U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No: License No:	50-255 DPR-20
Report No:	50-255/98003(DRS)
Licensee:	Consumers Energy Company
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
Dates:	March 25 - April 10, 1998
Inspector:	R. Westberg, Reactor Engineer
Approved by:	John Jacobson, Chief, Lead Engineers Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Palisades Nuclear Power Station
NRC Inspection Report 50-255/98003

This inspection reviewed the unresolved items and inspection follow-up items identified by the Design Inspection conducted from September 16 through November 14, 1997.

Engineering

- Good progress had been made in addressing the individual issues from the Design Inspection; however, the collective significance of the issues was still being reviewed.

- A violation was identified for a recent failure to scope and include in the inservice testing program, eight valves with specific functions in shutting down the reactor to a cold shutdown condition, in maintaining the cold shutdown condition, or in mitigating the consequences of an accident.

- Failure to follow procedures resulted in multiple violations:

    - Five examples were identified where recent safety related calculations were not revised when analytical inputs changed or were found to be in error as required by procedures.

    - Engineers failed to document justification of the acceptability of scaffolding installed adjacent to the safety related safety injection and refueling water tank and in the east engineering safeguards (ESG) room adjacent to safety related piping as required by the procedure.

    - An unsecured operations storage cabinet was found within nine feet of safety related valves CV-737 and CV-0747A in the west engineering safeguards room which was less than the procedure required 11.5 feet (cabinet height +5 feet).

    - Test results could not be located to verify that testing had been completed during the 1995 refueling outage for overcurrent relays for supply breakers 152-105 and 152-106 to Bus 1C as required by the procedure.

- A violation was identified for problems with the original plant design:

    - Two vent pipes, which connected the containment sump to the 590 ft elevation of the containment, did not have screens installed which were specified by the original design drawings. This piping configuration resulted in a pathway for debris to enter the recirculation system without being filtered by the containment sump screens with a potential to clog the containment spray nozzles.

    - Instrument tubing to the HPSI and LPS flow transmitters did not have the one inch per foot slope specified by the original design drawings.

- A deviation from a commitment to Regulatory Guide (RG) 1.97 was identified when CCW flow could not be measured from 0-110 percent of flow using the listed temperature instruments because their indication range was 0-180 °F and recent sensitivity studies indicated that the outlet temperature of CCW from the shutdown cooling heat exchanger would be 184 °F.

- A deviation from a commitment to RG 1.6 was identified when a design change moved the backup power source to a redundant power source, which resulted in Bus Y-01 being able to automatically transfer between two safety related busses.

Report Details

This inspection reviewed the items identified in the Palisades Nuclear Power Station, Design Inspection (NRC Inspection Report No. 50-255/97201) conducted from September 16 through November 14, 1997. The 18 unresolved items and 13 inspection follow-up items identified in the report are discussed below.

III. Engineering

E1 Conduct of Engineering

E1.1 Licensee Review of Collective Significance of the Issues

a. Inspection Scope (37550)

The licensee assessed the issues identified in the Design Inspection Report and issued internal commitments to address the programmatic significance in the areas of design control, calculation control, and setpoint control. In addition, the impact of the specific and programmatic inspection findings were also evaluated against the NRC's October 9, 1996 request for information pursuant to 10 CFR 50.54(f) regarding adequacy and availability of design basis information. The inspector reviewed the response to the Design Inspection report, the AE Inspection Actions Matrix dated March 20 and April 3, 1998, and documentation of corrective actions taken to date.

b. Observations and Findings

The inspector determined that the assessment of the collective significance of the issues identified in the Design Inspection Report was ongoing. While certain actions had been planned, such as the improvements to the Calculation Control Program, FSAR Verification and Validation Project, Setpoint Methodology and Control Program, and the Fuse Control Program, the scheduled completion date of these improvements was December 15, 1998.

Based on a review of the inspection findings and their comparison to the response to the 10 CFR 50.54(f) letter, the licensee concluded that their original response remained complete and accurate. However, to enhance knowledge of the plant's design basis, 10 additional Design Basis Documents and three safety system design confirmations similar to the NRC's safety system functional inspections were planned. A final review of the adequacy of the 50.54(f) response was scheduled for completion by December 15, 1998.

Pending NRC review of the results of the collective significance and planned programmatic improvements, this was considered an Inspection Followup Item (50-255/98003-01(DRS)).

c. Conclusions

Good progress had been made in addressing the individual issues from the Design Inspection; however, the collective significance of the issues was still being reviewed.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Open) Unresolved Item 50-255/97201-01 The licensee received a revised minimum flow requirement of 1600 gpm from the pump manufacturer. The team's review of the licensee's completed flow model calculation will be an Inspection Followup Item.

This item will remain open pending licensee completion of evaluation of the effects of higher predicted temperature on the CCW system and subsequent NRC review.

E8.2 (Open) Unresolved Item 50-255/97201-02 It appeared that the requirements of 10 CFR 50, Appendix B, Criterion III, "Design Control," were not met in this case in that the design basis for the CCW system, as defined in 10 CFR 50.2, did not encompass the entire range of bounding temperatures.

This item will remain open pending licensee completion of evaluation of the effects of higher predicted temperature on the CCW system and subsequent NRC review.

E8.3 (Closed) Unresolved Item 50-255/97201-03 Failure to perform IST in accordance with TSs for RV-0939.

RV-0939 is one of three relief valves inside containment for the CCW system. CCW piping inside containment is not required during an accident and is classified as non-Q, non-safety related. The system is Class JB rated at 125 psig with flanged joints and rubber gaskets. As a result of its function, the ISI/IST programs have classified the CCW system and related components, which includes RV-0939, as non-class and excluded it from the inspection and test requirements of the ASME Code.

Although RV-0939 is not required to be in the IST program, it is inspected, maintained, and its set point verified by preventive maintenance activity PPAC CCS043 on a ten year interval, which is essentially the same as the requirements of the Code, ASME OM-1987, Part1. This item is closed.

E8.4 (Closed) Unresolved Item 50-255/97201-04 Requirements of AP 9.11 were not fully met in that EA-GWO-7793-01, Revision 0, did not contain full substantiation of the conclusion.

Administrative Procedure 9.11, "Engineering Analysis," Revision 9, Section 2(d), stated that the analysis section of the engineering analysis (EA) will qualitatively and quantitatively (if applicable) present an argument which substantiates the conclusion of the EA and responds to the analysis objective. EA-GWO-7793-01, "CCW Piping Inside Containment HELBA," Revision 0, did not contain the necessary analysis to support the conclusion that the CCW piping inside containment was not affected by high-energy line break accidents. During the inspection, ES-GWO-7793-01 was revised to included a discussion of the walkdown analysis used to support the EA's conclusions.

The inspector determined that CCW piping inside containment was not required during an accident and was classified as non-Q, non-safety related. In addition, calculation ES-GWO-7793-01 was classified as non-safety related. No Violation of NRC requirements was identified. This item is closed.

E8.5 (Closed) Unresolved Item 50-255/97201-05 Failure to met a commitment to RG 1.97 in that the installed CCW temperature indicators were not capable of monitoring the full temperature range of the CCW system.

The inspector reviewed RG 1.97, UFSAR Appendix 7c, "Regulatory Guide 1.97 Instruments," and Condition Report (CR) C-PAL-97-1363E, Draft.

RG 1.97 described a method acceptable to the NRC staff for complying with the Commission's Requirements to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant and stated a range for CCW flow instrumentation of 0-110 percent of flow.

NRC letter to Consumers Power Company dated July 19, 1988, entitled "Palisades Plant - Response to Generic Letter 82-33 Conformances to Regulatory Guide 1.97, "Instrumentation for Light -Water-Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident," allowed use of temperature instruments to monitor CCW flow.

UFSAR Appendix 7C, Regulatory Guide 1.97, Rev 3, Parameter Summary Table, Type D Variables, Item D31, stated that the range of these temperature instruments used to measure CCW flow (TE-0912 and TE-0913) was 0-180 °F; however, recent sensitivity studies indicated that the outlet temperature of CCW from the shutdown cooling heat exchange would be 184 °F.

Failure to measure CCW flow from 0-110 percent of flow using temperature instruments with sufficient indication range is a deviation from a previous licensing commitment (50-255/98003-02(DRS)).

E8.6 (Closed) Unresolved Item 50-255/97201-06 Failure to perform IST in accordance with TSs which requires testing of valves which perform a safety function.

The inspector reviewed CRs C-PAL-97-1592, C-PAL-98-0427, C-PAL-98-0431, C-PAL-99-0433, and C-PAL-98-0477. The inspector also reviewed License Event Report 97-013, "Failure to Closure Test Two Check Valves Results in a Violation of Technical Specification 6.5.7."

Technical Specification 6.5.7 contained requirements for implementing an inservice testing (IST) program for ASME Code Class 1, 2, and 3 components, as required by the ASME Code. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, IWV-1100, "Valve Testing," states that valve testing shall be performed in accordance with the requirements stated in OM-10. Section 1.1, "Scope," of OM-10 states the active and passive valves covered are those which are required to perform a specific function in shutting down a reactor to cold shutdown condition, in maintaining the cold shutdown condition, or mitigating the consequences of an accident.

As of November 10, 1997, check valves CK-ES3339 and CK-ES3340 in the minimum flow recirculation piping from the discharge of each high pressure safety injection pump had a safety function to close to prevent the potential overpressurization of the pump suction piping. As of March 17, 1998, check valve CK-DMW400 in the flow path from the primary system make-up storage tank T-81 to the condensate storage tank T-2 had a safety function to open to supply make-up to the condensate storage tank. As of March 17, 1998, control valves CV-1813 and 1814 in the containment purge and ventilation system had an active safety function to close to provide a containment isolation function. As of March 26, 1998, control valves CV-1501, 1502, and 1503 in the plant heating system had an active safety function to close to provide a containment isolation function.

Failure to properly scope and include valves CK-ES3339, CK-ES3340, CK-DMW400, CV-1813, CV-1814, CV-1501, CV-1502, and CV-1503 in the IST program was a Violation of Technical Specification 6.5.7 (50-255/98003-03).

E8.7 (Closed) Unresolved Item 50-255/97201-07 Requirements of Procedure 9.11 regarding revising engineering analyses were not followed.

The inspector reviewed CR C-PAL-97-1558, "Nonconservative Input to ESR Heatup Calc EA-D-PAL-93-27F-01" and Administrative Procedure No. 9.11, "Engineering Analysis."

Administrative Procedure 9.11, "Engineering Analysis," Revision 9, Section 6.1.5.c, stated that an analysis shall be revised if analytical inputs changed.

In May of 1991, deficiency report F-CG-91-072 identified that Calculation EA-FC-573-2, "Calculated Required Air Flow for Inverter/Charger Cabinet Cooling Fan," assumed an ambient air temperature of 94 °F instead of the design basis temperature of 104 °F. F-CG-91-072 was closed in October 1994 without calculation EA-FC-573-2 being revised.

Failure to revise Calculation EA-FC-573-2 when the analytical input for design basis temperature was found to be in error was a Violation of 10 CFR 50, Appendix B, Criterion V (50-255/98003-04a(DRS)).

E8.8 (Closed) Unresolved Item 50-255/97201-08 Analysis were not revised when analytical inputs changed as required by administrative procedure 9.11

The inspector reviewed CRs C-PAL-97-1636, C-PAL-97-1603, C-PAL-97-1670, and Administrative Procedure No. 9.11, "Engineering Analysis."

Administrative Procedure 9.11, "Engineering Analysis," Revision 9, Section 6.1.5.c, stated that an analysis shall be revised if analytical inputs changed.

An assumption regarding pipe break size was not updated in EA-A-NL-92-185-01, "Worst Case Operating Conditions for the LPCI/SDC System MOVs," to determine the effect of the motor operated valves to close against the break when a more conservative pipe break assumption was used in a later analysis EA-C-PAL-95-1526-01, "Internal Flooding Evaluation for Plant Areas Outside Containment," Revision 0. Failure to update EA-A-NL-92-185-01 when a change regarding pipe break size was assumed is an example of a violation of 10 CFR 50, Appendix B, Criterion V (50-255/98003-04b(DRS)).

Assumptions 5.9 and 5.10 of EA-A-NL-92-185-01, which stated that the HPSI and LPSI flows to the loops were approximately equal under post-accident conditions were not revised when flow values calculated in EA-SDW-95-001, "Generation of Minimum and Maximum HPSI/LPSI System Performance Curves Using Pipe-Flo, " Revision 2, found that Assumptions 5.9 and 5.10 were incorrect. Failure to update EA-A-NL-92-185-01 when assumptions regarding flow rates were found to be incorrect is an example of a violation of 10 CFR 50, Appendix B, Criterion V (50-255/98003-04c(DRS)).

The required LPSI injection check valve flows identified in EA-E-PAL-93-004E-01, "IST Check Valve Minimum Flow rate Requirements to Support Chapter 14 Events, " Revision 0, were not revised after a new flow value was calculated in EA-SDW-95-001. Failure to revise EA-E-PAL-93-004E-01 when the analytical inputs changed or were found to be incorrect was a further example of violation of 10 CFR 50, Appendix B, Criterion V (50-255/98003-04d(DRS)).

E8.9 <u>(Closed) Unresolved Item 50-255/97201-09</u> Procedures MSM-M-43 and 1.01 and the "Palisades Ladder Control Policy for Operating Spaces" were not followed.

The inspector reviewed Maintenance Procedure MSM-M-43, "Scaffolding," Revision 2, Palisades Administrative Procedure 1.01, "Material Conditions Standards and Housekeeping Responsibilities," Revision 11, and CRs C-PAL-97-1417, C-PAL-97-1585, C-PAL-97-1586, C-PAL-97-1587, and C-PAL-97-1601.

Section 5.3.1 of MSM-M-43, "General," required that in addition to other requirements of this procedure, scaffolding constructed in the vicinity of safety related equipment shall not be used in any plant location which contains safety related equipment without prior engineering and approval and justification documented in Attachment 1, "data Sheet," Step 2.6. It also required that the responsible engineer provide justification and approval for any scaffold which deviates from the seismic requirements of this procedure, and document justification and approval in Attachment 1, "data Sheet," Step 2.6.

As of October 6, 1997, engineers had not reviewed the acceptability of scaffolding installed adjacent to the safety related safety injection and refueling water tank. In addition, on October 30, 1997, engineers had not reviewed the acceptability of scaffolding installed in the East engineering safeguards (ESG) room adjacent to safety related piping.

Failure to review and document the acceptability of scaffolding installed in the vicinity of safety related equipment was an example of Violation of 10 CFR 50, Appendix B, Criterion V (50-255/98003-05(DRS)).

Appendix 2 of Procedure 1.01 required that unrestrained and potentially damaging items which can topple should be separated from operable safety related equipment by a minimum horizontal distance equal to the height of the item plus five feet. During a plant tour on October 30, 1997, the inspectors observed an unsecured operations storage cabinet within 9 feet of safety related valves CV-0737 and CV-0747A in the West engineering safeguards room which was less than the required 11.5 feet (6.5 feet + 5 feet).

Failure to adequately maintain the required separation distance between an unsecured operations storage cabinet and safety related piping and valves in the West ESG room was an example of violation of 10 CFR 50, Appendix B, Criterion V (50-255/98003-06(DRS)).

E8.10 <u>(Closed) Unresolved Item 50-255/97201-10</u> A portion of the containment sump, designed to exclude debris from the ECCS pump suction piping, was not constructed in accordance with the design drawings.

The inspector reviewed drawing M-74, "Underground Piping Reactor Building," Sheet 1, Revision 10, drawing C-155, "Reactor Building Refueling Cavity and Sump Liner," Sheet 2, Revision 12, and UFSAR Section 6.4.2.3 which stated that the design of the spray nozzles was reviewed to confirm that the spray nozzles are not subject to clogging from debris entering the recirculation system through the containment sump screens. In addition, the inspector reviewed CRs C-PAL-97-1571 and C-PAL-97-1354.

During the Design Inspection, two vent pipes were identified which connected the containment sump to the 590-ft elevation of the containment, bypassing the containment sump screens. The design drawings specified screens on these two vent pipes; however, none were installed. Since the maximum predicted containment flood level was 597-ft which was two-ft above the top of these vent pipes, this piping configuration resulted in a pathway for debris to enter the recirculation system without being filtered by the containment sump screens. The licensee performed an operability assessment as part of C-PAL-1571 and concluded that the system was operable. Temporary modification TM-97-046, was installed on October 29, 1997 to add screens to the top of these vent pipes. Failure to correctly implement the design for the containment sump as specified in drawings M-74 and C-155 and UFSAR Section 6.4.2.3 was a Violation of 10 CFR 50, Appendix B, Criterion III (50-255/98003-07a(DRS)).

E8.11 (Closed) Inspection Follow up Item 50-255/97201-11 Review of licensee "extent of condition" review relative to rubber piping expansion joints used as penetration seals.

The inspector reviewed CR C-PAL-97-1627, "Inadequate Fire Barrier Evaluation." A review for similar conditions disclosed the existence of two similar fire barriers with rubber expansion joints in the floor of the CCW room above the West safeguards room. However, these expansion joints had been evaluated and the results documented in their respective engineering analyses. This item is closed.

E8.12 (Closed) Inspection Follow up Item 50-255/97201-12 Verify revision of setpoint methodology guide EGAD-PROJ-08 and training of engineers.

During the Design Inspection, EGAD-PROJ-08, "Design & Maintenance Guide on Instrument Setpoint Methodology," Revision 1, was approved and issued to provide guidance for instrument setpoint methodology. All Safety & Design Review Group Engineers were briefed as to the need to utilize this guidance. This item is closed.

E8.13 (Closed) Unresolved Item 50-255/97201-13 A portion of the instrument tubing to the HPSI and LPSI flow transmitters was not installed in accordance with the design drawings.

During a walkdown of the SI system, the inspectors observed that transmitters for containment spray flow, FT-0301 and FT-0302, and shutdown cooling heat exchanger flow, FT-0306, were properly mounted below their flow elements, but the process tubing was observed to be inadequately sloped back to the transmitters. Additionally, a walkdown performed by the licensee at the team's request during an in-containment inspection revealed that the process lines to the HPSI cold-leg flow transmitters FT-0308, FT-0310, FT-0312, and FT-0313 and the LPSI flow transmitters FT-0307, FT-0309, FT-0311, and FT-0314 were also installed with inadequate slope. The inspectors were concerned that inadequate slope in instrument tubing could contribute to significant instrument uncertainty by entraining unequal amounts of air in either leg of the transmitter, causing erroneous readings.

The inspector reviewed Drawing J-F-020, "Instrument Installation Notes - Flow," Revision 0, and Drawings J-F-152, "Flow Instrument Above Line W/Vents - Liquids," Revision 1 and J-F-153, "Flow Instrument Above Line W/Vents - Liquids," Revision 0. J-F-020 specified a 5-ft minimum drop leg before tubing is sloped to the meter to accommodate instruments mounted above flow elements and J-F-152 and 153 specified the installation of flow transmitters with a tubing slope of one inch per foot of instrument tubing run. The inspector also reviewed CRs C-PAL-97-1561 and C-Pal-97-1664.

Subsequent to the Design Inspection, the results of additional walkdowns determined that the HPSI and LPSI flow transmitters were properly installed in accordance with J-F-020; however, failure to properly implement the design basis for HPSI flow transmitters FT-0308, FT-0310, FT-0312, and FT-0313 and LPSI flow transmitters FT-0307, FT-0309, FT-0311, and FT-0314 and install instrument tubing with a one-inch per foot slope as specified in Drawings J-F-152 and 153 was a further example of Violation of 10 CFR 50, Appendix B, Criterion III (50-255/98003-07b(DRS)).

E8.14 <u>(Closed) Unresolved Item 50-255/97201-14</u> Calculations EA-ELEC-LDTAB-005 and EA-ELEC-VOLT-13 were not updated to document changes to plant parameters.

The inspector reviewed CR C-PAL-97-1619, "Electrical Engineering Calcs Not Updated to Reflect Changes in Plant Loads" and Administrative Procedure No. 9.11, "Engineering Analysis, Revision 9.

Administrative Procedure 9.11, "Engineering Analysis," Revision 9, Section 6.1.5.c, stated that an analysis shall be revised if analytical inputs changed. The team noted that EA-ELEC-VOLT-13, "Palisades Loss of Coolant Accident with Offsite Power Available," Revision 0, had not been revised since 1993 and that load magnitudes identified in EA-ELEC-LDTAB-005, Revision 4, and EA-SDW-95-001, Revision 2 had not been included. Failure to revise Calculation EA-ELEC-VOLT-13 when load magnitudes used as input to this calculation changed was a further example of Violation of 10 CFR 50, Appendix B, Criterion V (50-255/98003-04e(DRS)).

E8.15 <u>(Open) Inspection Follow up Item 50-255/97201-15</u> The licensee stated that evaluation of the effects of hot piping would be included under A-PAL-97-062.

The licensee will complete their Cable Ampacity Sizing Program by September 5, 1998, which will identify any cable degradation due to the close proximity of hot piping and any degradation of fire stops due to local heat sources. This item remains open.

E8.16 <u>(Closed) Unresolved Item 50-255/97201-16</u> Failure to meet a commitment to RG 1.6 in that an automatic transfer of loads between redundant power sources was created.

Licensee letter to the NRC dated January 24, 1978, stated that the recommendations of Regulatory Guide 1.6 would be implemented, in that, no provision would exist for automatically transferring loads between redundant power sources. NRC Safety Evaluation Report dated April 7, 1978, confirmed this commitment. Facility Change (FC)-364, "Feeder Change for Instrument Bus Y-01, Revision 0, implemented this commitment and powered bus Y-01 from Motor Control Center (MCC) 1 and non-safety related MCC 3; however, a subsequent modification, FC-854, moved the backup power source back to MCC-02. This modification also installed fuses between each of the MCCs and transfer switch Y-50; therefore, there was not a single failure vulnerability.

FC-854, moved the backup power source from MCC 3 to MCC 2, a redundant power source, which resulted in Bus Y-01 being able to automatically transfer between two safety related busses via transfer switch Y-50, which was a deviation from a previous licensing commitment (50-255/98003-08(DRS)).

E8.17 <u>(Open) Unresolved Item 50-255/97201-17</u> No system analysis existed to show that all the Class 1E 120Vac loads had adequate voltages.

The licensee will perform a bounding analysis by August 15, 1998, to confirm that Class 1E 120Vac loads have adequate voltage during accident conditions. This item remains open.

E8.18 (Closed) Unresolved Item 50-255/97201-18 Overcurrent relays for supply breakers 152-105 and 152-106 to Bus 1C had not been calibrated and tested as required by the surveillance test program.

The inspector reviewed CR C-PAL-97-1568 and the related operability assessment.

Periodic and Predetermined Activity APS025, "Bus 1C Relay Testing," required testing of the overcurrent relays. During the 1995 refueling outage work order 24416160 was issued dated June 28, 1995 to test the overcurrent relays for supply breakers 152-105 and 152-106 to Bus 1C. During the Design Inspection, the licensee discovered that no test results could be located for these relays. Plant records indicated that these relays had not been tested since 1992; however, the operability assessment in C-PAL-97-1568 found them operable based on low or lack of drift between documented calibrations and a lack of TS requirements for testing periodicity. Failure to calibrate the overcurrent relays for supply breakers 152-105 and 152-106 to Bus 1C was a further example of Violation of 10 CFR 50, Appendix B, Criterion V (50-255/98003-09(DRS)).

E8.19 (Open) Unresolved Item 50-255/97201-19 The design-basis lifetime for Agastat relays as stated by the manufacturer had not been correctly implemented in the facility.

During the A/E inspection, the licensee made an operability determination based on the E7000 series relay's similarity to the 7000 series relay. The operability determination concluded that the relays were operable. The licensee will complete their analysis of 7000 series and E7000 series in safety related applications by July 15, 1998. This item remains open.

E8.20 (Closed) Unresolved Item 50-255/97201-20 Failure to enter an LCO during battery charger switching evolution.

The inspector reviewed CR C-PAL-97-1537, Operating Procedure SOP-30, "Station Power," Revision 20, and Technical Specification 3.7.1h.

Battery charger 1 was supplied from MCC 1 and battery charger 3 was supplied from MCC 2. Administrative controls limited the operation so that only one charger per battery was in service. This prevented a common-mode failure from affecting both emergency busses. The supply to 125Vdc bus 2 was similar, with battery charger 2 fed from MCC 2 and battery charger 4 fed from MCC 1. Operating Procedure SOP-30, "Station Power," Revision 20, required the battery chargers to be operated in pairs (1 and 2 or 3 and 4). During the Design Inspection, the inspectors noted that TS

3.7.1h required two station batteries and the DC systems (including at least one battery charger on each bus) to be operable when the primary coolant system was above 300 °F.

The Station Blackout (SBO) calculations verified that the Class 1E batteries had the capacity to meet SBO loads for a period of four hours. In addition, in the event of a loss of coolant accident coincident with loss of offsite power with emergency generators available, one charger for each battery will be energized automatically to supply DC loads. Therefore, the station batteries will carry full load for approximately 10 seconds during this design basis accident and then they would be supported by the battery chargers. The time period when neither battery charger is connected to the 125Vdc bus during charger realignment would be expected to be shorter than the time period in the design basis when the batteries are expected to carry full load. Because of the short duration where the batteries carry full load, the batteries remain operable.

On December 27, 1995, a TS change request was submitted which revised the definition of 125Vdc bus operability based on specific bus voltages. In anticipation of the related TS amendment, operating procedure SOP-30 was revised to require an LCO entry whenever realigning battery chargers, an action more conservative than required by the existing TSs. The amendment was never issued. On January 26, 1998, the TS change request was resubmitted as part of the Improved Technical Specifications Program.

No violations of NRC requirements were identified, this item is closed.

E8.21 <u>(Open) Inspection Follow up Item 50-255/97201-21</u> Battery loading concern during LOOP/LOCA with single failure loss of AC power

The licensee will complete a formal analysis of battery loading considering the battery chargers are in their alternate alignment, a combined event of a LOCA/LOOP, and single failure of AC power by January 15, 1999. This item remains open.

E8.22 <u>(Open) Inspection Follow up Item 50-255/97201-22</u> Potential non-conservative TS Section 4.7.2c.

During the Design Inspection, an operability determination was made concluding that the 4-hr Station Blackout station battery load profile enveloped the 2-hr Design Basis Accident load profile. The licensee will complete a formal analysis of battery loading considering the battery chargers are in their alternate alignment, a combined event of a LOCA/LOOP, and single failure of AC power by January 15, 1999. This item remains open.

E8.23 <u>(Open) Inspection Follow up Item 50-255/97201-23</u> The team identified discrepancies concerning EA-ELECT-FLT-005 as part of an inspection follow up item.

The licensee plans to revise EA-ELECT-FLT-005, to correct the deficiencies by January 15, 1999. This item remains open.

E8.24 <u>(Open) Inspection Follow up Item 50-255/97201-24</u> Lack of analysis to ensure that adequate voltages would exist at the load terminals of the batteries.

The licensee will perform a bounding analysis to identify the worst-case minimum voltage levels at the load terminals to assure that minimum load voltage requirements are met by November 15, 1998. This item remains open.

E8.25 (Closed) Unresolved Item 50-255/97201-25 It appeared that the requirements of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," were not followed in that the design basis for the solenoid valve coils was not implemented in the plant.

The team questioned the capability of solenoid valves to operate at voltages of 87 Vdc as stated in DBD 1.01, "Component Cooling Water System," Revision 4. The licensee determined that the DBD was incorrectly worded and that the correct solenoid capability was 90-140 Vdc. Upon further review, the licensee identified that improperly rated coils, rated 102-126 Vdc, were installed in solenoid valves SV-0918 and SV-0977B. Engineering Assistance Request (EAR) 97-0652 was initiated to replace the coils.

Subsequent to the inspection, the licensee determined that there was no impact on the mitigation of an accident if solenoid valves SV-0918 and SV-0977B failed to open due to low voltage, since the closed position was both the failed position and the required safety position. In addition, ASCO catalog No. NP-1 stated that all ASCO valves are tested to operate at 15% under the nominal voltage

No violations of NRC requirements were identified, this item is closed.

E8.26 (Closed) Inspection Follow up Item 50-255/97201-26 Battery calculation discrepancies.

The discrepancies identified were minor in nature and did not affect the conclusions of the analyses. Supplied voltages remained within the equipment rating and the station batteries were not affected. This item is closed.

E8.27 (Closed) Inspection Follow up Item 50-255/97201-27 Section 3.0 of the Acceptance Criteria and Operability Sheet for Procedure RO-128-2 referred to TS Sections 3.7.1 and 4.7.1.11, and that these references would only be correct when the proposed improved TSs, which have been submitted to NRC for approval, became effective.

On January 26, 1998, a request for improved technical specifications was submitted which specified testing the diesel generators to the load intervals programmed by the sequencer and eliminated specific references to the sequence time intervals. This item is closed.

E8.28 (Open) Inspection Follow up Item 50-255/97201-28 Discrepancies in station battery test procedures RE-83A and B.

The licensee will revise surveillance tests RE-83A and B as appropriate to support the 1998 refueling outage. The licensee will also review DC system requirements by December 15, 1998. This item remains open.

E8.29 (Closed) Inspection Follow up Item 50-255/97201-29 The 10 CFR 50.59 safety evaluations were adequate, except for two examples:

Safety Reviews 95-1431 and 95-1432, dated July 7, 1995, for FES-95-206 stated that the battery duty cycle service test duration for station batteries ED-01 and ED-02 was changed from 2 hours to 4 hours. The licensee noted that TS Section 4.7.2.c was affected by this design change. However, the USQ evaluation, Question 2 of Section II, was not checked "Yes" for a TS change. TS 4.7.2.c required that a 2-hour battery test be performed; while design analysis ELEC-LDTAB-009 and FSAR Section 8.4.2 required a 4-hour battery duty cycle. The licensee has submitted a proposed TS change to reflect the proper battery test duration and issued CR C-PAL-97-1551 to address this discrepancy.

The preparer of the safety review did not consider that a TS change was necessary for FES-95-206 to eliminate reference to a specific duty cycle time since that TS change was planned to be submitted under the Improved Technical Specifications Program. The required TS change was subsequently submitted on January 26, 1998, as part of that program.

The safety review documentation for TM-96-027 stated that the FSAR was not reviewed. Administrative Procedure 3.07, "Safety Evaluations," page 12, required that the FSAR be reviewed and that those sections reviewed be noted on the safety review sheet. The licensee initiated C-PAL-97-1439 to evaluate this discrepancy.

The safety review, PS&L Log No. 96-05508, for temporary modification, TM-96-027, "Install 152-Spare #5 Breaker in 152-113 Cubicle," was approved via telecon. It inappropriately indicated that the FSAR had not been reviewed when in actuality, the FSAR was reviewed and found not to discuss the level of detail contained in the TM, that is, auxiliary contact configuration. The safety review was correctly revised and refiled with the original TM.

This item is closed.

E8.30 (Open) Unresolved Item 50-255/97201-30 Discrepancies had not been corrected and the FSAR had not been updated to ensure that the material in the FSAR contained the latest material.

- Section 6.7 classified the CCW penetrations as Class C-2, which was defined as penetrations with lines not missile protected. However, EA-GWO-7793-01 stated that the entire CCW system (both inside and outside containment) was missile protected. The licensee issued FSAR Change Request 6-143-R20-1427 to state that the CCW penetrations were not vulnerable to internally generated missiles.

  The CCW system was not designed to be missile protected. The statement in EA-GWO-7793-01 refers to the fact that due to system configuration the system is effectively protected from missiles, i.e., not vulnerable. The licensee issued a FSAR change clarify this point. This portion of the unresolved item is closed.

- Section 8.4.2.2 stated that the station batteries would be tested to Institute of Electrical and Electronics Engineers (IEEE) 450-1975. However, battery Testing Procedures RE-83A, Revision 9, and RE-83B, Revision 9, referred to IEEE 450-1995. FSAR Change Request 8-126-R20-1249 had been initiated, but the licensee did not intend to act on this change until approval was received from NRC of a related proposed TS change.

  The TS change request , which cites IEEE 450-1995 for battery testing, was submitted to the NRC on January 26, 1998. This portion of the unresolved item is closed.

- Table 5.7-8 listed the seismic design value for the station batteries and racks as "later" instead of including the actual values of the batteries installed by FES 95-206. The licensee issued EAR 97-0636 to evaluate this discrepancy and revise the FSAR.

  Table 5.7-8 was designated as containing the original seismic design values. The use of the term "later" was used in the original FSAR because at that time there was a planned upgrade to install a second redundant electrical train and the seismic criteria were not available. The licensee will remove the word "later" as a clarification and maintain the table as the original seismic design criteria. This portion of the unresolved item is closed.

The remaining portions of this unresolved item remain open. For the UFSAR deficiencies identified relative to the DC system, the licensee will review DC system requirements by December 15, 1998.

E8.31 (Open) Unresolved Item 50-255/97201-31 Documentation discrepancies were identified in the design basis documents (DBDs).

Design Basis Document Change Requests were generated and will be incorporated into the DBDs by December 15, 1998. This item remains open.

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of the inspection on April 10, 1998. The licensee acknowledged the findings presented.

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

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Rogers	General Manager - Plant Operations
G. Szczotka	Manager NPAD
D. Malone	Configuration Control Manager
N. Haskell	Licensing Director
K. Haas	Engineering Director
S. Wawro	Director Maintenance and Planning
K. Toner	Licensing Supervisor
R. Westerhof	Configuration Control
R. Brzezinski	Design

Nuclear Regulatory Commission

J. Lennartz Senior Resident Inspector

INSPECTION PROCEDURES USED

IP 37550	Engineering
IP 92903	Follow up on previously identified items.

ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

50-255/97201-03	URI	Failure to perform IST in accordance with TSs for RV- 0939.
50-255/97201-04	URI	Requirements of AP 9.11 were not fully met in that EA-GWO-7793-01, Revision 0, did not contain full substantiation of the conclusion.
50-255/97201-05	URI	Failure to met a commitment to RG 1.97 in that the installed CCW temperature indicators were not capable of monitoring the full temperature range expected for the CCW system.
50-255/97201-06	URI	Failure to perform IST in accordance with TSs which requires testing of valves which perform a safety function.
50-255/97201-07	URI	Requirements of Procedure 9.11 regarding revising engineering analyses were not implemented.
50-255/97201-08	URI	Analysis were not revised when analytical inputs changed as required by administrative procedure 9.11
50-255/97201-09	URI	Procedures MSM-M-43 and 1.01 and the "Palisades Ladder Control Policy for Operating Spaces" were not followed.
50-255/97201-10	URI	A portion of the containment sump, designed to exclude debris from the ECCS pump suction piping, was not constructed in accordance with the design drawings.
50-255/97201-11	IFI	Review of licensee "extent of condition" review relative to

		rubber piping expansion joints used as penetration seals.
50-255/97201-12	IFI	Verify revision of setpoint methodology guide EGAD-PROJ-08 and training of engineers.
50-255/97201-13	URI	A portion of the instrument tubing installation to the HPSI and LPSI flow transmitters was not installed in accordance with the design drawings.
50-255/97201-14	URI	Calculations EA-ELEC-LDTAB-005 and EA-ELEC-VOLT-13 were not updated to document changes to plant parameters.
50-255/97201-16	URI	The safety evaluation performed for FC 854 did not identify that prior NRC approval was required.
50-255/97201-18	URI	Overcurrent relays for supply breakers 152-105 and 152-106 to Bus 1C had not been calibrated tested as required by the surveillance test program.
50-255/97201-20	URI	Failure to enter an LCO during battery charger switching evolution.
50-255/97201-25	URI	The design basis for the solenoid valve coils was not implemented in the plant.
50-255/97201-26	IFI	Battery calculation discrepancies.
50-255/97201-27	URI	Section 3.0 of the Acceptance Criteria and Operability Sheet for Procedure RO-128-2 referred to TS Sections 3.7.1 and 4.7.1.11, and that these references would only be correct when the proposed improved TS.
50-255/97201-29	IFI	The 10 CFR 50.59 safety evaluations were adequate, except for two examples.

Opened

50-255/98003-01	IFI	Pending NRC review of the results of the programmatic improvements and the 10 CFR 50.54(f) comparison
50-255/98003-02	DEV	Deviation from a RG 1.97 commitment.
50-255/98003-03	VIO	Failure to properly scope valves CK-ES3339, CK-ES3340, CK-DMW400, CV-1813, CV-1814, CV-1501, CV-1502, and CV-1503 and include them in the IST program
50-255/98003-04	VIO	Failure to follow procedures and update calculations when analytic inputs changed.
50-255/98003-05	VIO	Failure to follow procedures and review and document the acceptability of scaffolding installed in the vicinity of safety related equipment
50-255/98003-06	VIO	Failure to follow procedures and adequately maintain the required separation distance between an unsecured operations storage cabinet and safety related piping and valves in the West ESG room
50-255/98003-07a	VIO	Failure to correctly construct a portion of the containment sump in accordance with the design drawings.
50-255/98003-07b	VIO	Failure to correctly install instrument tubing for the HPSI and LPSI flow transmitters with the correct slope.
50-255/98003-08	DEV	Deviation from a RG 1.6 commitment.
50-255/98003-09	VIO	Failure to test overcurrent relays as required

Discussed

50-255/97201-01	IFI	Review of the licensee's completed flow model calculation.
50-255/97201-02	URI	The design basis for the CCW system, as defined in 10 CFR 50.2, did not encompass the entire range of bounding temperatures.
50-255/97201-15	IFI	The licensee stated that evaluation of the effects of hot piping would be included under A-PAL-97-062.
50-255/97201-17	IFI	No system analysis existed to show that all the Class 1E 120-V ac loads had adequate voltages.
50-255/97201-19	URI	The design-basis lifetime for Agastat relays as stated by the manufacturer had not been correctly implemented in the facility.
50-255/97201-21	IFI	Battery loading concern during LOOP/LOCA with single failure loss of AC power
50-255/97201-22	IFI	Potential non-conservative TS Section 4.7.2c.
50-255/97201-23	IFI	The team identified discrepancies concerning EA-ELECT-FLT-005 as part of an inspection follow up item.
50-255/97201-24	IFI	Lack of analysis to ensure that adequate voltages would exist at the load terminals of the batteries.
50-255/97201-28	IFI	Discrepancies in station battery test procedures RE-83A and B.
50-255/97201-30	URI	Discrepancies had not been corrected and the FSAR had not been updated to ensure that the material in the FSAR contained the latest material.
50-255/97201-31	URI	Documentation discrepancies were identified in the design basis documents.

LIST OF ACRONYMS USED

AE	Architect/Engineers
ASME	American Society of Mechanical Engineers
CCW	Component Cooling Water
DBD	Design Basis Document
DEV	Deviation
EA	Engineering Analysis
EAR	Engineering Assistance Request
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FC	Facility Change
HPSI	High Pressure Safety Injection
IFI	Inspection Follow-up Item
IST	In-service Testing
LOCA	Loss of Cooling Accident
LOOP	Loss of Offsite Power
LPSI	Low Pressure Safety Injection
MCCs	Motor Control Centers
QA	Quality Assurance
SBO	Station Blackout
TS	Technical Specification
URI	Unresolved Item
VIO	Violation