdown Cooling Control Valves," Revision 5.

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

Further, the inspectors reviewed the following condition report to verify that identified problems regarding surveillance testing activities were being entered into the corrective action program with the appropriate characterization and significance:

CPAL0100840, “CV-3055 Shutdown Cooling Inlet Failed Acceptance Criteria of Inservice Test QO-42.”

b. Issues and Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification package and associated 10 CFR 50.59 evaluation:

- Temporary Modification TM 2000-006, “Temporary Isolation of Steam Generator E-50B Steam Supply to Auxiliary Feedwater Pump P-8B,” dated February 11, 2000.

The inspectors also discussed the modification implementation with engineering personnel, and walked down accessible portions of the installation. The inspectors reviewed the following documents:

- Palisades Plant Post-Fire Safe Shutdown Analysis, dated April 16, 1996;
- Engineering Analysis EA-PSSA-00-001, “Palisades Plant Post-Fire Safe Shutdown Analysis,” Revision 0, dated March 3, 2000;
  Design Basis Document 1.03, “Auxiliary Feedwater System,” Revision 5;
- Palisades Nuclear Plant 10 CFR 50.59 Safety Review Packages and supporting documentation for Log Nos. SDR-00-170, SDR-00-199 and SDR-00-0260, “Isolation of the Auxiliary Feedwater Pump Turbine Steam Supply from Pressure Control Valve CV-0522A”;
- Off Normal Procedure 25.1 Attachment 23, “Fire Area 23 -Turbine Building,” Revisions 9, 10 and 11;
- Test Report for Palisades Special Test T-202, “Auxiliary Feedwater P-8A and P-8C,” dated March 6, 1987;
- Action Item Record A-PAL-90-139, “Steam Generator Pressure to be used as Design Basis for Auxiliary Feedwater Pump P-8C Flow,” dated August 26, 1991;
- Engineering Analysis EA-APR-95-024, “Auxiliary Feedwater Initiation, Main Steam Line Isolation, and Steam Generator Blowdown Isolation Requirements Under Appendix R Conditions,” Revisions 0 and 1;
- Engineering Analysis EA-GEJ-96-06, “Minimum Auxiliary Feed Requirement for All Auxiliary Feed Pumps,” Revision 0;
- Engineering Analysis EA-APR-95-025, “Make-up and Secondary Side Heat Removal Under Appendix R Conditions,” Revision 0;
- Engineering Analysis EA-APR-95-007, “10 CFR 50, Appendix R, Fire Safe Shutdown Analysis,” Revision 1;

- Engineering Analysis EA-APR-01-02, “Auxiliary Feedwater Initiation Requirements under Appendix R Conditions with Primary Coolant Pumps Running,” Revision 0; and
- Engineering Analysis EA-RCH-01-02, “Calculation of Auxiliary Feedwater Pump P-8C Capacity During a Postulated Turbine Building Fire,” Revision 0.

In addition, the inspectors reviewed the following condition reports concerning this temporary modification to verify that identified problems were appropriately characterized and evaluated:

- CPAL0000496, “Use of the Steam Driven Auxiliary Feedwater Pump During a Fire in the Turbine Building (Fire Area 23) is not Consistent Between Design Basis Document 1.03 and Engineering Analysis EA-APR-95-007";
- CPAL0100972, “Condition Report Significance Level Changed with No Supporting Documentation and Evaluation Conclusion is Incorrect”;
- CPAL0100975, “Design Basis Document 1.03 (Auxiliary Feedwater System) Description of Auxiliary Feedwater Pump P-8C Capability Contains Inconsistencies”; and
- CPAL0100259, “Removal of Auxiliary Feedwater Control Valve CV-0522A Supply to Auxiliary Feedwater Pump P-8B was not Adequately Reviewed Against Appendix R Analyses.”

b. Issues and findings

The inspectors identified a green finding and associated non-cited violation of 10 CFR 50, Appendix B, Criterion 3, “Design Control,” regarding a temporary modification that removed the underground (backup) steam supply to the Turbine Driven Auxiliary Feedwater Pump P-8B.

Temporary modification 2000-06 was developed and installed to isolate the underground (backup) steam supply line through Control Valve CV-0522A to Turbine Driven Auxiliary Feedwater Pump P-8B after the line catastrophically failed in February 2000. Licensee personnel determined that only one fire area was impacted by the temporary modification during the design change evaluation reviews. Specifically, a 10 CFR 50 Appendix R and Fire Protection Program review for Engineering Assistance Request EAR-2000-064, and Safety and Design Reviews SDR-00-0170/0199/0260 indicated that only Fire Area 26, the Southwest Cable Penetration Room, was impacted.

However, the inspectors identified that the Post-Fire Safe Shutdown Analysis initially credited Auxiliary Feedwater Pump P-8B via local manual operation of the underground (backup) steam line Control Valve CV-0522A for a fire in the Turbine Building. (Additional discussion on this is contained in Section 1RO5.) The inspectors determined that the conclusions made by licensee personnel in the safety and design reviews were based on an uncontrolled summary table document generated from Engineering Analysis EA-APR-95-007, “10 CFR 50, Appendix R, Fire Safe Shutdown Analysis.” Also, the summary table contained erroneous information on Fire Area 23 (Turbine Building). Consequently the safety and design review incorrectly concluded that only the southwest cable penetration fire area was affected by the design change.

The licensee's evaluation of Condition Report CPAL0100259 for this issue identified that there were several previous opportunities to identify this error. For example, the technical reviewer of the safety and design reviews identified that per Design Basis Document 1.03, a fire in the Turbine Building credited the use of Control Valve CV-0522A to achieve safe shutdown and that this needed to be identified in the safety evaluation. However, the resolution of this comment was narrowly focused and incorrectly concluded that the Design Basis Document information was not correct.

The inspectors concluded that this finding had a credible impact on safety, in that, the underground (backup) steam supply line was removed from service based on a design change that did not identify, evaluate and reconcile that the line was credited in the Post-Fire Safe Shutdown Analysis for a fire in the Turbine Building. In addition, removing the underground (backup) steam supply line affected the operability, availability and function of Turbine Driven Auxiliary Feedwater Pump P-8B, for a fire in the turbine building.

The NRC Senior Reactor Analyst requested that the licensee perform a quantitative risk assessment for this finding, since this issue could not be accurately evaluated through the Significance Determination Process. The inspectors and Senior Reactor Analyst reviewed the completed quantitative Probabilistic Safety Assessment and concluded that this issue was of very low safety significance (Green).

In addition, licensee personnel performed detailed reviews and engineering analysis which concluded the following:

- Engineering Analysis EA-APR-01-02, in conjunction with other engineering analyses, concluded that Auxiliary Feedwater Pump P-8C would have performed the required decay heat removal functions for a Turbine Building Fire; and

- Off-Normal and Emergency Operating Procedures that were in place when the temporary modification was installed provided adequate guidance to the operators to safely shut down the plant in the event of a Turbine Building fire.

- Licensee personnel discovered Final Safety Analyses Report, Chapter 14, engineering analyses from the early 1990s which documented that Auxiliary Feedwater Pump P-8C was able to perform the necessary decay heat removal functions for a Turbine Building Fire.

10 CFR 50, Appendix B, Criterion III, requires, in part, that design changes, including field changes, be subject to design control measures commensurate with those applied to the original design and that measures be established to ensure that deviations from such standards, including applicable regulatory requirements and the design bases, are controlled. However, licensee personnel failed to identify that the Post-Fire Safe Shutdown Analysis credited Auxiliary Feedwater Pump P-8B to remove decay heat using the underground (backup) steam supply line for a fire in the turbine building. Therefore, the full impact on the Post-Fire Safe Shutdown Analysis for removal of the underground (backup) steam supply line via the Temporary Modification 2000-06 design change was not evaluated or reconciled.

Consequently, design control measures were not commensurate to those applied to the original design and did not ensure that deviations from applicable regulatory requirements and the design bases were controlled. In accordance, with Section VI.A.1 of the NRC Enforcement Policy, this inspector identified example of a 10 CFR 50, Appendix B, Criterion III violation is being treated as a Non-Cited Violation. Unresolved Item URI 50-255/01-06-01 associated with this issue will also be closed based on this finding. This issue was entered into the licensee's corrective action program as Condition Report CPAL0100259. (NCV 50-255/01-06-03)

4. OTHER ACTIVITIES (OA)

4OA3 Event Follow-up

.1 (Closed) LER 50-255/97009-00: Procedure weakness in implementing Appendix R shutdown methodology. This item was discussed in NRC Inspection Report 50-255/97011(DRP), Paragraph E.1.1. It involved the Appendix R analysis assuming all four primary coolant pumps being tripped if the fire caused an evacuation of the control room. However, the Off Normal Procedure for alternate shutdown directed the operators to trip two of the four primary coolant pumps. This procedure did not only cover fires where a control room evacuation is to take place, but also provided guidance for fires where the control room is still manned. The procedure assumed that monitoring of the primary coolant pumps is a condition for continued operation of the primary coolant pumps.

Operators have additional guidance from Off Normal Procedure 6.2 "Loss of Component Cooling Water." This procedure directs further securing of the primary coolant pumps for degraded component cooling water flow. In addition, operator training included the need to secure primary coolant pumps when component cooling water is no longer capable of being monitored. The licensee had since revised the alternate shutdown procedure to direct operators to trip all four primary coolant pumps prior to control room evacuation. Although the alternate shutdown procedure did not address securing all primary coolant pumps, the licensee had another existing procedure to direct operators to secure additional primary coolant pumps. Therefore, the performance goals as stated in 10 CFR 50 Appendix R were satisfied. This item is not a violation of regulatory requirements. This item is closed.

4OA5 Other

.1 (Closed) Deviation 50-255/98003-02(DRS): Failure to measure component cooling water flow from 0 to 110 percent of flow using temperature instruments with sufficient indication range. The range of temperature instruments used to measure component cooling water flow (TE-0912 and TE-0913) was 0-180˚F. However, the licensee's sensitivity study indicated that the outlet temperature of component cooling water from the shutdown cooling heat exchanger would be 184˚F. Temperature elements TE-0912 and TE-0913 had a design temperature of 32 to 750˚F. However, the temperature indications in the control room, (TI-0912 and TI-0913) could indicate only from 0 to 180˚F. The licensee replaced Temperature Indicators TI-0912 and TI-0913 in December 1998 to indicating scales of 20-200˚F. The inspectors considered the corrective action acceptable. This item is closed.

.2 (Closed) URI 50-255/98011-03: A non-environmentally qualified cable installed in containment. This power cable (125V dc) provided power to Solenoid Valve SV-0347, which was required to de-energize and vent control air from safety injection tank pressure control valve CV-3047. Although this cable was not EQ qualified, the licensee determined that failure of this cable would not re-open Control Valve CV-3047 or prevent Control Valve CV-3047 from closing. In addition, its failure would not impact the accident scenario, any other cables, or nearby equipment. This cable was subsequently removed from the Equipment Qualification Master List maintained in accordance with 10 CFR 50.49 (d). It was not required to meet qualification program requirements specified in 10 CFR 50.49. The inspectors and the Office of Nuclear Regulation agreed with the licensee's analysis as documented in Task Interface Agreement 99-032. This item is not a violation of regulatory requirements and is closed.

.3 (Closed) VIO 50-255/96004-01: Operability of safe and alternate shutdown capabilities. There were seven parts to this violation and they are as follows:

- The potential transformer fuses for the Emergency Diesel Generator 1-1 were not properly coordinated such that Emergency Diesel Generator 1-1 was not properly isolated from associated circuits.
- Procedures did not exist to conduct cold shutdown repair to restore a low pressure safety injection pump.
- Ineffective corrective action for motor operated valves which could be affected by fire-induced hot shorts.
- Emergency Diesel Generator 1-2 power and control circuits were not adequately separated from the redundant train.
- Improper setting of the alternate shutdown panel inverter low voltage cut-off setpoint.
- The main supply fuses for the 125V dc panels, Electrical Disconnects ED-11-1 and ED-21-1 were not properly coordinated with the branch circuit breakers, and
- Inadequate emergency lighting had not been provided for certain locations.

Corrective actions to the above violations are as follows:

- The fuse in question was replaced in June 1996.
- The licensee provided augmented procedural guidance to operators and placed spare fuses in stock to support the repair activities.
- The licensee made necessary modification to the identified motor operated valves such that the limit and torque switches are below the starter coils. Any single hot short could change the valve position; however, the limit and torque switches would be able to prevent over-driving conditions.
- The licensee completed a separation analysis for Emergency Diesel Generator 1-1 and east air plenum rooms and determined that no modifications were required.
- The set point for the alternate shutdown panel inverter low voltage cut-off was reset.
- The licensee changed fuses so the equipment would be electrically coordinated.

- The licensee completed a station lighting blackout test and confirmed that adequate emergency lighting existed.

The inspectors considered the above corrective actions to be acceptable with one exception. Regarding the third corrective action, a combination of a hot short and a ground occurring on the same multi-conductor cable would still cause an over-driving condition for motor operated valves. This issue was also discussed as Unresolved Item 50-255/98013-02 pending generic resolution and will be tracked as part of that item. Therefore, this item is closed.

.4 (Closed) URI 50-255/98013-01: Unverified assumption for C-factor and the lack of analysis for hydraulic flow test data. The licensee used a C-factor of 100 for aged unlined cast iron underground piping which could be a non-conservative assumption. The fire protection piping flow test results for the last 20 years showed minimal deterioration in the underground piping. In addition, the licensee re-performed the hydraulic calculation using C-factor of 80 and 65. For sprinkler systems located in safety-related areas, the calculation showed that the system was able to meet the water density requirement. Therefore, the sprinkler systems remained operable. This item is closed.

4OA6 Exit Meeting

The inspectors presented the inspection results to Mr. Cowan and other members of licensee management on April 9, 2001. The licensee acknowledged the findings presented.

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

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. E. Cooper, Plant General Manager
J. P. Cowan, Senior Vice President Nuclear Management Company / Site Vice President
N. L. Haskell, Director, Licensing and Performance Assessment
R. J. Kilroy, Fire Protection
D. W. Rogers, Licensing
D. J. Malone, Engineering Director
K. Smith, Operations Manager
R. A. White, Nuclear Engineering

NRC

D. Hood, Project Manager, NRR

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-255/01-06-01	EEI	10 CFR 50.9, Completeness and accuracy of information regarding the licensee's submittals for a Notice of Enforcement Discretion and exigent Technical Specification Change Request that were granted to remove the underground (backup) steam supply to Auxiliary Feedwater Pump P-8B.
50-255/01-06-02	NCV	Failure to satisfy seismic requirements specified in plant procedures for scaffolding and storage racks constructed near safety-related equipment.
50-255/01-06-03	NCV	10 CFR 50, Appendix B, Criterion III, "Design Control," violation regarding inadequate design control during the review and approval of Temporary Modification 2000-06 that removed the underground (backup) steam supply to Auxiliary feedwater Pump P-8B.

Closed

50-255/01-06-02	NCV	Failure to satisfy seismic requirements specified in plant procedures for scaffolding and storage racks constructed near safety-related equipment.
50-255/01-06-03	NCV	10 CFR 50, Appendix B, Criterion III, "Design Control," violation regarding inadequate design control during the review and approval of Temporary Modification 2000-06 that removed the

		underground (backup) steam supply to Auxiliary feedwater Pump P-8B.
50-255/01-02-01	URI	Appendix R Fire Hazards Analysis for turbine building credited use of the underground (backup) steam supply to the turbine driven auxiliary feedwater pump which was removed from service one year ago.
50-255/97009-00	LER	Procedure weakness in implementing Appendix R shutdown methodology.
50-255/98003-02	DEV	Failure to measure component cooling water flow from 0 to 110 percent of flow using temperature instruments with sufficient indication range.
50-255/98011-03	URI	A non-environmentally qualified cable installed in containment.
50-255/96004-01	VIO	Operability of safe and alternate shutdown capabilities.
50-255/98013-01	URI	Unverified assumption for C-factor and the lack of analysis for hydraulic flow test data.

Discussed

50-255/98013-02	URI	Item three of VIO 50-255/96004-01 discussed a combination of a hot short and a ground occurring on the same multi-conductor cable which would cause over-driving condition for a motor operated valve. This specific issue is being tracked under this Unresolved Item pending generic resolution and will be tracked as part of that item.

LIST OF INSPECTIONS PERFORMED

The following inspection-area procedures were used to perform inspections during the report period. Documented findings are contained in the body of the report.

Inspection Procedure Number	Title	Report Section
71111.04	Equipment Alignment	1R04
71111.05	Fire Protection	1R05
71111.11	Licensed Operator Requalification Activities	1R11
71111.12	Maintenance Rule Implementation	1R12
71111.13	Maintenance Risk and Emergent Work	1R13
71111.14	Non-routine Evolutions	1R14
71111.15	Operability Evaluations	1R15
71111.16	Operator Workarounds	1R16
71111.17	Permanent Plant Modifications	1R17
71111.19	Post Maintenance Testing	1R19
71111.20	Refueling and Outage Activities	1R20
71111.22	Surveillance Testing	1R22
71111.23	Temporary Plant Modifications	1R23
71153	Event Follow-up	4OA3