OFFICE OF NUCLEAR REACTOR REGULATION

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Licensee: Wolf Creek Nuclear Operating Corporation

Facility: Wolf Creek Generating Station

Location: 1550 Oxen Lane, NE
Burlington, Kansas

Dates: November 3, 1997, through January 9, 1998

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EXECUTIVE SUMMARY

A design inspection at Wolf Creek Generating Station was performed by the Events Assessments, Generic Communications, and Special Inspection Branch of the Office of Nuclear Reactor Regulation (NRR) during the period November 3, 1997, through January 9, 1998. This inspection included onsite inspections during November 1[illegible] December 1-5, December 15-19, 1997, and January 5-9, 1998. The inspection team consisted of a team leader from NRR, one inspector from NRR, and five contractor engineers from Sargent & Lundy Corporation (S&L).

The purpose of the inspection was to evaluate the capability of the selected systems to perform safety functions required by their design bases, the adherence of the systems to their design and licensing bases, and the consistency of the as-built configuration with the updated safety analysis report (USAR). The team selected the residual heat removal (RHR) system, the component cooling water (CCW) system, and their support-interface systems for this inspection because of the importance of these systems in mitigating design basis accidents at Wolf Creek. The team followed the engineering design and configuration control section of Inspection Procedure (IP) 93801 for this inspection. For the selected systems, the team reviewed the USAR, system descriptions, calculations, drawings, modification packages, surveillance procedures, and other documents.

Except as noted below, the team determined that the selected systems were capable of performing their design-basis safety functions and that design and licensing bases were adequately adhered to. The licensee implemented appropriate measures to resolve the immediate concerns identified by the team, and no immediate operability concerns exist. For other issues, the licensee initiated appropriate reviews and evaluations using the corrective action process or took corrective actions such as revising design documents and changing procedures.

The team identified the following weaknesses in 10 CFR 50.59 evaluations and design changes: (1) The design change and safety evaluation for the replacement of Class 1E batteries with AT&T round cell batteries did not address the effect on Technical Specifications (TS). Currently, TS Sections 4.8.2.1.e and f regarding battery capacity replacement criteria and battery degradation criteria appear to be nonconservative because the batteries were sized without aging factors and the battery performance characteristic was changed. The battery design capacity margin was less than that stated in the staff's safety evaluation report (NUREG-0881) and USAR. The NRR staff will review the design change to determine the adequacy of the existing TS. (2) The design change and safety evaluation for lowering the CCW temperature did not address the effects of low temperature on the spent fuel pool reactivity and on diesel generator loading. (3) A reactor coolant system (RCS) draindown procedure for installing the nitrogen bottles during plant refueling outages did not have a safety evaluation to address seismic restraint requirements to preclude potential missiles.

10 CFR 50.59 evaluations generally lacked adequate documented justification. The licensee had recently revised its procedure to emphasize that sufficient details and basis should be supplied in 10 CFR 50.59 evaluations.

Surveillance testing of batteries was deficient because the battery capacity and service test procedures did not provide appropriate acceptance criteria. The recent capacity test for the NK11 battery did not identify and properly evaluate the impact on battery capacity when the battery load current was not maintained constant during the test.

The team identified calculations with errors or inappropriate or nonconservative assumptions. In some cases analysis did not exist to support the design bases. For example: the refueling water storage tank instrument (RWST) loop uncertainty calculations did not consider density variations due to temperature and boron concentrations which affected the alarm setpoints, the swapover setpoints, and the level indications; the Westinghouse cooldown analysis did not assume correct essential service water flow; a dc voltage drop calculation did not identify the worst-case minimum battery voltage; nonconservative downstream pressures were assumed for some CCW motor-operated valves in valve closure calculations resulting in incorrect design differential pressures for valves; the calculation to estimate the maximum control circuit wire lengths for motor control center starter control circuits did not model the auxiliary loads correctly; and no analyses existed to show that the 120 Vac feeders and control circuits are protected adequately during a fault, and that 120 Vac safety-related loads have adequate voltages.

The team noted that several calculations for each system have similar purposes and/or similar results. These calculations were not contradictory, but the practice tends to confuse identification of the design basis.

In two design change packages, the electrical calculation was not revised in accordance with the licensee's procedure when dc load changes occurred. There was an inconsistency between the electrical load growth control procedure and the engineering screening form regarding the threshold for analyzing electrical load changes and discrepancies with some dc load changes in the licensee's load data base

Other discrepancies identified included the following: the licensee did not correct a previously identified design basis requirement discrepancy in operating procedures for the CCW system; the electrical distribution system was not modeled down to the 208/120 volt level or compared to field voltage measurements as specified in Branch Technical Position P.B.-1; and no documentation existed to show the RHR pump suction pressure gauges were seismically qualified to maintain the RCS pressure integrity.

The as-built configurations of the systems were generally consistent with the USAR. In general, the availability of the design bases documentation was good, as was the material condition of the areas observed by the team. However, the team identified a number of discrepancies in the USAR, system descriptions, and other plant documents.

The team's findings indicated an ongoing need to emphasize design and configuration-control related issues in maintaining the design and licensing bases for Wolf Creek. Wolf Creek had established a design basis/licensing basis (DB/LB) review program to address these types of concerns. This new DB/LB program had not yet produced widespread or consistent results at the time of the inspection.

III. Engineering

E1.0 CONDUCT OF ENGINEERING

E1.1 Inspection Scope and Methodology

The purpose of the inspection was to evaluate the capability of the selected systems to perform safety functions required by their design bases, the adherence of the systems to their design and licensing bases, and the consistency of the as-built configuration with the updated safety analysis report (USAR). The systems selected for inspection were the residual heat removal (RHR) system, the component cooling water (CCW) system, and their support-interface systems. These systems were selected on the basis of their importance in mitigating design basis accidents at Wolf Creek.

The inspection was performed in accordance with NRC Inspection Procedure (IP) 93801, "Safety System Functional Inspection." The engineering design and configuration control section of the procedure was the primary focus of the inspection.

The open items resulting from this inspection are included in Appendix A. The acronyms used in this report are listed in Appendix C.

E1.2 Residual Heat Removal (RHR) System

E1.2.1 Mechanical Design Review

E1.2.1.1 Inspection Scope

The team evaluated the capability of the RHR system to achieve and maintain cold shutdown and to mitigate the consequences of a small or large break loss-of-coolant-accident (LOCA). The team also reviewed portions of the high head and intermediate head safety injection system that interfaces with the RHR system, the refueling water storage tank (RWST), and portions of the containment spray system.

As part of this effort the team reviewed the plant design drawings, calculations, accident analyses, the containment flooding analysis, design change packages, the USAR, the system design description, technical specifications, operating procedures, maintenance and surveillance tests, information notices, generic letters, environmental qualification (EQ) files, and engineering evaluations associated with the system.

E1.2.1.2 Observations and Findings

a. RHR System Flow and Decay Heat Removal Requirements

The team evaluated the capability of the RHR system to remove the sensible and decay heat generated in the reactor, and to achieve and maintain the plant in cold shutdown. The time required to bring the plant to cold shutdown so that neither the cooldown rate nor the cooldown time is exceeded is dependent on the RHR flow and the temperature of the component cooling water (CCW), which again is dependent on the essential service water (ESW) system that cools the CCW. In the current Westinghouse analysis, FSDA-C-365, "NSSS Uprating Analysis," Revision 1, which was performed for power uprate, it was determined that the plant can be

brought to cold shutdown in about 17 hours with both RHR trains in operation. The Westinghouse calculation used reactor coolant system (RCS), RHR, and CCW flow rates and temperatures that were consistent with design bases. However, this calculation used ESW flow rates (13500 gpm) to the CCW heat exchangers that were higher than used in licensee calculation EG-06-W, "Determine Flow and Heat Load Requirement," Revision W3 (8800 gpm). Calculation EG-06-W included flow effects associated with plugging 46 tube pairs to provide margin for the future should plugging of a significant number of CCW heat exchanger tubes become necessary. Westinghouse did not use the tube plugging assumption in their analysis. Currently only two tubes in one heat exchanger are plugged. A preliminary evaluation of cooldown time with appropriate assumptions indicated that cooldown will be consistent with current commitments to achieve cooldown in 20 hours as stated in USAR Section 5.4.2.1.7 at a cooldown rate not exceeding 100°F as specified in the TS. The licensee issued PIR 97-4145 to resolve the design issue. The team determined that the licensee's design control measures did not meet the requirements specified in Criterion III of Appendix B to 10 CFR Part 50 regarding verifying or checking the adequacy of the design. (Unresolved Item 50-482/97-201-01)

The team reviewed licensee calculations SA-90-067, "Calculation of the Decay Heat Load to the RHR Heat Exchanger Following Normal Shutdown," Revision 0, and SA 92-074, "Decay Heat Load to the RHR Heat Exchanger Following Normal Shutdown for Uprated Power to 3565 MWt," for decay heat removal. Whereas the above Westinghouse analysis determined the cooldown for the most limiting design case for CCW temperature, the licensee's analysis evaluated the cooldown time for various CCW temperatures. For lower CCW temperatures, cooldown can be achieved faster, but to maintain the cooldown rate within TS limits licensee's procedures SYS EJ-120, "Startup of a Residual Heat Removal Train," and SYS EJ-121, "Startup of RHR Train in Cooldown Mode," are implemented. The team determined that the above procedures are adequate to maintain the cooldown rate within the TS limit.

The team reviewed licensee calculations SA-89-009, "The Minimum RHR Flow Requirement for Decay Heat Removal During Mid-Loop Operation," Revision 0; AN-93-009, "Minimum RHR Flow Requirement for Decay Heat Removal During Mid-Loop Operation to Support the Power Rerate Program," Revision 0; and RE-EJ-005, Revision 1, which were performed to demonstrate that low water level in the reactor would not limit RHR flow for decay heat removal and allow vortexing and air binding of the RHR pumps. The team also evaluated the licensee's responses for information notices IN 90-06, "Potential for Loss of Shutdown Cooling while at Low Reactor Coolant Levels due to Loss of Power to the RHR Flow Control Valve;" IN 87-23, "Loss of Decay Heat Removal During Low Reactor Coolant Level Operation (Loss of Residual Heat Removal) (Diablo Canyon Event);" and IN 86-101, "Loss of Decay Heat Removal due to Loss of Fluid Levels in Reactor Coolant System," and operating procedures GEN-007, "RCS Drain Down," and GEN-008, "Reduced Inventory Operations." Based on a review of the above documents the team concluded that as long as mid-loop operation was delayed for a minimum of 48 hours after cold shutdown was initiated, vortexing and air binding of the RHR pumps would not occur.

b. RHR System Flow Requirements and Flow Rates for Emergency Core Cooling

To evaluate RHR system capability to provide the limiting flow specified in Westinghouse report WCAP-13447, "3579 MWt NSSS Rerating Engineering Report," Volume 1, dated October 1992, the team reviewed the licensee's calculations for RHR system hydraulic resistance, piping isometric drawings, and pump surveillance test procedures. For the small break LOCA analysis, no credit is taken for flow from the RHR system. The team's review for the small break LOCA was, therefore, essentially limited to verifying the "piggyback" mode of operation of the RHR

system to provide flow to the suction of the intermediate and high head pumps when their suction is switched from the RWST to the containment sump.

Injection Phase

The calculation that determines the RHR flow rates during the injection phase of emergency core cooling system (ECCS) operation for the large and small break LOCA is documented in licensee calculation SA-91-016, "ECCS Design Basis Flow Rates Reanalysis in Support of the WCGS Re-rating Project," Revision 0. The calculated RHR flow rates were provided as input to Westinghouse to use in the LOCA analysis which is summarized in the above Westinghouse report. The required and calculated total RHR flow injecting to the three intact loops at an RCS pressure of 0 psig is about 2834 gpm. Calculation SA-91-016 very conservatively assumed a 10 percent pump degradation and open pump miniflow lines. The above RHR pump flow is assured per TS Surveillance Requirement 4.5.2.i, which verifies the total pump flow is equal to or greater than 3800 gpm and equal to or less than 5500 gpm, which is the pump runout flow.

Cold Leg Recirculation Phase

No specific evaluation had been performed by the licensee to determine the minimum ECCS flow requirement during cold leg recirculation. Guidance provided in Westinghouse document NSAL-95-001, "Minimum Cold Leg Recirculation Flow," dated January 12, 1995, requires that the ECCS flow during cold leg recirculation should be at least 1.2 times decay heat boiloff when cold leg recirculation is initiated. The team determined that this amounted to about 608 gpm for a cold leg switchover time of 0.4 hours. The results of licensee calculations SA-92-056, "CCP and SI Pump Runout Flowrate in Recirculation Phase," Revision 0, and AN-95-021, "Determine ECCS Flow Rates in Recirc. Phase," Revision 0, which were performed to evaluate the runout flow of ECCS pumps under "piggyback" operation, provide reasonable assurance that sufficient cooling flow during cold leg recirculation is available from one operating ECCS train (the other ECCS train assumed inoperable).

Hot Leg Recirculation Phase

The team reviewed the minimum required cooling flow for a large break LOCA during hot leg recirculation that was specified in Westinghouse document NSAL-92-010, dated January 9, 1993. The licensee's flow calculations discussed earlier indicated that the RHR and ECCS pumps had the capability to provide the required flow, assuming the most limiting single failure specified in the Westinghouse document.

The team reviewed industry technical information program (ITIP) No. 02178, which was in response to a 10 CFR Part 21 notification from Westinghouse regarding the potential for boron precipitation in the reactor core in the long-term cooling mode following a postulated LOCA. This evaluation confirmed that sufficient flow is available from the intermediate and high head pumps to satisfy the minimum flow requirement during the hot leg recirculation, assuming a single failure of the RHR hot leg header isolation valve and a loss of one diesel generator so that only one train of ECCS is available. The team also reviewed Westinghouse calculation SEC-TSA-3958-CO, "Wolf Creek (SAP) Hot Leg Switchover Time for Power Uprating," Revision 0, which determines how long after a LOCA a switchover to hot leg recirculation should be initiated to prevent boron precipitation in the reactor vessel. This calculation determined that the minimum switchover time to hot leg recirculation is 10 hours. Emergency Operating Procedure EMG ES-13, "Transfer to Hot Leg Recirculation," was also reviewed to verify the switchover time to hot leg recirculation. No unacceptable conditions were identified.

Pump Performance

The team reviewed the pump quarterly surveillance test (procedure STS EJ-100A) and trending data on pump degradation to verify the pump's ability to provide the required flow. The surveillance test and trending data over the last 4 years indicated no pump degradation. Also, a review of procedure STN EJ-100A, "RHR Pump "A" Reference Curve Determination," showed that the reference curve determined in 1993 showed no noticeable degradation in pump performance when compared to the original pump performance curve.

c. RHR Pump Net Positive Suction Head (NPSH)

The team reviewed calculation ECCS-35, "RWST to RHR pump "A" Suction Mode A," and TS Section 3/4.5.5 to verify the adequacy of the NPSH for the RHR pumps when taking suction from the RWST. The NPSH was determined based on the level in the RWST at the time of switchover to the containment sump. The team determined that the available NPSH was much greater than the NPSH required. The team also concluded that since the head of water above the suction piping at the time of switchover to the containment sump was about 14 feet, the probability of any air ingestion by the RHR pumps was very low.

The team's evaluation of the available NPSH for the RHR pumps when taking suction from the containment sump was based on the guidance provided in Regulatory Guide (RG) 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," and NRC Generic Letter (GL) 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps." The team reviewed calculations and analyses EJ-30, "RHR Pumps A&B NPSH," Revision 1; EJ-29, "RHR Flow Orifice Sizing," Revision 0; FL-18, "LOCA & MSLB Ctmt. Flood Levels," Revision 1; USAR Table 6.2.2-6; and Containment Recirculation Sump Hydraulic Performance analysis. The review determined that the NPSH available for the RHR pumps was greater than the NPSH required. The RHR pump suction lines from the containment sump have vortex breakers installed in the pipe inlets. The vortex suppressor test results in the "Sump Hydraulic Performance Report," showed that the suppressor was effective in decreasing both vortex activity and swirl. The licensee's tests indicated no vortex activity for RHR pump operation at the water levels at which ECCS pump switchover and containment spray pump switchover occur.

In determining the original setpoints for the RWST level instruments, no consideration was made for the temperature and boron density (see Section E1.2.3.2.1). The team noted that any corrections to the setpoints to account for the temperature and boron density could alter the volume of water transferred to the containment sump and affect the containment flood level and the available NPSH for the RHR pumps. Based on preliminary input provided to the team by the licensee regarding setpoint changes, the team determined that the available NPSH would decrease but not significantly enough to affect pump performance.

d. ECCS Leakage Testing

The team reviewed the provisions made in the system, in compliance with RG 1.139, "Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown," to allow for normal leakage during long-term cooldown without affecting plant safety or violating the radiation limits established by 10 CFR Part 100.

Contrary to the statement in the USAR Table 5.4A-1, the licensee had not established in the TS an acceptable leakage limit at which the RHR train is to be declared inoperable. However, as per

USAR Section 18.3.4, and TS Section 6.8.4, the licensee has established a program to reduce leakage from systems outside containment that contain highly radioactive fluids. The program requires monitoring and correcting leakages identified during surveillance tests or routine plant walkdowns. This program meets the requirement of Item III.D.1.1 of NUREG-0737. The licensee stated that the existing program met the intent of RG 1.139 and the USAR table will be revised appropriately. Review of the licensee analysis for leakage requirements, Calculation AN-97-049, "Radiological Consequence of a LOCA and Available NPSH Determination Due to the Leakage from the RHR and CS Encapsulation Tanks," Revision 0, indicated that Wolf Creek can tolerate a total ECCS leakage of 2 gpm and still be within the limits established by 10 CFR Part 100, and that long-term plant cooling would not be affected. The team noted that the staff's safety evaluation report (NUREG-0881) Section 15.4.5.1 states the maximum operating leakage limit to be 1 gpm. The team reviewed licensee procedures AP 25C-001, "WCGS Leak Reduction of Primary Coolant Sources Outside of Containment;" STN BG-001, "Leakage Inspection Program of CVCS;" STN EJ-001, "Leakage Inspection Program of RHR;" and STN EM-001, "Leakage Inspection Program of SI," to determine if any of these procedures addressed a requirement for establishing leakage limits or leakage acceptance criteria to determine what is an acceptable leakage rate.

The team determined that the above procedures did not establish acceptance criteria to account for individual system leakage or cumulative ECCS leakage. The team reviewed leakage data collected for the ECCS between the eighth and the ninth refueling outages and the plant manager report for radioactivity leaks from the last refueling outage (the ninth). The team concluded that except for a few sightings of boron precipitation during plant walkdowns, no leaks existed in the ECCS and controls existed for detection and elimination of leakage. The team noted that the licensee also identified this issue. PIRs 97-3563, 97-3138 and 97-3738 were written to address the above issues. (Inspection Followup Item 50-482/97-201-02)

e. Potential for Radioactive Leakage from the RWST to Atmosphere

The team reviewed the potential for radioactive leakage from the RWST to the atmosphere during the recirculation phase of a LOCA, when radioactive water from the containment sump could enter the RWST due to leakage through the isolation valves. The team reviewed ITIP 1737, which was the licensee's evaluation in response to information notice (IN) 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere." This evaluation determined that to remain within 10 CFR Part 100 limits, Wolf Creek could tolerate a leakage of 1.8 gpm flowing to the top of the RWST and 9 gpm flowing to the bottom of the RWST. The team reviewed procedure STS-BN-206, "Borated Refueling Water Storage System Inservice Valve Test," Revision 7, for back-leakage testing of the RWST isolation valves. This procedure established the acceptance criteria for leakage through valve EJ-8717 at 10 gpm at 600 psig and 4 gpm at 20 psig for each of the eight ECCS pump suction check valves that isolate the RWST. This leakage would be lower for valve EJ-8717 at RHR system pressure but could be higher at the higher containment pressure for the pump suction check valves. The team's concern was that with the current acceptance criteria, the leakage of contaminated radioactive fluid into the RWST and to the atmosphere at high containment pressure may be greater than the limits established in ITIP 1737. The team noted that the existing analysis considered leakage into the RWST for 30 days, which may be conservative since the containment pressure drops below 6 psig within 24 hours, and the actual dose may, therefore, be lower than evaluated in ITIP 1737. This was confirmed in a preliminary analysis done by the licensee. In addition, a review of valve tests data showed no significant leakage of any consequence. Therefore, there is no immediate safety concern. The team noted that the licensee identified this issue and was taking appropriate corrective actions via PIR 97-4124.

f. Consequence of Flooding of Containment Sump Isolation Valves

For Wolf Creek, containment isolation for the RHR suction line from the containment sump is provided by only one containment isolation valve. To contain any radioactive sump fluid leaking from these valves, the valves are encapsulated in a tank. The team reviewed the design of the valve and its encapsulation to determine if leakage due to any failure or normal wear of the valve could prevent the valve from performing its safety function.

The encapsulation, or enclosure, for containment isolation valves HV-8811A & B has no automatic drain or any level measuring device in the enclosure to indicate any leakage in the control room for operator action. The possibility exists for these valves to be flooded due to leakage from normal wear of valve packings or gaskets. Because of this possibility, the valves and piping have been analyzed for mechanical integrity. Also, a review of electrical schematics E-13EJ05A and E-13BN03 showed that flooding of the valve motor operator and its circuit would not affect the valve safety function to remain open. However, flooding of the limit switch in the operator may affect the containment isolation indication for interlocked valve HV-8701A. The licensee stated that operators would depend on the indication provided by the plant computer for valve HV-8701A in case of loss of containment isolation indication. However, the team noted that none of the existing design documents or plant procedures identify the plant computer as an alternate source for containment isolation indication. The licensee issued PIR 98-0083 to address this issue.

g. Piping Design Pressure and Temperature

The team reviewed the RHR system Piping and Instrumentation Diagram (P&ID) M-12EJ01, Revision 19, System Description M-10EJ(Q), Revision 0, Piping Class Summary MS-01, Revision 40, and piping isometric drawings to verify the piping design pressure and temperature classification for the suction lines from the RCS, RWST, and containment sump; and discharge, minimum recirculation, test and tie lines to the intermediate and high head safety injection systems. The team determined that the design pressure and temperature ratings of the lines were acceptable. However, minor discrepancies in documentation of normal service ratings were noted. The licensee issued PIR 98-0002 to address this issue.

h. RHR Pump Operation in Minimum Recirculation Mode

For a small break LOCA the RHR pumps would start and operate on minimum recirculation flow for an extended duration, therefore the team verified the duration for which the pumps could run without any cooling flow to the RHR heat exchanger without being damaged.

From a review of procedures EMG E-0, "Reactor Trip or Safety Injection;" E-1, "Loss of Reactor or Secondary Coolant;" and ES-11, "Post-LOCA Cooldown and Depressurization," the team determined that the RHR pumps may be allowed to run for a duration of 2.5 hours before an operator decision is made to either shut the pump down or initiate cooling flow to the RHR heat exchanger. This period of 2.5 hours was beyond the pump manufacturer's (Pacific Pumps) time limit of 30 minutes. The team reviewed licensee calculation EJ-M-018, "RHR Pump Recirc. Operation vs. Time of Initiation of CCW flow to RHR Heat Exchanger," Revision 0, which justified this duration of 2.5 hours. This calculation essentially determined the time it would take for the temperature of the circulating water to reach 212 °F, a temperature at which the pump could start vortexing. The team noted that this calculation assumed an incorrect initial water temperature of 90 °F. As per USAR Table 3.11b, the maximum temperature in the RHR pump rooms is 104 °F. The basis for the initial water temperature used in the calculation was not readily apparent and

was not consistent with the USAR. Even with a higher initial water temperature, i.e., 104 °F, the licensee determined that sufficient margins existed in the calculation to demonstrate that the pump operation for 2.5 hours without any cooling water would not damage the pump. The licensee issued PIR 97-4150 to address the issue. The team determined that the licensee's design control measures did not ensure that the design was verified or checked adequately in accordance with Criterion III of Appendix B to 10 CFR Part 50. (Unresolved Item 50-482/97-201-03)

i. Environmental Qualification (EQ) of RHR Pump and Motor.

The team reviewed the environmental qualification of the RHR pump and motor to determine whether the as-built configuration matched the qualification documents. The team determined that pumps and motors were qualified for their application. However, the following weaknesses were identified in EQ documentation:

- The EQ files for the RHR mechanical pump and electrical motor did not provide a reference for the normal and accident environmental parameters used to justify qualification. The normal and accident radiation parameter (1.28×10^7) used for qualification was lower than the total integrated radiation dose of 1.82×10^7 provided in USAR Table 3.11(b) -2. The pump and motors were qualified to a total radiation dose of 5×10^7 rads, and hence, there was no qualification concern. The licensee issued PIR 98-0037 to address this issue.

- The current revision of the EQ file for the RHR pumps indicated that the pumps used lube oil. It was determined that the pumps did not use any lubrication oil as the pumps have carbon bearings lubricated by the pumped water. The licensee issued PIR 98-0038 to address this issue.

j. Freeze Protection for the RWST and RHR Suction Lines

The team reviewed the design documents and operating procedure for protecting the RWST tank and lines exposed to the outside environment from freezing at very low temperatures during winter (freezing could affect emergency core cooling during an accident). The team determined that the existing design and operating procedure precluded any freezing of the tank or lines. However, weakness in a 10 CFR 50.59 evaluation documentation was identified, as discussed below.

Procedure STN-GP-001, "Winterization Procedure," was revised to keep the bypass valve across the automatic control valve open to allow continuous steam flow to the RWST heaters and so prevent any freezing in the piping and assure that the RWST temperature is maintained above a nominal 50 °F when temperatures fall below 35 °F. The team reviewed the 10 CFR 50.59 evaluation done by the licensee for implementing this procedure change, CCP07251. This evaluation, 59 97-0008, stated in Section 1-a that "the temperature control valve set point is a nominal 50 °F"; and the proposed USAR change stated that," the RWST is maintained above a nominal 50 °F set point." The team verified from the licensee setpoint data base that the cut-in set point is 65 °F and the cut-out is 80°F for the temperature switches used to control the RWST temperature. The reference to the valve setpoint of 50 °F in 59 97-0008 was incorrect. The team determined that this was a documentation error and it did not affect the temperature settings. The licensee issued PIR 97-3830 to address this issue. Subsequent to the inspection the licensee revised the document to reflect the setpoint data.

E1.2.1.3 Conclusion

The team concluded that the mechanical design for the RHR system was adequate to provide the required flow and heat transfer capability to bring the plant to cold shutdown from hot shutdown conditions within 20 hours and maintain the plant in cold shutdown, and was also adequate to provide the required flow for decay heat removal during mid-loop operation. The system was adequately designed to provide the required flow to the RCS to mitigate the consequences of a small or large break LOCA, considering a 10 percent reduction in the pump flow capacity. The system provided the required NPSH for the RHR pumps when taking suction from either the RWST or the containment sump. The team had concerns regarding nonexistent or nonconservative acceptance criteria for normal system or radioactive leakage from ECCS and for leakage of radioactive sump fluid to the RWST. The team noted some nonconservative assumptions used in calculations, one of which was a Westinghouse cooldown analysis. Discrepancies were observed in a piping class summary document, EQ files, and a 10 CFR 50.59 evaluation.

E1.2.2 Electrical Design Review

E1.2.2.1 Inspection Scope

For the electrical design review, the team focused on the essential power supplies to the RHR, CCW, and support-interface systems. The following power supplies were chosen for review: emergency diesel generators, the 4160 Vac system, the 480 Vac system, the 120 Vac system, and the 125 Vdc systems. The following attributes for the above areas of review were assessed by the team: equipment sizing; regulatory and standard compliance; electrical separation; voltage drops and available voltages; protective device sizing, coordination and setpoints; controls and interlocks; operating procedures; plant modifications; surveillance tests; and design and configuration control.

The team reviewed USAR Section 8.0, TS 3/4.8, system descriptions, electrical requirements, design change packages, surveillance test requirements, and other miscellaneous electrical documents related to the design basis.

E1.2.2.2 Observations and Findings

E1.2.2.2.1 AC System Review

a. Emergency Diesel Generator (EDG)

The team reviewed the emergency diesel generator (EDG) system description, loading calculation, elementary diagrams, protective relay setpoints, and TS test requirements. A review of Drawing E-11005, "List of Loads Supplied by Emergency Diesel Generator," Revision 19, determined that the EDG loads under various postulated accident conditions remained within the EDG continuous rating and had adequate capacity margin. The team evaluated power demands for major pumps such as ESW, RHR, and CCW pumps, to verify the loads on the above drawing. The team determined that the licensee properly estimated the loads. A review of recent EDG surveillance test report STS KJ-001A showed that electrical loads, including ECCS loads, were sequenced onto the EDG within the required times and that the drop and recovery in output voltage and frequency were acceptable. The licensee's test met the TS requirements and complied with Regulatory Guide 1.9, Revision 3.

b. 4160 Vac System

The team reviewed the system description, drawings, voltage and short circuit analysis, relay setpoints, and coordination calculations, focusing on the equipment performance under worst-case voltage conditions and the equipment short circuit duty. The calculations indicated that the undervoltage setpoints were adequate for proper operation of the electrical loads and for shedding loads in the event of a loss of offsite power. The breakers' interrupting duty was found to be adequate. However, the team noted the following discrepancies.

Based on calculation XX-E-006, "AC System Analysis," Revision 3, some equipment fed from Class IE bus NG01 could experience voltage levels slightly higher than their maximum ratings with the switchyard at 105% of rated voltage. The calculation stated that this was considered a rare occurrence since switchyard voltage was not normally higher than 104 percent; however, switchyard data for 1996-1997 showed several instances where switchyard voltage had reached 105 percent for short durations. The licensee indicated that an evaluation would be done to determine the impact of this overvoltage condition (down to the 120 volt level) and calculation XX-E-006 would be revised. The licensee's preliminary evaluation indicated no operability concerns. The licensee issued PIR 98-0035 to address this issue.

A review of calculations XX-E-007, "Verification of Voltage Analysis at Wolf Creek," Revision 0, and XX-E-006, Revision 1, indicated that the short circuit contribution from the EDG was considered only for one plant scenario. An analysis is required to determine the worst-case fault current, and the consequence, when the EDG is connected to the bus in other scenarios such as when the safety buses are fed from one startup transformer. The licensee issued PIR 98-0071 to address this issue.

c. 480 Vac System

The team reviewed the system description, drawings, voltage drop and short circuit calculations for the 480 Vac unit substations and motor control centers (MCCs) focusing on the equipment's short circuit duty and the equipment's performance under minimum available voltage conditions. The review determined that adequate voltage would be provided at the terminals of all Class 1E equipment. The switchgear breakers' interrupting duty for both the instantaneous and short time delay trips was found to be adequate. The interrupting ratings of the 480 Vac molded case circuit breakers at the MCC level were also found to be adequate, except that the available short circuit current at the buses of MCCs NG01A and B for 480 Vac EF3 type breakers exceeded their interrupting rating. The licensee indicated that a design change package, PMR 03907, was replacing these breakers. A number of these breakers had already been replaced by breakers with a higher interrupting rating and the remaining EF3 type breakers will be replaced during the current fuel cycle.

d. 120 Vac Systems

The team reviewed the drawings and voltage drop calculation EB-10, "Determine Voltage Drop in MCC Control Circuits," Revision 3, for control circuits driven by control power transformers (CPTs) located at MCCs. The above calculation, which determined the maximum permissible circuit wire lengths to allow proper pickup of motor starters, was found to have used assumptions which were not conservative. For example, all auxiliary loads were erroneously assumed to be at the location of the CPT. The team picked one circuit (valve BBPV8702A circuit) which had an actual installed circuit wire length of 4228 ft for review. The team's review indicated that sufficient voltage would not have been available for the circuit if the wire length of 5374 ft allowed

in the calculation was used. The licensee's review confirmed that allowable wire length for that circuit to operate properly should be 5189 ft rather than 5374 ft. Based on the sample review, the team determined that if the maximum permissible circuit wire lengths in the calculation for other motor starter circuits (with different starters and loading configuration) were used, the potential existed that the starter coils, in some cases, might not pickup. The team raised this concern. In response the licensee performed an operability evaluation that was supported by an analysis. The licensee's analysis and operability evaluation concluded that the actual control circuit lengths installed were lower than the allowables and were acceptable and would perform their safety-related function. The team noted the licensee's calculation for allowable maximum wire length was still nonconservative and needed to be revised. The licensee issued PIR 97-4159 to revise the calculation. The team determined that the licensee's design control measures did not meet the requirements specified in Criterion III of Appendix B to 10 CFR Part 50 regarding verifying or checking the adequacy of the design. (Unresolved Item 50-482/97-201-04)

No analysis existed to show that 120 Vac feeders and control circuits are protected adequately during a fault. Also, no voltage drop analysis existed to show that all 120 volt safety-related loads had adequate voltage at their terminals. The licensee's preliminary evaluation of some sample circuits indicated that there were no operability concerns. In addition, the team noted that the licensee's electrical system modeling did not include Class 1E buses down to 120/208 volt level. Power System Branch Technical Position PSB -1, positions 3 and 4 (licensing confirmatory issue B.19), require electrical system modeling down to 120/208 volt buses, verifying the modeling by actual tests and comparing the difference between measured and modeled values with the equipment rating. The licensee was not able to demonstrate or provide documentation to show how it met the PSB-1 commitments for the 120 volt system. It was noted that the licensee met the PSB-1 requirements for voltages up to 480 volts. The licensee issued PIRs 97-4041 and 97-4032 to address the above issues. (Inspection Followup Item 50-482/97-201-05)

e. Evaluation of Plant Modifications

Three modification packages (DCP 07195, 05588 and PMR KN 84-0044) were randomly picked for evaluation, with particular emphasis on the 10 CFR 50.59 evaluation and procurement of components. The 10 CFR 50.59 evaluation conclusions were adequate, and the design changes were consistent with the design basis. However, the team had a concern with DCP 05588, which covered, in part, procurement of an overexcitation relay for the EDG. This component was installed as a safety-related item through the commercial grade dedication process conducted by the supplier (Farwell & Hendricks, Inc.). The analysis done as part of the dedication process established the qualified shelf life of the relay to be 16 years. Monitoring of degradation is required to determine the level of degradation and to establish frequency of replacement of the relay or any of its parts to maintain its qualification in accordance with Certificate of Conformance, #62152.1. Documentation of the methodologies used to meet these requirements could not be provided by the licensee during the inspection period, nor could any documentation of surveillance results. The licensee issued PIR 98-0085 to address this issue. (Inspection Followup Item 50-482/97-201-06)

f. Cable Ampacity and Short Circuit Rating

The team's review of cable ampacity calculations F02, "Cable Sizing," Revision 0, and F03, "Cable Sizing," Revision 4, revealed that cables in covered trays were given a derating based on 96% of the ampacity of cables routed in open trays (4 percent derating factor in addition to the derating factor used for cable tray fill). While this is a not in accordance with current industry

design practice (18-27 percent derating), the team could not identify a licensing or regulatory basis that would require a greater level of derating. A review of a sample of as-installed RHR system cable installations revealed that the cables were adequately sized even for the current industry design derating value.

The team also reviewed calculation F09, "Determine the minimum cable sizing based on maximum AC short circuit rating," Revision 3. The review determined that the cables were sized adequately to withstand the maximum short circuit current.

g. Protective Coordination

The relay setting tabulation, E-11023, "Relay Setting Table and Coordination Curve," Revision 4, was reviewed, and settings were verified for RHR, CCW, CS, and ESW pump motors. The relay settings were consistent with values indicated in the coordination diagram. Coordination was found to be adequate among the following: motor full load amperes (FLA) at 75 percent and 100 percent, motor locked rotor amperes (LRA) at 75 percent and 100 percent, motor thermal limit, feeder cable thermal limit, and relay characteristic curves. To ensure that changes in relay settings had been adequately transferred to the coordination diagram, the team reviewed an instantaneous trip setting that had been revised in the relay setting tabulation. It was verified that the characteristic curve on the coordination diagram had been appropriately revised to reflect the changes in settings.

h. Electrical Penetration Protection

The team reviewed the protection of electrical penetration modules and conductors that were shown in calculation A-6-W, "Thermal Capability of Electrical Penetration Assemblies (EPAs) vs Dual Short Circuit Protection to Satisfy Reg. Guide 1.63," Revision W2. Several circuits covering sizes #12 AWG, #10 AWG, #8 AWG, #6 AWG, #4 AWG, #2 AWG, 2/0 AWG, 250MCM, 350MCM, and 500MCM cables were selected for review. The team determined that the penetration and circuits were adequately protected with primary and secondary protective devices and the licensee had complied with RG. 1.63.

i. Class IE 480 Vac Molded Case and 4160 Vac Switchgear Breakers - Test and Surveillance

The team reviewed the licensee's surveillance testing and maintenance procedures on selected 480 Vac penetration molded case circuit breakers and 4160 Vac switchgear breakers for the RHR and CCW systems. The procedures reviewed were STS-MT-024 and MPE E-009Q-02. The procedures provided detailed instructions for testing and inspection of breakers in accordance with the manufacturer's instructions. No discrepancies were identified. The team noted that the testing program was previously reviewed by NRC.

j. Calculation Documentation Discrepancy

During the review of calculations for cable sizing the team noted that calculation F-03, "Cable Sizing," Revision 4, referenced calculation F-08, "Electrical - Provide Essential Service Water Pump Cable Sizing," Revision 0, which has been superseded. The licensee stated it would review calculation F-03 for any potential impact via a PIR.

E1.2.2.2.2 125 Vdc System Review

a. Electrical Penetrations

The team reviewed the electrical protection for the dc circuits directed through penetrations, as presented in calculation A-6-W, "Thermal Capability of Electrical Penetration Assemblies (EPAs) vs Dual Short Circuit Protection to Satisfy Regulatory Guide (RG.) 1.63," Revision W2, and determined that the licensee had complied with RG. 1.63. Each dc circuit routed through an electrical penetration had proper overload protection by suitably sized redundant fuses in its positive and negative circuit sides. The calculation, however, did not contain a complete listing of all safety-related dc circuits routed through electrical penetrations, including related schematic drawings, and all worst-case time-current plots of relevant dc overcurrent devices and associated thermal capability curves of dc electrical penetrations. The licensee initiated PIR 97-3910 to evaluate and resolve this issue.

b. Sizing of Class 1E Batteries

The team evaluated the sizing of the new AT&T round cell batteries in accordance with calculation NK-E-002, "Class 1E Battery Sizing," Revision 3. The new batteries were installed in March of 1996. The team was able to verify that the 4-hour load profile (load values and running times) for battery NK11 was correct except for the following errors: assumed amperage and duration of EDG field flash circuit (first attempt), exclusion of some inrush currents during 1st minute, and total value of continuous loads for entire battery discharge cycle (likewise, for NK12, the exclusion of inrush current for valve FCHV0312). The licensee issued PIRs 97-3988 and 97-4063 to address the discrepancies and to validate the portions of battery load profiles in question. (Inspection Followup Item 50-482/97-201-07)

The team also reviewed DCP 5846, which implemented the installation of the new AT&T round cell batteries in early 1996. The licensee concluded that the Technical Specifications (TS) were not affected by the change. Neither the licensee's 10 CFR 50.59 evaluation nor design change package provided any basis for its conclusions. Currently, TS Sections 4.8.2.1.e and f (battery capacity replacement criteria and battery degradation criteria) appear to be nonconservative because the batteries were sized without any aging factors and the battery performance characteristic was changed. The team determined a TS change was warranted as indicated below.

The 80 percent battery capacity replacement criterion used in the TS is in accordance with Institute of Electrical and Electronics Engineers Inc. (IEEE) 450-1975. Section 1 of IEEE 450-1975 states that battery sizing is one of several applications that are beyond that document's intended scope. In Section 6, the same IEEE standard states that "the timing of the replacement is a function of the sizing criteria utilized and the capacity margin available, compared to the load requirements." It further states that a battery should be replaced within 1 year of when its capacity deteriorates to 80 percent.

Since the sizing criterion was the basis for the replacement of the batteries, the licensee should look for the bases of the replacement criterion outside of IEEE 450. IEEE 485-1978 describes its scope as defining dc loads for generating stations and sizing the batteries to supply those loads. Section 6.2.3 of IEEE 485-1978 recommends that an aging factor of 125 percent of the expected load demand be utilized in order to meet the battery replacement criterion of 80 percent used in IEEE 450-1975, which is therefore also applicable to TS Section 4.8.2.1.e and USAR Section 8.3.2.1.2. The licensee stated that IEEE 485 is not within their licensing basis. The team noted

that calculation E-3-W, "Class 1E Battery System," Revision W0, for the original square cell batteries (Gould) was performed in accordance with IEEE 485-1983 with respect to aging, design margin, and temperature. The present battery sizing calculation (NK-E-002) also referenced IEEE 485 but specified an aging factor of 1.00. At 80 percent of its nominal rated capacity, a battery can still supply its rated 100 percent demand load if a 1.25 aging factor is used. Since an aging factor was not considered in the design of AT&T round cells, at 80 percent of rated capacity a Wolf Creek round cell battery could possibly not be capable of meeting 100 percent of the demand load. The existing design capacity margin of 25 percent as stated in USAR 8.3.2.1.2 is intended for load growth and could be used up at the battery's end of life. Further, the licensee has experienced some problems with recognizing total load on the dc buses (see Section E.1.2.3.2.e).

The vendor manual states that the round cell battery capacity under float conditions increases with age. The design change package also states that the round cell battery capacity will increase with age during the life of the battery. Neither the vendor manual nor the design change package discussed how a round cell battery would perform under discharge conditions. The team inquired about the testing performed by the manufacturer or the licensee to verify the battery degradation rate. The team noted that there is not enough test data or historical data available to statistically validate a specific performance criterion during its service life. During the inspection the licensee provided some test data from the discharge tests performed by the vendor which indicated that the cells may experience some capacity loss over time. The licensee stated that the loss would be gradual and the capacity may increase if recharged properly after the discharge. The licensee believes that the round cell batteries would degrade similar to the typical square cell batteries under discharge conditions. The team noted that how a round cell battery would degrade, whether gradually or suddenly, once degradation started was not known at the time the battery was installed and still is not known because sufficient test data is not available. The licensee failed to question whether the capacity performance requirements in the TS were exacting enough to measure adequately the performance of the round cells.

The team determined that since the battery was sized without any aging factors and the performance characteristics of the battery were changed for the reasons stated above, the TS was affected by the design change. The licensee's 10 CFR 50.59 safety evaluation failed to identify the effect on the TS. Presently, TS Sections 4.8.2.1.e and f appear to be nonconservative. The licensee presently disagrees with the team's contentions; thus this matter is being referred to the NRR staff for further review and resolution. (Inspection Followup Item 50-482/97-201-08)

The team also reviewed Section 8.3.2.2, "Battery Capacity," of the NRC safety evaluation reports (SERs) NUREGs-0881 and-0830, and determined that the licensee had failed to comply with the staff's position on battery capacity for the standardized nuclear unit power plant system (CNUPPS) plants. The licensee had sized its new AT&T round cell batteries for a 25 percent margin as stated in USAR Section 8.3.2.1.2, but not with the 50 percent margin stipulated by the NRC staff since the correct values of applicable battery sizing factors were not utilized. The staff's SER states that the TS are written assuming a 50 percent-greater-than-required capacity for each Class 1E battery. Presently, the new AT&T round cell batteries are sized with the following margins, as stated in PMR 03845, Revision 0, and battery sizing calculation NK-E-002: NK11- 32 percent; NK12- 35 percent; NK13- 65 percent; and NK14- 35 percent. However, the team found errors in the way the design capacity of the batteries was calculated in the battery sizing calculation. The licensee used an incorrect discharge rate in determining the percentage of ampere-hours required for each cell. The team calculated the design capacity margin for the NK11 battery as 23 percent, instead of 32 percent as stated in the above documents. Similar

results may be expected for other batteries. The team determined that the Class 1E batteries are not sized with appropriate design capacity margin as stated in the staff's SER and USAR Section 8.3.2.1.2. This issue is being refereed to the NRR staff for further review along with the battery TS issue mentioned in the above paragraph. (Inspection Followup Item 50-482/97-201-09)

c. DC Fault Contribution

The team reviewed calculation NK-E-003, "Class 1E 125 Vdc batteries short-circuit study," Revision 0, and determined that all dc buses and associated cabling were conservatively sized for the available short circuit currents. Fuses provided the correct overload and fault protection for the DC system distribution circuits, and the correct sizing of fuses ensured the requisite selective coordination between fuses in series when applicable. The licensee originally assumed that the AT&T round cell batteries had a lower fault contribution and did not perform an analysis before declaring them operable in early 1996. In response to the team's questions, the licensee performed a quick calculation during this inspection to verify its assumed lower fault current contribution to dc buses from the AT&T round cell batteries. In addition, the licensee has decided to clarify assumptions 3.4 and 3.6 in the above calculation both to achieve consistency and to state which motors provide a fault contribution to the total fault current under PIR 97-4063.

d. DC Load Flow/Voltage Drop

The team reviewed calculation NK-E-001, "Class 1E DC Voltage Drop," Revision 1, and determined that it adequately demonstrated that all available dc components would have sufficient voltage to properly operate, except those components for which the licensee assumed a value of 100 Vdc in the load data base of the above calculation because the vendor did not stipulate a minimum value. The team questioned the licensee about this apparent discrepancy in this data base. In response, the licensee issued PIR 97-4180 which indicated that all dc equipment procured having an operating voltage of 140 Vdc would function adequately. For those devices assumed as having a minimum operating voltage of 100 Vdc, the licensee has decided to evaluate them under PIR 97-4043. Preliminarily, the licensee determined that there were no operability concerns and that it would either analyze each circuit on a case-by-case basis or devise a generic solution applicable to all the affected circuits. Assuming 100 Vdc for devices with no specified minimum voltage is an example of an unverified assumption. The licensee failed to demonstrate the adequacy of this design. This is contrary to Criterion III of 10 CFR Part 50, Appendix B, which requires that design control measures verify or check the adequacy of a design. (Unresolved Item 50-482/97-201-10)

The team also reviewed the basis for battery NK11's minimum terminal voltage and questioned the licensee about the justification for the stated value of 107.3 Vdc for the worst-case minimum battery voltage. This voltage of 107.3 Vdc was originally derived by using the minimum operating voltage of 105 Vdc for the Class 1E inverter and then a conservative voltage drop between the inverter and the battery's terminals. The licensee initiated an effort under PIR 97-4185 to evaluate the minimum required voltage for each Class 1E battery and determined that the end voltage for each battery's discharge cycle, for either station blackout (SBO) or LOCA, would have to be raised. The licensee determined that the worst case is NK11, whose required end voltage is to be raised from 107.38 Vdc to 111.659 Vdc to operate the worst case loads 15 PS and 48 PS. No operability concerns were identified by the licensee. The team noted that the battery discharge current during a battery performance test envelops the current drawn during the 4-hour service test, and the minimum battery voltage based on analytical results is higher than 111.659 Vdc either for the SBO or LOCA conditions. The licensee failed to establish the correct design

basis information in surveillance procedure STS MT-021, and calculations NK-E-001 and NK-E-002 in regard to minimum required terminal voltages for all the batteries. This failure is contrary to requirements specified in Criterion III of 10 CFR Part 50, Appendix B, which requires that the design basis be correctly translated into specifications, drawings, procedures, and instructions and that design control measures verify or check the adequacy of a design. (Unresolved Item 50-482/97-201-11)

e. DC Load Control

The team reviewed procedure AI 05-00, "Electrical Load Growth," Revision 0, and determined that Step 6.5 categorizes all dc load growth changes (positive or negative) as significant. Step 6.6 of AI 05-006 requires the Load Growth Coordinator to revise the applicable dc calculations when significant change occurs to the dc system or during a refueling outage, whichever comes first. The licensee has not always operated in this manner in the past. For instance, DCP 5248 and PMR 4394 listed some dc calculations, including NK-E-002 for battery sizing, as affected documents. To date the calculations have not been revised to incorporate the respective dc load changes of those two design change packages, even though the systems affected by DCP 5248 were declared operable in the last refueling outage in November of 1997 and the systems affected by PMR 4394 even earlier. The licensee has issued PIR 97-4123 to address this issue. The licensee in this instance failed to adhere to the procedure on electrical load growth. This is contrary to Criterion V of 10 CFR Part 50, Appendix B, which requires that activities affecting quality shall be accomplished in accordance with instructions, procedures, or drawings. (Unresolved Item 50-482/97-201-12)

In addition, the team reviewed the current dc load data base which enables the Load Growth Coordinator to ascertain the total cumulative value of outstanding dc load changes. Step 6.3 of procedure AI 05-006 requires an Electrical Data Input Sheet for each permanent modification that has an effect on (adds, deletes, or changes a load of) the electrical distribution system. This load data is to be confirmed by an independent verifier and sent to the Load Growth Coordinator. The team ascertained the following errors in the data sheets submitted for DCP 5248. The team noted that relay 52XX (page 8 of the load data base) and relay 52YY (page 18 of the load data base) in the breaker control circuit of the battery charger were slated for deletion when in fact they were not. After the inspection, the licensee provided a revised load data base for just the load changes due to DCP 5248. The revised data showed additional loads that were not accounted for and that the total load had increased (momentary loads by about 7 amperes and continuous loads by 0.25 ampere). Thus, there are a number of discrepancies in the original data base being maintained by the licensee. The licensee is evaluating this issue under PIR 97-4125. (Second example of Unresolved Item 50-482/97-201-12)

The team also noted inconsistencies between the load growth procedure AI-05-006 and engineering screening form APF 05-002-01. Presently, engineering screening form APF 05-002-01 allows Design Engineering to evaluate electrical load changes only for increases or decreases of 10 kW or more when other screening conditions for load changes (such as addition of cables, separation problems, or increased amperage in existing cables) are not applicable. Procedure AI-05-006, Step 6.5, invokes different load change values to initiate evaluation and requires different actions. The licensee is evaluating this under PIR 97-4123. (Third example of Unresolved Item 50-482/97-201-12)

f. Surveillance Tests

The team reviewed surveillance procedures STS MT-021, "Service Test for 125 VDC Class 1E Batteries," Revision 10, and STS MT-022, "5 Year 125 VDC Discharge Battery Test," Revision 9. The licensee did not fully incorporate the requirements and acceptance limits contained in applicable design basis documents into the above surveillance procedures. The acceptance criterion in Step 6.1 of STS-MT-021 is that the test be successfully completed with no other, more definitive requirements. The licensee did not incorporate the design basis requirement contained in calculations NK-E-001 "Class 1E DC Voltage Drop," Revision 1, and NK-E-002, "Class 1E Battery Sizing," Revision 3 pertaining to whether the battery discharge current adhered to the load profile and whether final battery terminal voltage was greater than the minimum allowable value for the battery being tested. These parameters are not being verified by the licensee. Similarly, the acceptance criterion in Step 6.1 of STS MT-022 is only that the battery being tested show no signs of degradation with no details on how to successfully complete the test. IEEE 450-1975 in Section 5.4.1(2) requires that a constant discharge rate be maintained until battery terminal voltage falls to a value equal to the minimum specified average voltage per cell (1.75 Vdc per Section 8.6 of STS MT-022, or 105 Vdc for 60 cells). A nondetectable failure in the load bank would allow a battery's capacity to be incorrectly evaluated as adequate, still meeting the TS requirements and needing no corrective actions. However, battery terminal voltage or discharge current could deviate from their acceptable test ranges without being detected by technicians. As an example, in the latest capacity test for NK11, completed in the last refueling outage, the decreasing current was not detected by technicians or reviewers (see the following paragraph for more details). In addition, Step 4.7 of STS MT-021 and Step 4.11 of STS MT-022 are somewhat unclear about corrective actions to be taken for test deviations or for nonadherance to the less than thorough acceptance criteria in these procedures. The licensee is evaluating these issues under PIR 97-3941 for STS-MT-022 and PIR 97-3989 for STS-MT-021. These surveillance test procedures did not meet the test control measures specified in Criterion XI of 10 CFR Part 50, Appendix B, which require tests to be conducted in accordance with written test procedures that incorporate requirements and acceptance limits contained in applicable design documents. (Unresolved Item 50-482/97-201-13)

The licensee performed a capacity test for Class 1E battery NK11 on November 11, 1997, in accordance with surveillance procedure STS-MT-022. The load bank failed with approximately 20 minutes remaining in the test and before the battery terminal voltage reached a final voltage of 105 Vdc. The licensee performed an evaluation of the results of the test and decided to terminate the test. The minimum value of available battery capacity as verified by actual test was 95 percent; post-test analysis put it at 104 percent, which satisfied the TS requirement. The inspection team reviewed the results of the test and determined that the battery discharge current gradually decreased for about the last 2 hours of the test. In doing the capacity test, the licensee did not follow IEEE 450, which requires that battery discharge current be maintained constant during the test and the test be continued until the final voltage, typically 105 Vdc for a 60 cell battery, is reached. The licensee did not notice the decreasing battery current until questioned by the team and subsequently during the inspection had to determine by analysis what the battery capacity would have been if the test had been completed. The actual test value remained at 95 percent, but the proposed analytical value, based on an ongoing analysis at the time inspection was completed, decreased to 100 percent from the 104 percent stated above. The licensee failed to consider declining battery discharge current during its initial analytical determination of the expected capacity of the NK11 battery and failed to take appropriate corrective actions. The licensee is evaluating this issue under PIR 97-3941. It appears that the licensee took improper or incorrect corrective actions because the analysis for battery capacity did not consider the decreasing battery current. The licensee's corrective action measures did

not meet the requirements specified in Criterion XVI of 10 CFR Part 50, Appendix B. (Unresolved Item 50-482/97-201-14)

g. Fuse Control

The team reviewed the licensee's fuse control program. Procedure AP 03A-001, "Fuse Verification and Control," Revision 1, governs the licensee's program for controlling both ac and dc fuses. It is supplemented by the fuse list document, WCRE-08. The program seeks to design and field-verify all safety-related fuses by the end of 1998 or during the next refueling outage. This effort is more than 90 percent complete. The team selected a sample of installed dc fuses and of fuses depicted on electrical schemes. The licensee was able to demonstrate that all fuses in the sample were correctly assigned and labeled in the present fuse list.

h. Battery Charger

The team noted the following discrepancies in a 10 CFR 50.59 evaluation and the design basis for the Class 1E battery chargers:

1. The 10 CFR 50.59 evaluation for modification DCP 5248 (swing battery charger addition) stated that there would be no changes in the operating parameters at Wolf Creek. The team noted that this statement is not true because there is a small increase in operating parameters (amperes and kilowatts) for both ac and dc buses due to additional loads from the new swing charger. The licensee's analyses adequately supported the design. But the safety evaluation did not refer to these analyses or discuss the impact on operating parameters. The team determined that the licensee's safety evaluation documentation was weak. The team also determined that this problem did not have an adverse effect on the results of the 10 CFR 50.59 evaluation.

2. Load data tables on drawings E-11NG01 and E-11NG02 show an ac input current value of 59 amps when the dc output of the train A Class 1E charger is 300 amps. Actual ac input current, taking into account power factor and inefficiency, is 81 amps. The licensee issued PIR 97-4044 to address this issue.

3. Procedure AP 05-002, Step 6.4.5.3, requires that calculations affected by a design modification be listed as affected documents in that design change. However, calculation NK-EW-001 was not listed as an affected document in DCP 5946 (design change for addition of swing battery chargers) until its absence was pointed out to the licensee by the team. It was listed as a reference document instead. The licensee took prompt measures to address this issue.

E1.2.2.3 Conclusion

The team concluded that generally the essential power supplies for the RHR, CCW, and support systems were capable of performing their safety functions as required by their design bases. The team identified deficiencies in surveillance testing and sizing of batteries and load growth control. The majority of electrical calculations were adequate but sufficient errors, nonconservative assumptions, and omissions have been identified to warrant a review of all critical electrical calculations to reverify that their design basis is accurate and consistently applied. In some cases analysis did not exist to support the design bases. The team identified weaknesses in some 10 CFR 50.59 evaluations and design changes.

E1.2.3 Instrumentation and Controls Design Review

E1.2.3.1 Inspection Scope

The scope of the instrumentation and controls (I&C) design assessment was to review RHR system design documents such as the USAR, TS, system descriptions, setpoint documents, setpoint calculations, instrument loop uncertainty calculations, specifications, maintenance, surveillance, and operating procedures, design drawings, modification packages, and miscellaneous I&C documents.

E1.2.3.2 Observations and Findings

The system design documents reviewed by the team were consistent with the design bases except for the items identified in the following subsections.

a. RWST Level Instrumentation

The team reviewed the RWST level instrumentation design with respect to RG 1.105, "Instrument Setpoints," Revision 1.

TS Section 3/4.5.5 requires a minimum contained RWST volume of 394,000 gallons with a 2400 to 2500 PPM boron concentration for the ECCS function. Four redundant level instrument loops (LT-930 through LT-933) provide input for indication, initiate RHR pump suction switchover from the RWST to the containment recirculation sump on low-low level, and alarm on low level to signal that the RWST is approaching the TS limit. Level setpoint bases for the RWST are provided under USAR Figure 6.3-7 and Bechtel calculation BN-20, "RWST Setpoints," Revision 1, as follows:

Setpoint Function	Contained Volume	Height
HI alarm	413,000 gal.	529"
LO alarm	400,000 gal.	513"
LO-LO alarm & switchover	236,200 gal.*	208"
Empty alarm	54,600 gal.	53"

*Minimum volume required for RHR suction switchover

Surveillance procedures STS IC-508A, "RWST Level Transmitter Calibration," Revision 4, and STS IC-508B, "Calibration of RWST Level Instrumentation," Revision 6, and the setpoint document translated the above values to 99.8 percent, 96.88 percent, and 36 percent for high, low, and low-low level setpoints, respectively, using the level transmitter taps (located 24" above tank bottom) as the zero reference point. The team also reviewed calculation SA-90-056, "Reactor Protection System ESFAS Channel Error Allowances," Revision 0, which calculated the RWST level instrument loop uncertainty. Based on the review, the team found the following discrepancies:

1. Calculation SA-90-056 did not consider density variation due to temperature and boron concentration in determining the RWST level instrument loop uncertainty. As a result, previously calculated instrument inaccuracies were incorrect. This could affect alarm

setpoints, the RHR suction switchover setpoint, and the control room level indicators that maintain Technical-Specification-required inventory. The calculation did not fully comply with RG 1.105, Section C.4, which requires consideration of environmental effects on the instrument setpoint determination. The new estimated inaccuracies had the following impact:

Control room indication (RG 1.97 Type A variable) - The actual RWST level could be 2.93 percent less than the indicated value; therefore, an indicated minimum Technical Specification value of 94 percent would actually be 91.07 percent, corresponding to 383,962 gallons (the TS value is 394,000 gallons). This variable is monitored in the control room per procedure STS CR-001, "Shift Log for Modes 1, 2 and 3," Revision 35, and SDTS CR-002, "Shift Log for Modes 4, 5 and 6," Revision 22. The team noted that these procedures do not include any correction for instrument loop inaccuracy. The licensee gave the team a copy of a letter sent to NRC (SLNRC 84-0089, dated May 31,1984) justifying the use of indicated readings (without regard for instrument uncertainties) to satisfy TS surveillance requirements. The licensee could not find documentation of the NRC's acceptance of the licensee's position. However, preliminary evaluation by the licensee indicated that there is adequate margin in the NPSH analysis to compensate for level indication inaccuracies up to 17 percent.

Low-low level switchover setpoint - With an estimated inaccuracy of 3.24 percent, the switchover point would be reduced from 36 percent to 33.89 percent, corresponding to a tank volume of 154,657 gallons. This would reduce the volume available for injection between the minimum indication level of 91.07 percent and the corrected swapover setpoint of 33.89 percent, which is 383962 - 154657 = 229,305 gallons. This is less than the required volume of 236,200 gallons, as specified in USAR Fig. 6.3-7 and calculation BN-20. On the basis of a preliminary evaluation by the licensee, the team considered this reduced injection volume did not impact the pump NPSH.

Low alarm setpoint - With an estimated inaccuracy of 2.51 percent, the low level alarm could be as low as 94.57 percent, which is very close to the minimum TS reading of 94 percent. This condition would reduce the margin for operator response in case of an actual leak in the RWST.

Empty Alarm - As a result of the transmitter error and [illegible]ble inaccuracy in the instrument loop, it is estimated that the "empty" alarm could drift 14" below the existing setpoint of 53". This level is approximately 3" above the RWST suction pipe. The effect would be a late alarm, which would reduce the margin for operator response to protect pumps that are taking suction from the RWST from loss of NPSH.

2. Calculation BN-20 had assumed instrument inaccuracies of 1 percent for bistables and 3 percent for total loop error to establish the existing RWST level setpoints. The team was unable to find uncertainty calculations supporting these values.

The licensee's preliminary evaluation indicated that the above issues did not constitute an operability concern. The licensee issued PIR 97-3974 to address the instrument setpoint and indication inaccuracies. The team determined that the requirements defined in Criterion III of Appendix B to 10 CFR Part 50, which requires that the design basis be correctly translated into specifications, drawings, procedures, and instructions and also verifying or checking the adequacy of the design were not followed for the RWST level instrumentation design. (Unresolved Item 50-482/97-201-15)

b. Seismic Qualification of the RHR Pump Suction Pressure Indicators

A review of P&ID M12EJ01, Revision 15, showed the RHR pump suction pressure gauges PI-601 and PI-602 as normally valved open. The Q-List shows these instruments as safety-related for pressure boundary only. These instruments are the original equipment furnished by Westinghouse as commercial grade items. They were later qualified through engineering judgment (ref. Westinghouse document RCS/CIEI (89)-299, dated 07/10/89) and commercial grade dedication (ref. Package 028-P0015, Rev. 0). Based on review of the commercial grade dedication package, pressure integrity was verified through hydrostatic testing but there was no verification to ensure that the pressure boundary is maintained during a seismic event. To satisfy IEEE 344-1975, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," and RG 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power plants," Revision 1, both of which the licensee has committed to, the instruments should also be seismically qualified for pressure integrity. However, the licensee had not done an adequately documented analysis to establish the seismic qualification of the installed units. The licensee provided a seismic qualification report, Farwell & Hendricks Report 50089.6, for a pressure gauge of similar make and model to demonstrate that the installed gauges are qualifiable, but in order to validate this report for the pressure gauges that were furnished by Westinghouse, a similarity analysis was required. Subsequent to the inspection, the licensee issued calculation XX-F010 to establish the qualification. The team did not review this calculation. The team determined that the licensee's design control measures did not verify or check adequacy of the pressure gauge design in accordance with Criterion III of Appendix B to 10 CFR Part 50. (Unresolved Item 50-482/97-201-16)

c. RHR Instrument Loop Accuracy and Setpoint Calculations

The team reviewed the licensee's setpoint methodology, uncertainty calculations, and related calculations for various RHR instrument loops to verify that adequate tolerance for instrument errors had been incorporated in the design. The team reviewed documents CWS-SNP-470C, "SNUPPS Flow Switch Setpoints"; M-74[illegible]J0025-W35, "Precautions, Limitations and Setpoints for Nuclear Steam Supply Systems," Revision 4; J2A02, "Accuracy, Foxboro Bistable," Revision 0; J2F01, "Accuracy of Standard Orifice Plates," Revision 0; AN 96-074, "RWST Water Level Necessary to Supply Adequate NPSH for the ECCS Pumps," Revision 0; XX-E003, "RHRS-RCS-ISO-VLV Open Setpoint," Revision 0; J-K-GEN, "Instrument Loop Uncertainty Estimates," Revision 0; and J1GEN, "Instrument Loop Uncertainty Estimates," Revision 1. The team's review determined that the calculations adequately demonstrated loop accuracies and setpoints.

d. System Modifications

The team reviewed five I&C modification packages for the RHR and RWST systems:

1. PMR-02820, "Containment Sump Level Indicator Scale Replacement"
2. PMR-03142, "Addition of Flow Indicators for EJ System"
3. CCP-05804, "Containment Sump Level Indication Modification"
4. PMR-03529, "Deletion of RWST Level Indicator BNLIS0001"
5. PMR-04637, "RHR Heat Exchanger Outlet Valve Nitrogen Backup"

Based on the review, the team concluded that the design, 10 CFR 50.59 evaluation, and document closeout were performed adequately for these modifications. However, the team noted that 10 CFR 50.59 evaluation documentation could be improved to provide sufficient details and basis for an independent reviewer to verify the change without performing an in-depth

review of the design change package. The licensee had recently revised its procedure to emphasize that sufficient details and basis be supplied in 10 CFR 50.59 evaluations.

E1.2.3.3 Conclusions

The I&C design for the RHR system and the interfacing portion of the RWST was considered adequate. The team was concerned with the level instrument loop uncertainty determination, which failed to consider density changes due to environmental effects on the RWST. Unverified instrument accuracies were used in the RWST level setpoint determination and incomplete seismic qualification documentation was observed for the safety-related pressure gauges.

E1.2.4 System Interfaces

E1.2.4.1 Inspection Scope

The team inspected mechanical aspects of the emergency diesel generators (EDGs), the essential service water (ESW) system, and the emergency core cooling system (ECCS) area and battery room coolers to ensure that these systems properly supported functioning of the RHR and CCW systems.

The review of interfacing system attributes involved the USAR, TS, calculations, drawings, test procedures and results, and licensing commitments. System walkdowns and discussions with licensee personnel were also conducted.

E1.2.4.2 Observations and Findings

a. Emergency Diesel Generators (EDGs)

The team performed a walkdown of the "B" EDG room, the fuel oil system and the combustion/ventilation air intake/exhaust system and found no material condition or physical configuration problems. Sampling procedure STS CH-015 for the fuel oil tanks and fuel oil shipments was reviewed and found to meet the requirements of TS Sections 4.8.1.1.2.d & e. Surveillance test procedures STS JE-003A and -004A were also reviewed and found to conform with the periodic sampling and draining of water from the fuel oil storage tanks required by TS Sections 4.8.1.1.2.b & c.

The team verified that the missile protection design of the EDG ventilation/combustion air intake structures was in accordance with USAR Section 9.4.7.2.3 by reviewing calculation 06-05-F and structural drawings M-1G052, M-1G054, C-1C5311, and C-1C5904.

To verify that adequate cooling and combustion air are available to the EDGs, the team reviewed USAR Section 9.4.7; calculation GM-320, "Diesel Generator Building HVAC - Required Air Flow," Revision 0; and EER 90-GM-02, Revision 2. The team found that EDG combustion/ventilation air sizing was adequate and that the system was designed such that combustion air is available even if the ventilation supply fan and system dampers fail.

EDG room design basis temperature was verified by reviewing calculation GM-M-002, "Diesel Generator Building Minimum Room Temperature," Revision 1.

The team noted that a portion of the EDG exhaust stacks was exposed to the plant exterior environment. The stacks were therefore reviewed for protection against design basis events.

During this review it was determined that the stacks had not been analyzed for design basis wind loads. A close review of the Wolf Creek licensing basis (USAR Sections 3.3, 3.5.2.6, and 9.5.8 and the corresponding sections of the SER (NUREG-0881)) indicated that this analysis was not specifically required; however, the licensee agreed that it would be prudent engineering design practice to include wind loads in the current analysis, which is found in calculation P-311. Preliminary analysis indicated that the existing support structures were adequate for design basis wind loading. PIR 98-0060 was issued to track this item to closure.

Electrical aspects of the EDGs inspected are discussed in Section E1.2.2.2.1

b. Essential Service Water

The team verified the ESW system flow and heat transfer capabilities during normal plant shutdown and accident conditions. The team's review is discussed in Sections E1.2.1.2 and E1.3.1.2. The team determined that the ESW system could provide the required flow of 7150 gpm at an ultimate heat sink (UHS) worst case temperature of 95°F to achieve and maintain safe shutdown conditions. To verify the pump has adequate NPSH, the UHS minimum level stated in calculation EF-M-014, "UHS Thermal Analysis - Review for Power Rerate," Revision 1, was reviewed against the ESW pump submergence requirement given on ESW pump curves M-089-K043-02 and -03, dated August 8, 1979, and July 26, 1979, respectively. The review determined that UHS minimum level was greater than the required level by a small margin (about 4 inches), and was found to be acceptable.

The team verified that the missile protection design was in accordance with USAR Section 9.4.8.2.3 by reviewing calculation K-20-05-F and structural drawings M-KG080, M-KG081, C-KC304, C-KC305, C-KC306, and C-KC309.

The team performed walkdowns of the ESW intake structure and ESW/SW distribution area at the 1974' elevation of the control building and found no material condition or physical configuration problems. Both ESW trains were found to share space in a common room at elevation 1974' of the control building. This room also contained non-safety related piping and equipment. For safety-related ESW equipment, the team checked compliance with fire separation, seismic II/I and flood- protection criteria and found it adequate.

ESW self-assessment report SEL 96-055 was reviewed by the team. This self-assessment, performed during the first half of 1997, was found to be adequate and appeared effective in initiating both system-specific and programmatic changes. Several generic design basis and licensing basis issues were identified by the licensee during the ESW self-assessment. Not all of the issues identified by the self-assessment had been closed out at the time of this inspection.

c. ECCS Area and Battery Room Coolers

The team reviewed USAR Section 9.4.3 and capacity calculations for the ECCS area and battery room coolers to determine whether the coolers had the capability to maintain temperatures below maximum design. For the ECCS areas the team reviewed calculations GL-04-W, "RHR Pump Rooms 1109 and 1111, Heat Loads," Revision 1; GL-03-W, "Auxiliary Building HVAC," Revision W-1; and GD-234, "Essential Service Water Pumphouse, Cooling and Heating Requirements," Revision 1. To ensure adequate capacity for the battery rooms, the team reviewed calculations GK-M-001, "Safety Related Control Room Building HVAC Capabilities During Accident Conditions," Revision 2; and GK-M-004, "Loss of Ventilation During Normal Operating Conditions

- Battery and SWBD Rooms - Control Building," Revision 0. The team concluded that the coolers could maintain the design basis temperatures under the worst-case ESW temperature.

E1.2.4.3 Conclusions

Interfacing system attributes reviewed were found to adequately support RHR and CCW system design basis functions.

E1.2.5 System Walkdown

E1.2.5.1 Inspection Scope

The team conducted a walkdown of the RHR system and the plant areas, including the RHR pump rooms, the RHR heat exchanger rooms, the RWST, the control room and penetration area, the switchgear rooms, the battery and inverter rooms, and the cable spreading room. The team focused on comparing system configuration to the design basis documents and the USAR. The team also looked closely at equipment condition, area cleanliness, tagging, and the means used to avoid potential hazards such as missiles, flooding, fire, and pipe rupture.

E1.2.5.2 Observations and Findings

The team determined that the overall material condition of the plant areas was good. The equipment sampled matched the design documents. However, the team identified the following issues.

a. RWST Valve House Room

During a walkdown of the RWST valve house room, 9102, the team observed that the auxiliary steam line used to supply the heater coils wrapped around the RWST was not seismically supported. Two level transmitters, LT-930 and-931, were close to the auxiliary steam line and a rupture in the steam line could potentially affect the function of these transmitters. These transmitters are identified in USAR Section 7.4.1 as being required for plant safe shutdown. Contrary to the USAR Section 3.6.2.1.2.3, no hazard analysis existed that determined the effect of a failure of the steam line on the RWST system. The team was concerned that failure of this steam line, which could go undetected for 6-8 hours until detected during operator rounds (procedure CKL-ZL-001, "Auxiliary Building Reading Sheets"), could affect plant safe shutdown.

The licensee issued PIR 97-3896 to address the team's concern. The licensee performed a hazard analysis during the inspection and determined that a failure of the non-safety-related auxiliary steam piping would not affect plant safe shutdown. This analysis also determined that the RWST level transmitters, currently identified in USAR Section 7.4.1 as being required for plant safe shutdown, were not listed in USAR Table 3.11(b)-3 as being required for cold or hot shutdown. The licensee also determined that emergency boration procedure OFN BG-009 did not reference use of the RWST level parameter as required for the proper and rapid insertion of negative reactivity to achieve plant shutdown. The licensee has determined that the RWST level transmitters are not required to bring the plant to a safe shutdown and PIR 97-3958 has been initiated to delete reference to the RWST level transmitters in USAR Section 7.4.1. The hazard analysis was reviewed by the team and found acceptable. The team determined that the above issue represents a weakness in the licensee's design and configuration control process.

b. Control Room RG 1.97 Instrumentation

The RHR and RWST instrumentation on control room panels 017, 018, 019, and 020 was verified to comply with RG 1.97 and with USAR Appendix 7A, which documents the licensee's commitment to RG 1.97. Per the RG, special identification tags are required for Type A, B, and C variables, Category 1 & 2 indicating and recording instruments in the main control room. Special identification is not required for Type D Category 2 instruments. During the walkdown the team noted that RWST level indicators LI-930 through LI-933 and RCS pressure indicators PI-934 through PI-937, classified as Type A variables, do not have unique markings to identify them as RG 1.97 instruments, however, the instrument nameplates are color coded according to their engineered safety feature (ESF) grouping and separation, which the licensee considers a method of identification for post-accident use. USAR Appendix 7A, Section 7A.1, which states "strict compliance to the many prescriptive recommendations [of RG 1.97] is not provided in all cases," appears to be the basis for not identifying RG 1.97 instrumentation by special tags. Based on the team's review, the existing RG 1.97 design for the RHR and RWST is considered adequate.

c. Equipment Tagging

The inspection team noted missing tags for RHR pumps and motors PEJ01A/B and DPEJ01A/B and sump pump level switches LS0021, LSH0008, and LSH0022. The licensee took prompt measures to correct this problem.

d. 10 CFR 50.59 Evaluation for RCS Drain Down Procedure.

A walkdown of the RHR heat exchanger rooms was performed during the refueling outage and the inspection team observed that nitrogen bottles were temporarily installed inside the rooms. The bottles were tied to support steel by #9 wire. It appeared that they were not seismically restrained and the condition could pose a potential missile hazard. Those nitrogen bottles were installed in accordance with procedure GEN 00-007, "RCS Drain Down Procedure," Revision 19, to provide backup air for the RHR heat exchanger outlet valve operators (EJHCV-606 and 607) during the refueling outage.

Based on a review of this procedure and its 10 CFR 50.59 screening (GEN 00-007 19, No.59, approved 9/9/94), the team noted that the 10 CFR 50.59 screening did not check the screening question "yes" to generate a new 10 CFR 50.59 safety evaluation or develop guidance on seismically restraining the nitrogen bottles. Before the establishment of this procedure, backup air and nitrogen bottles were installed through the temporary plant modification process. The team reviewed those temporary modification packages and their 10 CFR 50.59 evaluations and found that the bottles needed to be restrained by chains, wire rope, or tube frame stanchions instead of the #9 wire that was used. It appears that the seismic restraining requirements for the nitrogen bottles under the temporary modification were not adequately translated during the development of the RCS drain down procedure or in the 10 CFR 50.59 screening review. The licensee's 10 CFR 50.59 screening review failed to perform a safety evaluation as required by 10 CFR 50.59. Since the bottles were removed at the end of the outage, this condition did not constitute an operability concern. The licensee issued PIR 97-3961 to initiate a revision to procedure GEN 00-007 and to generate a 10 CFR 50.59 safety evaluation. In addition to Criterion III of Appendix B to 10 CFR Part 50, the requirements of 10 CFR 50.59 were apparently not met. (Unresolved item 50-482/97-201-17)

E1.2.5.3 Conclusions

In general, the RHR system design observed during walkdown was consistent with the design basis requirements. However, the team identified several issues: the absence of a hazard evaluation for the auxiliary steam line break in the RWST room, missing tags on some equipment, failure to perform a 10 CFR 50.59 safety evaluation and lack of procedural guidance for the nitrogen bottle installation in the RHR pump room.

E1.3 Component Cooling Water (CCW) System

E1.3.1 Mechanical Design Review

E1.3.1.1 Inspection Scope

The team evaluated the mechanical aspects of the CCW system to determine its ability to perform the design duty and safety functions during normal power operation and accident conditions. The evaluation included a review of the system descriptions, USAR, TS, drawings, calculations, modifications, operating and surveillance testing procedures and test records, information notices, generic letters and environmental qualification (EQ) files.

E1.3.1.2 Observations and Findings

a. System Flow and Heat Removal Capability

The team reviewed the following calculations:

1. EG-06-W, "Component Cooling Water System Calculation," Revision W-3

2. EG-09-W, "Tube Plugging for CCW Heat Exchangers EEG01A/B Max. CCW Temperature-LOCA," Revision 0

3. EF-10-W, "Essential Service Water Flows at 90°F," Revision 1

4. EG-02-W, "Component Cooling Water Pumps PEG01A, C & D Performance," Revision 0

5. EG-11, Component Cooling Water Heat Exchanger and By-Pass Flow," Revision 0

6. EG-11-W "Component Cooling Water Heat Exchanger and Bypass Pressure Drop Evaluation," Revision 0

7. EG-18, CCW Circulation Time via RHR HX," Revision 0

8. EG-27, "Effect of Diesel Generator Frequency Degradation on CCW Pump Operation," Revision 0

9. SA-92-006, "Updated Heat Rejection Rate to the UHS," Revision 1

10. SA-90-042, "Heat Rejection to the Ultimate Heat Sink During LOCA," Revision 0

11. SA-91-085, "CONTEMPT-LT Component Cooling Water (CCW) Heat Exchanger," Revision 0

12. SA-89-017, "Evaluation of CCW & RHR Heat Exchanger Performance for the Extended Fuel Operating Cycle (18 Months)," Revision 0

The above calculations documented that adequate flow and heat removal capability were provided for various plant operating modes and postulated design basis events. However, the team identified the following discrepancies:

1. USAR Tables 9.2-9, 10, and 11 list flow of 40 gpm to a removed reverse osmosis unit. The flow is maintained in the piping to prevent corrosion. This flow is not accounted for in calculation EG-06-W. The reverse osmosis unit was removed and the changes to the USAR were made as part of DCP 03406. Calculation EG-06-W was overlooked. The licensee initiated PIR 97-3857 to resolve this flow discrepancy.

2. USAR Tables 9.2-9, 10 and 11 and calculation EG-06-W Tables IA, IB and IC do not agree on total flow and heat load duty and some minor discrepancies exist on individual heat loads. For example, the total CCW flow and heat load during normal operation are given in the USAR as 9,974 gpm and 75.15 X 10^6 BTU/hr, respectively. Calculation EG-06-W gives this flow and heat load as 10,011 gpm and 8[illegible].32 X 10^6 BTU/hr, respectively. The licensee initiated PIRs 97-1341 and 97-3983 to resolve this item.

3. Calculation SA-89-017, "Evaluation of CCW & RHR Heat Exchanger Performance for the Extended Fuel Operating Cycle (18 Months)," Revision 0, determined that CCW temperature reaches 126°F. Calculation EG-06-W Table IC is based on a CCW temperature of 120°F. The CCW system has been analyzed for a temperature of 130°F; therefore this is only a documentation discrepancy. The licensee initiated PIR 97-4052 to resolve this item.

b. Pump Net Positive Suction Head (NPSH) and System Transients

Calculations EG-5, "Component Cooling Water System," Revision 0; EG-10, "Calculation of Available NPSH for CCW Pump," Revision 0 ; and EG-M-016, "Time Delay for Isolation of CCW Flow to RCP Thermal Barriers," Revision 0, determined adequate NPSH for CCW pumps. The licensee also properly evaluated the system transients in calculations EG-12, "Component Cooling Water System Pipe Break," Revision 0, and EG-24, "CCW Nuclear Aux. Component Train Switchover Single Valve Failure Analysis," Revision 1. However, the team noted nonconservative assumptions and errors in some of the calculations as follows:

1. In calculation EG-5, the licensee used the height of water in the CCW surge tank at the ½ level to determine that adequate NPSH is available at the CCW pump. This was nonconservative because the CCW surge tank could be at the lowest normal level. A preliminary, conservative analysis was provided which showed adequate CCW pump NPSH is provided at the lowest normal level. The licensee initiated PIR 97-3837 to resolve this item. After the inspection, the licensee revised calculation EG-5, confirming that adequate CCW pump NPSH was available if the surge tank level was at the bottom of the tank. The NPSH available at the CCW pumps is 37 ft as compared to 43 ft in the original analysis. The required NPSH is 12 ft.

2. In calculation EG-M-016, the licensee determined that a 10 second time delay for isolation of the CCW high flow from the RCP thermal barriers was acceptable. A smaller break that results in a flow lower than the common header flow element setpoint (210 gpm), with no action taken until the surge tank is at the high level alarm, was not evaluated. A preliminary, conservative analysis was provided which showed the small break has worse consequences than the large break; however, the radiological consequences are bounded by a chemical and volume control system

(CVCS) break with a loss of 8460 gallons (analyzed in USAR Section 15.6.2). The licensee initiated PIR 97-3837 to resolve this item. Subsequent to the inspection, the licensee revised calculation EG-M-016, confirming the results of the preliminary analysis. The loss of reactor coolant outside of containment is 1178 gallons, compared to 202 gallons in the original analysis.

3. In calculation EG-12, the licensee assumed that the isolation valve (for non-seismic CCW piping to Radwaste) closure time was linear with respect to valve position and then used this assumption to determine that the average flow will be half the starting flow. This is an incorrect extrapolation of the original assumption. Attachment 1 to the calculation showed that the valve flow coefficient (Cv) is not linear with respect to valve position. Additionally, the change in valve Cv should be added to the total system resistance to determine the effect on flow. The team noted that instrument error was not accounted for in the calculation. A preliminary, conservative analysis was provided by the licensee which showed adequate surge tank level (i.e., CCW pump NPSH) is maintained. The licensee initiated PIR 97-3837 to resolve this item. Subsequent to the inspection, the licensee revised calculation EG-12, confirming the results of the preliminary analysis. The remaining inventory in the surge tank is 636 gallons, compared to 1278 gallons in the original analysis. In addition, in the above calculation the licensee assumed the guillotine break as the worst case. A smaller break that results in a flow lower than the flow element setpoint (4500 gpm) and does not initiate a valve closure signal until the surge tank is at the low-low level trip setpoint was not evaluated. The above PIR also addressed this issue. The revised calculation with the smaller break determined that more water was lost in this scenario and that the remaining inventory in the surge tank is 5 gallons. Adequate NPSH is still maintained for the CCW Pump.

c. Motor-Operated Valve Design

The team evaluated the adequacy of CCW containment isolation valves to meet their design basis requirement by reviewing motor-operated valve (MOV) design document E-025-00007(Q)-W10, "MOV Design Configuration Document," Revision 9W, and calculation EG-M-007, "Motor Operated Valve Bounding Conditions Determination," Revision 2. The team noted that in pages 218 and 230 of design document E-025-00007(Q)-W10 the differential pressure to close CCW valves EG-HV-062/132 is identified as 1120 psi. These valves are required to close against reactor coolant pressure resulting from a RCP thermal barrier break. A nonconservative assumption, that the downstream pressure was the average of the pressure before and after closing the MOV (1130 psig), was used and resulted in a lower than actual differential pressure. The team determined that the downstream pressure would be 22 psig based on the static head from the CCW surge tank and the differential pressure for the valves to close would be 2228 psid. The licensee performed a review and determined that the only other valves affected are the BB-HV-0013/14/15/16 valves. These valves are closed on limit switch control. There is no operability concern as the licensee provided a preliminary analysis which showed that the motor operators have sufficient thrust to close the valves against the required differential pressure. The licensee initiated PIR 97-4054 to resolve this item. The licensee's design control measures did not ensure that motor operated valve design was adequately verified or checked in accordance with Criterion III of 10 CFR Part 50, Appendix B. (Unresolved Item 50-482/97-201-18)

d. Other Discrepancies in Calculations

The team noted the following weaknesses in calculations:

1. Calculation EG-13, "Component Cooling Water Radiation Monitor Flow Orifice Calc.," Revision 1, analyzes the loss of CCW from a break in nonseismic piping to the radiation monitors RE9 and

RE10. This calculation uses an acceptance criteria of 30 minutes available for operator action to isolate the leak after receiving a low level alarm. No basis is provided for the 30 minute criteria in the calculation. In response to the team's question the licensee performed an analysis which showed that the operators would have approximately one hour to isolate the leak which is adequate.

2. The licensee could not establish the basis for the required capacity of CCW heat exchanger relief valves EG-V-027 and EG-V-052 or for the set pressure of CCW surge tank relief valves EG-V-159 and EG-V-170. The CCW surge tank relief valves provide overpressure protection for the CCW system. During the inspection the licensee issued calculations EG-M-030 and EG-M-031, which showed that these valves have adequate capacity and proper setpoints. The team noted that the surge tank vacuum relief valve was sized adequately in accordance with calculation EG-04-W, "Determine Acceptable Surge Tank Vacuum," Revision 0.

e. Modifications and Safety Evaluations

The team reviewed seven modification packages and nine unreviewed safety question determination (USQD) evaluations to ensure that the system design basis was being maintained and that no unreviewed safety questions existed. The team determined that the system design basis was being maintained and that no unreviewed safety questions existed. However, incomplete design change and 10 CFR 50.59 (USQD) reviews had been performed in the following instances. The team also determined that the licensee's 10 CFR 50.59 evaluations generally lacked documentation of sufficient justification.

Design change PMR 4380, "CCW Temperature Change," Revision 2, and its associated safety evaluation, 59 92-0216, Revision 0, was reviewed. This PMR changed the allowable CCW minimum temperature from 60°F to 32°F. During normal plant operation, only one train of CCW is in operation. The redundant CCW train is on standby with no flow through the CCW side. However, there is always flow on the ESW side. Upon initiation of a safety injection signal, CCW pumps in the standby train start operating, thereby circulating cold water through various CCW components. The operating CCW train also experiences cold temperatures, as the nonessential heat loads are dropped off the system and flow through the CCW Heat Exchanger is no longer being regulated because the air supply to the bypass flow control valve is not safety related. The valve is designed to fail closed. The design change assumed, conservatively, that CCW is at the same temperature as the lake water, 32 °F. The following items were not adequately addressed either in the design change or in the safety evaluation.

- The lower CCW temperature causes lower lubricating oil temperature for several motors, resulting in higher power requirements. The increased EDG loading was not addressed in either the PMR or the unreviewed safety question determination (USQD). There is no operability concern as the loading increase is small and the diesel generator has a large loading margin. The licensee initiated PIR 97-3978 to resolve this item.

- The lower CCW temperature results in a lower spent fuel pool (SFP) water temperature. The lower SFP temperature effect on reactivity was not addressed in either the PMR or the USQD. The minimum temperature for which the SFP reactivity was analyzed is 60°F (USAR Section 9.1A). The licensee has stated that the SFP temperature could approach within 4°F of the CCW temperature. There was no current operability concern as On The Spot Change 97-0898 to procedure CKL ZL-003, "Control Room Daily Readings," was issued to place an administrative limit of 65°F on minimum SFP temperature until a reactivity analysis at lower temperatures was completed. The licensee initiated PIR 97-4062 to resolve this item. Subsequent to the inspection the licensee completed an analysis which determined that lowering

the SFP temperature from 60°F to 35°F would reduce the reactivity. In addition, the lower temperature has no adverse effect on the solubility of boron because the SFP boron concentration of 2000 - 2500 ppm is well below the saturation curve at 35°F.

The team determined that contrary to 10 CFR 50.59, the licensee's safety evaluation did not completely verify the absence of an unreviewed safety question and that, contrary to Criterion III of Appendix B to 10 CFR Part 50, the design was not adequately verified or checked to ensure that spent fuel pool design was not affected by the design change. (Unresolved item 50-482/97-201-19)

f. Operating Procedures

The team reviewed the CCW system operating procedures to ensure that the system was being operated in accordance with its design basis and the commitments contained in the USAR. The review determined that operating procedures were consistent with the CCW design basis. However, one concern was identified which required resolution.

The USAR Section 9.2.2.2.3 states that CCW flow to the spent fuel pool heat exchanger is reduced or terminated at start of cooldown at 4 hours. The licensee's procedures EJ-120, "Startup of a Residual Heat Removal Train," Revision 32, and EJ-121, "Startup of a RHR Train in Cooldown Mode," Revision 11, only contain a caution to monitor spent fuel pool temperature if cooling is secured. Furthermore, there is no requirement to reduce or terminate CCW flow in procedure EMG ES-11, "Post LOCA Cooldown and Depressurization", Revision 12. The above procedures did not implement the USAR requirements. This discrepancy was previously reported in PIR 95-1167 but the procedures were not corrected. The licensee initiated PIR 97-3897 to resolve this item. This is contrary to the corrective action measures specified in Criterion XVI of Appendix B, 10 CFR Part 50. (Unresolved Item 50-482/97-201-20)

g. Testing Procedures

The team reviewed system flow verification procedures TMP-EN171, "ESW Train A Post-LOCA Flow Balance," Revision 1; TMP-EN173, "ESW Train B Post-LOCA Flow Balance," Revision 1; STN EG-001A, "Train A Component Cooling Water System Flow Verification," Revision 0; and STN EG-001B, "Train B Component Cooling Water System Flow Verification," Revision 0; local leak rate test (LLRT) valve lineup procedures STS PE-174, "LLRT Valve Lineup for Penetration 74," Revision 0, STS PE-175, "LLRT Valve Lineup for Penetration 75," Revision 0, and STS PE-176, "LLRT Valve Lineup for Penetration 76," Revision 0; component cooling water pump inservice pump test procedure STS EG-100B, "Component Cooling Water Pumps B/D Inservice Pump Test," Revision 13; heat exchanger flow and differential pressure trending procedure STN PE-037, "ESW Heat Exchanger Flow and DP Trending," Revision 11; and heat exchanger performance test procedure STN PE-033, "CCW Heat Exchanger Performance Test," Revision 4. The reviews included the latest results and trending data.

The team's review indicated that the CCW system valve leak rate testing was being performed properly as were check valve testing, pump testing, heat exchanger flow, differential pressure and performance testing in accordance with NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," and system flow balancing. No adverse trends were noted in the test results.

E1.3.1.3 Conclusions

The team concluded that the mechanical aspects of the CCW system could perform the design functions of cooling the safety-related equipment during the normal operating mode and in post-

accident conditions. The team noted nonconservative assumptions and errors in calculations, discrepancies in 10 CFR 50.59 and design changes, and an inadequate corrective action for operating procedures.

E1.3.2 Electrical Design Review

The electrical design of the CCW system is reviewed in Section E1.2.2.

E1.3.3 Instrumentation and Controls Design Review

E1.3.3.1 Inspection Scope

The scope of the instrumentation and controls design assessment was a review of the CCW system documents such as Chapters 7 and 9 of the USAR, Section 3/4.3 of the Technical Specifications (TS), system descriptions, calculations, setpoint documents, design specifications, procedures, drawings, and modification packages.

E1.3.3.2 Observations and Findings

a. Isolation of Nonseismic, Non-Safety-Related Portion of the CCW System

The CCW system instrumentation was assessed to verify its capability to isolate the non-safety-related portion of the CCW system, consisting of those lines downstream of valves HV69A and HV70A and upstream of valves HV69B and HV70B. These valves close on low-low surge tank level, high CCW flow, or a safety injection signal. USAR Sections 7.6.8 and 9.2.2.1.1 provide the design bases for this isolation function. The team reviewed drawings M-12EG02, M-12EG03, J-02EG03, J-12EG02, J-12EG08A, and J-12EG08B, schematic diagrams, and setpoint documents and verified that the design meets the design requirements as described in the USAR. TS 4.7.3 requires that a channel operational test of the surge tank level and flow instrumentation for the isolation logic be performed every 31 days, with channel calibration and valve actuation verified every 18 months. Based on a sample review of the surveillance test data for procedures STS IC-916, "Channel Calibration CCW System Automatic Isolation of Non-Nuclear Safety Related Components," Revision 0, and STS IC-915A, "ACOT A Tr Component Cooling Water Sys. NSSR Isolation," Revision 1, the team determined that the tests met the TS requirements.

b. Radiation Monitoring Interlock With Surge Tank Isolation Valve

The radiation monitoring system was assessed to verify its capability to perform the required isolation function. The team reviewed calculation EG-13, drawings M-12EG01, M-12EG02, J-12EG02, and J-12EG03, schematic diagrams, setpoint documents, and vendor drawings and determined that the design meets the system functional requirements. Since the monitors were non-safety-related, the team verified that proper separation from 1E circuits was maintained. Walkdowns were also performed to verify locations of sampling and return points. The team noted that PIRs 96-1129 and 97-0457 had previously identified a recurring problem concerning a reduced sample flow through the radiation monitors, which might affect the values that are listed in USAR Table 11.5-5. The licensee determined that the reduced sample flow did not affect the monitoring function of the radiation monitors. The team noted that the problem was corrected and both PIRs were closed. As a result of the review, the team concluded that the design of the CCW radiation monitoring system interlock was adequate.

c. CCW Instrument Loop Accuracy and Setpoint Calculations

The team reviewed the following licensee setpoint methodology documents, uncertainty calculations and related calculations for various CCW instrument loops to verify that adequate tolerance for instrument error had been incorporated in the design:

1. J-K-GEN, "Instrument Loop Uncertainty Estimates," Revisions 0 and 1

2. J-K-EG01, "Instrument Uncertainty Estimate and Safety Related Setpoints, System EG, Loops 1 and 2," Revision 1

3. J-K-EG03, "Instrument Uncertainty Estimate and Safety Related Setpoints, System EG, Loop 62," Revision 1

4. J-K-EG04, "Instrument Uncertainty Estimate and Safety Related Setpoints, System EG, Loops 77 and 78," Revision 1

5. J-K-EG05, "Instrument Uncertainty Estimate and Safety Related Setpoints, System EG, Loops 107 and 108," Revision 1

6. J-EG01, "Stress Analysis of Instrument Lines, EG Component Cooling Water Surge Tank A," Revision 2

7. J-EG08, "Stress Analysis for Instrument Tubing from RHR HX 1B to Accum. Inj. LP 3 & 4," Revision 4

The team's review determined that the above calculations adequately demonstrated the capability of the instruments to perform their intended function.

d. RG 1.97 Instrumentation for CCW System

USAR Appendix 7A provides the design bases for the CCW RG 1.97 instrumentation. The required instrumentation consists of local indication for CCW flow to the engineered safety features (ESF) systems and main control room indication for CCW inlet temperature to the ESF systems. The licensee took exception to the RG 1.97 requirement for main control room indication of CCW flow, which had previously been evaluated and found acceptable by the NRC. Alternate indication is provided by the local indicators and the plant computer.

Four local flow indicators are provided for CCW flow to the ESF systems (loops FT-95 through FT-98). CCW heat exchangers 1A and 1B outlet temperature indicators (loops TE-31 and TE-32), located in the main control room, provide indication of CCW inlet temperature to the ESF systems. RG 1.97 identifies these instrument loops as Type D, Category 2 variables with a reliable power source. Based on the team's review of USAR Appendix 7A, design documents, and the as-built condition, both the CCW flow and temperature indicators are in accordance with the design bases.

E1.3.3.3 Conclusions

The instrumentation and controls design for the CCW system was considered adequate. All instrumentation setpoints that were reviewed have adequate margin and the technical specification limits were met.

E1.3.4 System Interfaces

System interfaces are reviewed in Section E1.2.4 of this report.

E1.3.5 System Walkdown

E1.3.5.1 Inspection Scope

The team conducted a walkdown of the CCW system and the plant areas, including the CCW pump and heat exchanger room, surge tank room, control room and penetration area, but excluding containment, that housed the CCW system. The team compared system configuration to the design documents and the USAR and looked closely at equipment condition, area cleanliness, tagging, and means used to avoid potential hazards such as missiles, fire, and pipe rupture.

E1.3.5.2 Observations and Findings

The team determined that the overall material condition of the plant areas was good. The equipment sampled matched the design requirements. No concerns were identified concerning configuration, equipment condition or potential hazards. The System Engineer demonstrated a good knowledge of the system and its components and exhibited good "system ownership."

During the walkdown, the team noted that valves EG-V003, V313, V016 and HV059 were missing equipment identification tags. The licensee took prompt measures to correct this condition.

E1.3.5.3 Conclusions

The team determined that generally the CCW system design observed during the walkdown was consistent with the design basis requirements.

E1.4 Updated Final Safety Analysis Report (USAR) and Other Document Reviews

E1.4.1 Inspection Scope

The team reviewed applicable USAR sections for the RHR, CCW, EDG, ESW, auxiliary building HVAC, instrumentation, and electrical systems. The team also reviewed the system descriptions, drawings, calculation control, and program plans concerning design basis and licensing basis issues.

E1.4.2 Observations and Findings

a. USAR Review

The team identified the following discrepancies in the USAR.

- Different values were referenced for the RWST water volumes in the documents listed below. PIR 97-4018 was initiated to address the inconsistencies.

  (1) USAR Section 6.3.2.2 (page 6.3-6) stated that the minimum RWST volume "available" or "assured" for ECCS injection mode operation is 394,000 gallons. Another paragraph in the same USAR section refers to "usable" volume. However, TS 3/4.5.5 specified the 394,000 gallons as the minimum contained water volume; (2) USAR Table 6.3-1 listed 419,000 gallons as maximum

volume, 407,000 gallons as normal capacity, and 394,000 gallons as assured water volume. These three RWST volumes are also shown in USAR Figure 6.3-7 and system description, M-10BN(Q), Figure 1; (3) USAR Table 6.2.1-5 listed RWST water volume of 370,000 gallons for containment analysis; and (4) USAR Table 6.3-10 listed 326,860 gallons as RWST volume for ECCS cooling.

- NUREG-0881 (Wolf Creek), Section 8.3.2.1.2, refers to the same section in NUREG-0830 (Callaway) for a discussion of the NRC staff's position on battery capacity. That section of NUREG-0830 states that the licensee revised the USAR in Revision 6 to state that batteries are sized in excess of the 50 percent margin required. Callaway USAR was revised, but the Wolf Creek's USAR has not been revised to reflect similar changes. (Refer Section E.1.2.2.2.2.b)

- USAR Section 9.2.2.2.2 states, "The normally closed parallel sets of containment isolation valves will allow the operator to establish cooling water to the reactor coolant pumps and the excess letdown heat exchanger under emergency conditions, with a single failure." However, USAR Table 3-11(B)-3 listed the motor operators for these valves as category C, EQ not required. The currently installed motor operators are class RH, that is, environmentally qualified (reference EQWP-Limitorque, Checklist 1, Supplement 15). The licensee initiated PIR 97-4126 to resolve this item.

- Calculation EG-06-W, "Component Cooling Water System Calculation," Revision W-3, determined that the CCW heat exchanger heat transfer coefficient was 190 Btu/hr-ft^2-F based on the revised ESW flow of 7150 gpm. The USAR stated that the transfer coefficient was 193 Btu/hr-ft^2-F.

- Calculation SA-89-017, "Evaluation of CCW & RHR Heat Exchanger Performance for the Extended Fuel Operating Cycle (18 Months)," Revision 0, determined that CCW temperature reaches 126 °F. However, USAR Table 9.2-11 is based on a CCW temperature of 120 °F. The licensee initiated PIR 97-4052 to resolve this item.

- USAR Fig. 5.4-8 showed suction for RHR pumps A and B as coming from RCS hot leg loop 4, whereas system description M-10EJ(Q) and P&ID M-12EJ01 showed loop 1 for pump A and Loop 4 for pump B. The licensee initiated PIR 97-3823 to resolve this item.

- USAR Section 6.3.5.3, "Flow Indication," stated that the flow from each RHR subsystem to the RCS cold legs is recorded in the main control room. This contradicted USAR Table 7.5-1 and P&ID M-12JE01 (Loop FT-988), which showed this parameter as being indicated (instead of recorded) in the main control room. The licensee issued PIR 97-4179 to update the USAR.

- USAR Table 7A-3 showed a range of 0-60 psig, whereas, the actual installed range was 0-69 psig. PIR 98-0062 was issued to update the USAR.

- USAR pages 6.3-6, 9.2-43, 9.2-45, and 9.2-48 incorrectly described the control function of the RWST auxiliary steam heating system with respect to winterization procedure STN GP-001. USAR Change Request Log No. 97-044 was issued to update the USAR.

- USAR Section 8.3.2.1.2 stated that a Class 1E battery is to supply the loads in Tables 8.3-2 and 8.3-3 for 200 minutes where it should actually be for 240 minutes.

- USAR Section 7.4.1 states that the RWST level transmitters are required for safe shutdown. However, Table 3.11(b)3 does not list these transmitters as required for hot or cold shutdown.

The above discrepancies had not been corrected and the USAR updated to assure that the information included in the USAR contained the latest material as required by 10 CFR 50.71(e). (Unresolved Item 50-482/97-201-21)

b. System Description Review

The following system description discrepancies were identified:

- USAR Section 5.4.7.2.1 stated that no motor-operated valves (MOVs) in the RHR system are subject to flooding. The RHR system description, M10-EJ(Q), stated that the only MOVs subject to flooding were the suction isolation valves. The suction isolation valves were above the 2007'-11" elevation in containment, whereas, as per USAR Section 6.3.2.2, the maximum flood level in containment post-LOCA was 2004'-6." The licensee issued PIR 97-3782 to revise the system description.

- RHR system description, M10-EJ(Q), stated that, "A leaktight seal is provided so that neither the pressure vessel nor the guard pipe is connected directly to the sump or containment atmosphere." This statement contradicted drawings M-109A-00015, M-03EJ05 and C-1L2311. These drawings indicated that the guard pipe is connected to the containment sump and that

  the outer diameter of the pipe does come in contact with the containment sump as it enters the sump. The licensee issued PIR 97-3805 to revise the system description.

- Chemistry Specification Manual, AP 02-003, Revision 5, gave different chemistry parameters than the CCW system description M-10EG(Q), Revision 2. The licensee initiated PIR 97-3466 to resolve this item.

- The USAR stated that one of the safety design bases of the CCW system was to provide heat to maintain the ESW inlet trash racks from being blocked with frazil ice. This safety design basis was not discussed in system description M-10EG(Q), Revision 2. The changes to the USAR were made as part of DCP 06349, which apparently overlooked the CCW system description. The licensee initiated PIR 97-3885 to resolve this item.

- Load profiles in the Class 1E battery system description did not agree with those in calculation NK-E-002 for the Class 1E batteries; in addition, incorrect values were stated in the system description for Class 1E batteries' minimum voltage, amp-hour rating, etc. The licensee initiated PIR 97-4190 to resolve this item.

- USAR Tables 9.2-9, 10, and 11 and CCW system description M-10EG(Q), Revision 2, Table 2 did not agree on total flow and heat load duty. The licensee initiated PIR 97-3983 to resolve this item.

c. Drawing Review

The team identified the following discrepancies:

- Drawing M-12EJ01 showed loop FT618 low flow alarm with a PAL(low pressure alarm) designation, which appeared to be in error. This conflicted with drawing E-03EJ12 which shows

FAL(low flow alarm). The licensee concurred that M-12EJ01 is in error. PIR 97-3824 was written to update the drawing.

- Drawings J-110-319 and 320 for the RWST temperature monitoring system, indicated connections to the control room temperature indicators with ungrounded shield or with no shield. These connections were not consistent with the licensee's standard wiring design for instrumentation. The licensee determined that the as-built wiring to the instruments were shielded and that the drawings were in error. PIR 98-0063 was issued to update the drawings.

d. Calculation Control Review

Wolf Creek inspection activities involved the review of over 130 calculations. The inspection team observed that there were many active calculations for each system. Some of the calculations reviewed did not form a part of the design basis, making the design basis difficult to identify in some cases. Further, several calculations for each system had similar purposes and/or similar results. Although these calculations were not found to be contradictory, the practice tended to further confuse identification of the design basis. Finally, the licensee apparently preferred supplementing existing calculations with new calculations to superseding, archiving, or revising existing calculations. This practice increased the population of active calculations. The inspection team was concerned that the current situation could cause inadvertent use of non-design-basis data in future design change or analysis activities.

e. Design Basis / Licensing Basis (DB/LB) Review Program Review

The inspection team noted that in early 1997 Wolf Creek staff established a design basis/licensing basis (DB/LB) review program to address the types of concerns identified by this inspection. Examples of the seven initiatives embodied in the licensee's DB/LB program are a USAR "fidelity" review project, a plan for periodic safety system functional self-assessments (the first of which was conducted on the essential service water system during the first half of 1997), and the establishment of an Engineering Information System. The latter was expected to help track design basis information. During an interview with the program manager for the USAR fidelity review effort, the team noted that the licensee was identifying many discrepancies and was taking required actions to update the USAR.

While this DB/LB program had not produced widespread or consistent results at the time of the inspection, the initiatives were highly encouraging. The scope of this inspection did not include an effectiveness review of the DB/LB program.

E1.4.3 Conclusions

The team concluded that in several instances the USAR was not revised in accordance with 10 CFR 50.71(e) requirements. Discrepancies were also identified in system descriptions, drawings, and the identification of design basis data in calculations.

XI Exit Meeting Summary

After completing the onsite inspection, the team conducted an exit meeting with the licensee on January 9, 1998. During the meeting the team presented the results of the inspection. A list of persons who attended the exit meeting is contained in Appendix B. Proprietary material was reviewed during this inspection but this report contains no proprietary information.

APPENDIX A

OPEN ITEMS

This report categorizes the inspection findings as unresolved items and inspection follow-up items in accordance with NRC Inspection Manual, Manual Chapter 0610. An unresolved item (URI) is a matter about which more information is required to determine whether the issue in question is an acceptable item, a deviation, a nonconformance, or a violation. The NRC Region IV office will issue any enforcement action resulting from their review of the identified URIs. An inspection followup item (IFI) is a matter that requires further inspection because of a potential problem, because specific licensee or NRC action is pending, or because additional information is needed that was not available at the time of the inspection. The URIs and 0IFIs found in this inspection are listed below:

Item Number	Finding Type	Title
50-482/97-201-01	URI	Cooldown Analysis (Section E1.2.1.2(a))
50-482/97-201-02	IFI	ECCS Leakage (Section E1.2.1.2(d))
50-482/97-201-03	URI	RHR Pump Operation in Minimum Recirculation Mode (Section E1.2.1.2(h))
50-482/97-201-04	URI	Motor Control Center Circuit Length (Section E1.2.2.2.1(d))
50-482/97-201-05	IFI	120 Vac Short Circuit and Voltage Drop Analysis (Section E1.2.2.2.1(d))
50-482/97-201-06	IFI	Procurement of EDG Relay (Section E1.2.2.2.1(e))
50-482/97-201-07	IFI	Battery Load Profile (Section E1.2.2.2.2(b))
50-482/97-201-08	IFI	TS Change for Batteries (Section E1.2.2.2.2(b))
50-482/97-201-09	IFI	Battery Sizing (Section E1.2.2.2.2(b))
50-482/97-201-10	URI	DC Voltage Drop Calculation (Section E1.2.2.2.2(d))
50-482/97-201-11	URI	Minimum Battery Voltage (Section E1.2.2.2.2(d))
50-482/97-201-12	URI	Load Growth Control (Section E1.2.2.2.2(e))
50-482/97-201-13	URI	Acceptance Criteria for Battery Test (Section E1.2.2.2.2(f))
50-482/97-201-14	URI	Corrective Action for Battery Test (Section E1.2.2.2.2(f))
50-482/97-201-15	URI	RWST Level Instrumentation (Section E1.2.3.2(a))
50-482/97-201-16	URI	Seismic Qualification (Section E1.2.3.2(b))

50-482/97-201-17	URI	Nitrogen Bottle Installation (Section E1.2.5.2(d))
50-482/97-201-18	URI	MOV Differential Pressure (Section E1.3.1.2(c))
50-482/97-201-19	URI	CCW Low Temperature (Section E1.3.1.2(e))
50-482/97-201-20	URI	Corrective Action for CCW Operating Procedure (Section E1.3.1.2 (f))
50-482/97-201-21	URI	USAR Discrepancies (Section E1.4.2(a))

APPENDIX B

EXIT MEETING ATTENDEES

Wolf Creek Nuclear Operating Corporation

O. Maynard, President and CEO
C. Warren, Vice President , Operations/COO
R. Muench, Vice President, Engineering
T. Garrett, Manager, Design Engineering
C. Younie, Manager, Operations
R. Sims, Manager, Systems Engineering
M. Angus, Manager, Licensing and Corrective Action
W. Norton, Manager, Performance Improvement and Assessment
C. Fowler, Manager, Integrated Plant Scheduling
N. Hoadley, Manager, Support Engineering
R. Flannigan, Manager, Nuclear Safety and Licensing
R. R. Osterrider, Supervisor, Safety Analysis
B. Smith, Lead Engineer, Design Engineering
J. Yunk, Senior Engineering Specialist
J. Stamm, Supervisor, Safety Analysis
R. Rietmann, System Engineer
M. Blow, Superintendent, Chemistry
T. Damashek, Supervisor, Licensing
R. Holloway, Project Engineer
B. Masters, System Engineer
L. Solorio, Design Engineer
W. Eales, Design Engineer
W. Selbe, Project Engineer
M. Guyer, Operations

U.S. Nuclear Regulatory Commission

R. Mathew, Team Leader, NRR
S. Richards, Chief, PECB, NRR
T. Stetka, Acting Chief, EB, RIV
W. Johnson, Chief, Project Branch, RIV
F. Ringwald, SRI, NRC
K. Neubauer, Contractor, S&L
M. Sanwarwalla, Contractor, S&L
A. Rahman, Contractor, S&L
G. Bizarra, Contractor, S&L
R. Sheldon, Contractor, S&L

APPENDIX C

LIST OF ACRONYMS

AC, ac	Alternating Current
AMPS	Amperes
AWG	American Wire Gauge
BTU	British Thermal Unit
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CPT	Control Power Transformer
CS	Containment Spray
Cv	Valve Flow Coefficient
DB	Design Basis
DC, dc	Direct Current
DCP	Design Change Package
DP	Differential Pressure
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPA	Electrical Penetration Assemblies
EQ	Environmental Qualification
ESF	Engineered Safety Features
ESW	Essential Service Water
F	Fahrenheit
FE	Flow Element
FLA	Full Load Amperes
ft, FT	Feet or Foot
gal	Gallons
GL	Generic Letter
gpm	Gallons Per Minute
HVAC	Heating, Ventilating, and Air Conditioning
I&C	Instrumentation and Controls
IEEE	Institute of Electrical and Electronics Engineers Inc.
IFI	Inspection Follow-up Item
IN	Information Notice
IST	Inservice Testing
ITIP	Industry Technical Information Program
kVA	Kilovolt-Ampere
kV	Kilovolt
kW	Kilowatt
LB	Licensing Basis
LC	Locked Rotor Current
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLRT	Local Leak Rate Test
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LT	Level Transmitter
MCC	Motor Control Center
MOV	Motor Operated Valve

NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation, Office of (NRC)
NSSS	Nuclear Steam Supply System
P&ID	Piping & Instrumentation Diagram
PI	Pressure Indicator
PIR	Performance Improvement Request
PMR	Proposed Modification Request
ppm, PPM	Parts Per Million
PSB	Power System Branch
psi, PSI	Pounds per Square inch
psia, PSIA	Pounds per Square Inch Absolute
psid, PSID	Pounds per Square Inch Differential
psig, PSIG	Pounds per Square Inch Gauge
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REV	Revision
RG	Regulatory Guide
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SBO	Station Blackout
S&L	Sargent & Lundy
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SI	Safety Injection
SIS	Safety Injection Signal
SNUPPS	Standardized Nuclear Unit Power Plant System
STP	Surveillance Test Procedure
SW	Service Water
TS, Tech. Spec.	Technical Specifications
USAR	Updated Safety Analysis Report
UHS	Ultimate Heat Sink
URI	Unresolved Item
USQ	Unreviewed Safety Question
USQD	Unreviewed Safety Question Determination
Vdc	Volts DC
Vac	Volts AC
W	Watts
WCAP	Westinghouse Containment Analysis Program
WCGS	Wolf Creek Generating Station
WCNOC	Wolf Creek Nuclear Operating Corporation