ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-285/95-11

License: DPR-40

Licensee: Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 399, Hwy. 75 - North of Fort Calhoun
Fort Calhoun, Nebraska

Facility Name: Fort Calhoun Station

Inspection At: Blair, Nebraska

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EXECUTIVE SUMMARY

This inspection was conducted to evaluate the licensee's ability to provide effective engineering and technical support to the plant. The inspection activities encompassed the activities of production engineering (which includes both system and design engineering), the functioning of the Nuclear Safety Review Group, and the licensee's processing of 10 CFR 50.59 safety evaluations.

The inspection focussed on three systems, the 125 Vdc power system, the emergency diesel generator system, and the high pressure safety injection system. The team found that the licensee was effectively maintaining these systems. The team considered the modifications to upgrade obsolete equipment as a positive step towards improving overall system reliability and the use of innovative post-modification testing techniques to be a strength.

The team's review of system design information identified several minor inaccuracies between the physical system configurations, the design basis documents, and the updated safety analysis report in each of the three systems reviewed. The licensee was aware of these problems and had good programs to correct the problems. The team concluded that the programs to correct the inaccuracies in the design bases documents and the updated safety analysis report appeared to be effective and that good progress was being made to reduce the number of these inaccuracies.

The team noted that engineering was involved and supporting plant activities. System engineers were observed to be very knowledgeable and involved with their assigned systems. This knowledge and involvement in system activities was considered a strength. However, the team found that the system engineers' knowledge was lacking in the area of system risk. The team attributed this lack of knowledge to the fact that the probability risk assessment information for the systems were not current nor distributed to the system engineers. As a result, the system engineers were not aware of the effects of modeling assumptions, system sensitivity issues, operator actions, system component failures, and system interactions on their systems. The team noted that the licensee was making acceptable progress in resolving this concern.

The team concluded that design engineering was accomplishing it's goals and managing it's workloads. Examples included the program to correct inaccuracies in the design basis documents and updated safety analysis report and the effective erosion/corrosion program. However, the team also concluded that design engineering had made nonconservative recommendations, had not effectively supported day-to-day plant operations through a lack of communications, and had demonstrated a lack of knowledge of design requirements. For example, design engineering had recommended an increase in the ambient temperature limits on the diesel generators prior to including all instrument errors into the final calculations. In addition, the lack of timely notification to the operations department of new diesel generator operating limits demonstrated a lack of interdepartmental communication. Finally, the failure to consider the effect of the high ambient temperatures on the diesel generator air start system indicated a less than thorough review and knowledge of all system parameters. The team also noted that engineer training was inconsistent. Although the system engineer training was complete and current, design engineer training was lagging especially in the electrical and reactor engineering areas. The team also identified a minor concern regarding the administrative control of engineering assistance requests. Specifically, the team identified that 34 of the approximately 50 engineering assistance requests reviewed had untimely responses.

The team noted that the licensee's response to a recent industry event involving a pipe rupture because of a less than adequate erosion/corrosion program was exemplary. The licensee was aware of the event and had already addressed the potential for failure. The team considered the licensee's activities in the erosion/corrosion area to be a strength. However, in one area involving diagnostic testing of air operated valves, the licensee's activities were only of a reactive nature. Engineering's long-term plans involved the use of such testing for only post-maintenance testing and not in-service testing activities.

The team found the effectiveness of the licensee's nuclear safety review group to be a strength. Their assessments were very detailed and identified substantial issues. The team also noted that the corrective action recommendations from these assessments were effective and implemented in a timely manner.

The team's review of the plant modification process identified that the licensee's use of the substitute replacement item process was not properly implemented. The licensee's procedures allowed the use of the substitute replacement item process if the modification did not involve a change to the plant's design basis. However, the team found that the process was used to perform the modification involving the diesel generator cooling system which did involve a change to the plant's design basis. In addition, the team's review of seven other modifications that used the substitute replacement item process, identified that these modifications also involved a change to the plant's design basis. The team considered the failure to perform 10 CFR 50.59 safety evaluations for these modifications to be a violation.

In addition, the team's review of 10 CFR 50.59 safety evaluations indicated that a lack of attention to detail by the licensee's reviewers still existed. For example, the team identified a number of examples where the evaluations were not properly completed. Although many of these examples were minor in nature, this problem was identified as the result of NRC inspection approximately 2 years ago.

DETAILS

1 SYSTEM REVIEWS (37550)

1.1 125 VDC POWER SYSTEM

1.1.1 System Operation and Testing

1.1.1.1 Surveillance Testing

The team reviewed the surveillance tests performed on the dc system and compared these tests with the requirements of the Technical Specifications to assure that the Technical Specification requirements were being accomplished. The team also reviewed the last two Technical Specification required surveillance tests for the 125 Vdc system to determine if all acceptance criteria were met. The team concluded that the surveillance tests were technically adequate and that all Technical Specification requirements were met.

1.1.1.2 System Physical Condition

The team performed a walkdown of the 125 Vdc system using Operating Instruction OI-EE-3,"125 VDC System Normal Operation." The team verified the breaker and switch positions as defined in Procedure Checklist OI-EE-3-CL-A. All breakers and switches were found to be in the proper position for normal plant operation. The team also verified that the breakers had the correct current interrupting capability as defined in Drawings 2C6288 and 2C6289. The team found the housekeeping conditions within the electrical equipment to be very good. There were no signs of loose connections or component problems.

1.1.2 System Reliability

1.1.2.1 System Engineering Activities

The team interviewed the 125 Vdc system engineer to determine the engineer's knowledge and involvement in system activities. The team found the system engineer to be very cognizant of the physical condition of the system, indicating that frequent plant walkdowns were being conducted. When questioned, the engineer was able to accurately identify the deficiency tags in place and the status of outstanding modifications and engineering assistance requests. The team concluded that the system engineer's activities were a positive factor in improving system reliability.

1.1.2.2 System Report Cards

The team noted that system engineering issued system report cards on a quarterly basis. The report card format was determined by the system engineer. The report card for the 125 Vdc system comprised the following topics:

- Assessment and trend of overall system performance and repair backlog;

- General discussion and justification of overall system performance and repair backlog assessments and trends;

- System performance, equipment performance, and reliability concerns or trends;

- Major activities which occurred during the report period;

- Major activities anticipated during the next reporting period; and,

- Significant items awaiting resolution.

The team concluded that the system report cards provided useful information that would provide management with an accurate and timely status of the condition of the 125 Vdc system.

1.1.3 Design Changes

1.1.3.1 Design Basis Review

The team reviewed the dc distribution design basis document, SDBD-EE-202, the updated safety analysis report, and the electrical distribution system training manual. The team identified a discrepancy in updated safety analysis report Section 8.3.4.2, in that the description of the station batteries was for the batteries that had been replaced in 1991. In addition, the nameplate on the battery chargers listed the input current as 94 amperes, while the system training manual and the design basis documents listed this value as 96 amperes. These minor discrepancies are further discussed in Section 2.3.1.

1.1.3.2 System Modifications and Calculations

The team reviewed three modifications and five calculations involving the 125 Vdc system. The calculations appeared to be adequate and no problems were identified. Of the three modifications reviewed, Modification MR-FC-94-002 was reviewed in detail.

Modification MR-FC-94-002 involved the replacement of both 125 Vdc distribution buses. The team reviewed the modification package and determined that a proper 10 CFR 50.59 safety evaluation had been performed. The team also reviewed the results of post-modification testing and found it to be acceptable. The team noted the use of thermography on the electrical connections to assure that the connections were properly secured. The team considered the use of thermography to enhance post-modification testing to be a strength.

The team inspected the physical installation of the new 125 Vdc Distribution Buses EE-8F and EE-8G, the battery chargers, and the inverters. During this inspection, the team noted that approximately 3 strands were cut from a

19-strand 2/0 cable at the terminals of Switches EE-8F-CB20 (Bus EE-8F) and EE-8G-CB22 (Bus EE-8G). The team was concerned that the cut strands could potentially reduce the ampacity of the cable at these connections.

The licensee initiated Incident Report 950545 to determine the cause of the cut strands and to evaluate its acceptability. The licensee indicated that the 2/0 cable routed to these switches was supplied by the vendor of the new electrical busses. The licensee had determined during bus installation that the 2/0 cable would not fit in the lugs of the switches; therefore, the field engineer had contacted design engineering for assistance. Design engineering, using engineering judgement, determined that cutting the conductor strands was acceptable because of the large margin of cable ampacity versus load. Thus, three cable strands were cut and the cable lugged. Incident Report 950545 documented the licensee's calculation that the 2/0 cable had a current carrying capability of 186 amps with three strands cut. This was considerably higher than the actual load of 42 amps.

The team noted that this cable modification was not documented in a field design change request, modification package, or the construction work orders as required by the licensee's modification control procedure and engineering judgement procedure. This failure constitutes a violation of minor safety significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy.

During this inspection, the team also noted that a 186-ampere rating conductor was protected by a 300-ampere fuse. Additional review by the licensee indicated that the new 125 Vdc buses came with breakers that had magnetic trips. Since the previous configuration had switches that had no interrupting capability, these circuits were each protected by 300-ampere fuses. The electrical circuit consisted of a small length of 2/0 cable located within the 125 Vdc distribution panel, which was connected to a manual switch rated at 100 amperes. A 1/0 cable (160-amp rating) was connected on the outlet side of the switch and ran to a dc distribution panel in the control room. The team discussed this arrangement with licensee personnel and noted that it appeared that a potential fault could occur on this circuit that was greater than the cable rating, but less than 300 amperes. This could result in a cable insulation fire inside the 125 Vdc distribution panel and possibly the loss of the entire bus prior to circuit interruption by the 300-ampere fuse. The licensee documented this concern in Incident Report 950575. The licensee's evaluation determined that any high impedance fault caused by a control room fire would not exist long enough for the 1/0 cable in the 125 Vdc distribution panel to reach its ignition temperature. The team's independent review concluded that a short circuit would not result in a cable fire within the dc distribution panel prior to being interrupted by the 300-ampere fuse.

The team also reviewed Engineering Change Notices 93-076 and 93-130. These engineering change notices involved the replacement of the input 125 Vdc circuit breakers on all four safety-related inverters (93-076) and the two nonsafety-related inverters (93-130). The engineering change notices were required since the existing circuit breakers were obsolete and unavailable for replacement. The team concluded that these engineering change notices were properly implemented.

1.1.4 Other System Attributes

The team reviewed selected incident reports that were written against the 125 Vdc system over the past 2 years. The team concluded that incidents were adequately described, proper corrective action was taken or proposed, and that no negative trends existed.

The team reviewed nine engineering assistance requests to determine whether the responses by engineering were adequate and timely. The review of engineering assistance requests on the 125 Vdc system indicated that, while the responses were adequate, there were several cases in which the responses, reviews, or formal closure were not timely.

Based upon these reviews, the team concluded that the licensee's administrative controls were lacking, in that they did not provide for the timely disposition of engineering assistance requests. However, the team also concluded that, based upon the engineering assistance requests sampled, the delay in engineering assistance requests closeout did not affect system operability. The team's concern regarding the timeliness of the closure of engineering assistance requests is further discussed in Section 2.4 of this report.

The team also reviewed the list of maintenance work orders that had been written against the 125 Vdc system in the last 3 years. The team noted that there had been nine maintenance work orders written for troubleshooting dc bus grounds. The team reviewed these work orders and found that several were for routine troubleshooting and not because of alarms received. The team concluded that the maintenance history for the 125 Vdc system did not indicate an adverse trend.

1.2 Diesel Generators

1.2.1 System Operation and Testing

1.2.1.1 Surveillance Testing

The team reviewed the surveillance tests for the diesel generators and the associated surveillance test results. The results of the monthly operability tests for the diesel generators as well as the quarterly tests for the diesel generator fuel oil system and the starting air compressor discharge check valves were compared against Technical Specification surveillance requirements. In addition, the team reviewed the 18-month surveillance test that checked the circuit for the diesel generator auto start initiation. The team concluded that the Technical Specification surveillance requirements were met by the applicable surveillance procedures and that actual test results fell within the allowable tolerances.

The team reviewed Licensee Event Reports 93-017 and 95-01, which documented as-found data for the offsite power low signal timing relays were out of tolerance. The team noted that the licensee's root-cause analysis did not identify the cause for the instrumentation drift. As a result, the licensee implemented a reduced tolerance range in the calibration procedure to assure that the Technical Specification tolerances were not exceeded. The team concluded that the licensee's actions were effective.

1.2.1.2 System Physical Condition

The team walked down the diesel generators with the responsible system engineer. During the walkdown, the team noted that the system engineer was familiar with the outstanding deficiency tags on the system. Many of the deficiency tags were for minor leaks on the engine and generator portions of the diesel generator system. Based on walkdowns and discussions with the system engineers, the team noted that the housekeeping was adequate. The team also noted that system problems were usually documented in system report cards on a quarterly basis.

During this walkdown, the team identified an out-of-date Operating Instruction, OI-DG-1, "Diesel Generator No. 1 Normal Operations," Revision 14, posted in the Diesel Generator No. 1 room. Revision 15 of OI-DG-1 was the current and correct revision. Revision 14 did not reflect the current physical condition of Diesel Generator No. 1. The team noted that this discrepancy was previously identified by the licensee during a quarterly audit on July 7, 1995; however, the out-of-date instruction had not been replaced. As the result of further investigations, the licensee determined that three additional out-of-date procedures, abnormal operating procedures (AOP-06 and -07) and operating procedure (OP-11) were posted in the plant. The licensee immediately corrected these out-of-date procedure conditions.

The team also noted that operating instructions for the two emergency diesels were not consistent in that the instructions for one diesel generator were not included in the other diesel generator's procedure. As the result of these observations, the licensee issued procedure corrections and Incident Reports 950534 and 950551.

The team concluded that the licensee failed to properly maintain the diesel generator operating instructions and that these findings did not affect diesel generator operation. This failure constitutes a violation of minor safety significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy.

1.2.1.3 Special Diesel Generator Ambient Temperature Testing

As the result of a modification made to the diesel generator radiators to improve engine cooling, the licensee conducted post-modification testing to determine the maximum ambient temperature at which the diesel generators could operate. While this modification had been installed in 1993, the final maximum limit temperature testing (which required an ambient temperature of 95°F or greater) could not be completed until 1995 because of the lower than normal ambient temperature conditions over the past 2 years.

The team reviewed the engineering staff's activities involving diesel generator ambient temperature testing. While the testing for Diesel Generator No. 1 indicated that diesel generator's load limits could be exceeded, the results for Diesel Generator No. 2 (due to it's less than capacity loads) indicated that much higher temperatures were needed to exceed that diesel generator's capacity. The licensee completed the diesel generator ambient temperature testing for Diesel Generator No. 1 on July 10, 1995. Licensee Calculation FC03382, based on preliminary test information, showed that Diesel Generator No. 1 could still supply the emergency loads at ambient temperatures of less than or equal to 110°F. This information was documented by design engineering to the plant review committee in Memorandum PED-DEN-95-519, dated July 11, 1995. Following the plant review committee's approval, the licensee revised Technical Data Book Curve III.26.A, which provided diesel generator output load limits at various ambient temperature conditions. This change raised the ambient temperature operational limit for Diesel Generator No. 1 from 104°F to 110°F. The 10 CFR 50.59 safety evaluation for this procedure change documented that updated safety analysis report Table 8.4-1 did not need to be revised because the limits in this table would not be exceeded.

Following the completion of testing activities, the licensee checked the calibration of the test instrumentation used in the performance of the test to determine the effect of instrument accuracy tolerances on the initial calculations. On July 19, following receipt of this post-test instrument calibration data, the licensee performed a new Calculation FC05916, and documented the results in Memorandum PED-DEN-95-535, dated July 19, 1995, to the lead operations engineer. This memorandum indicated that Diesel Generator No. 1 would exceed the Technical Specification mandated 2000-hour load rating of the engine for a 15-minute period during the first 40 minutes following a design basis accident when ambient temperatures were in excess of 107°F. In addition, the memorandum identified that inconsistencies existed between the diesel generator parameters found in Section 8.4 of the updated safety analysis report, the Technical Data Book, and Calculation FC05916. Design engineering considered the exceeding of the 2000-hour load rating (when ambient temperatures were in excess of 107°F) to be acceptable even though it exceeded the Technical Specification limit, and, as a result, did not notify the operations department of this new information. Design engineering based their determination that the higher temperature was acceptable on the fact that the 2000-hour load limit was only a manufacturer's limit that shortened the diesel generator overhaul frequency and that exceeding this load limit would not affect diesel generator operation. As the result of this new calculation, the licensee determined that the 10 CFR 50.59 safety evaluation was inconsistent and issued Incident Report 950550.

Between the time that the post-test instrument calibration data was received by design engineering on July 19 and the beginning of the NRC inspection on August 7, the team noted that the curves in the licensee's Technical Data Book and posted in the control room were still based on the 110°F limit. In addition, although the team's review of recorded ambient temperatures indicated that ambient temperatures at the plant had not exceeded the revised limit of 110°F, the ambient temperatures had exceeded the actual temperature limit of 107°F on July 12, 1995.

When the plant review committee was informed by design engineering of the new ambient temperature limits on August 7, the licensee initiated Incident Report 950532, to identify that the revised 110°F ambient temperature limit for Diesel Generator No. 1 would violate license conditions. The licensee issued Operations Memorandum 95-12 on August 8, 1995, to identify that the interim ambient temperature limit for Diesel Generator No. 1 be reduced to 107°F and provided updated Technical Data Book curves to reflect the more limiting condition.

Design engineering performed further evaluations and determined that a modification to the design basis accident (large break loss-of-coolant accident) model reduced the Diesel Generator No. 1 loads. The licensee determined that since the auxiliary feedwater pumps and containment spray pumps would only be in a recirculation mode of operation during the initial phase of the accident, that the loading effect of these pumps could be reduced. The engineering staff concluded that with the reduced load demand modeled in the analysis the subsequent load for Diesel Generator No. 1 would be within the Technical Specification 2000-hour load limit at ambient temperatures up to 110°F.

The team concluded that the design engineering recommendation to raise the ambient temperature operating limit of Diesel Generator No. 1 before considering the effect of the post-test instrument tolerance calibration data and without taking actions to assure that the Technical Specification limit would not be exceeded, was nonconservative. Additionally, even though subsequent calculations performed by design engineering resulted in the determination that the initial temperature limit of 110°F was an acceptable operational limit, the team concluded that there was a lack of timely communications by design engineering to other plant departments of the new operational limits.

1.2.1.4 Diesel Generator Operation During Shutdown Conditions

The team reviewed Engineering Analysis EA-FC-91-034, "AOP-32 480 Volt Island Bus Transfer," and confirmed that transferring island bus loads under accident conditions can cause the diesel generators to exceed their design load limits. The licensee had implemented administrative controls to prevent these conditions through procedural precautions and notes in Operating Instruction OI-EE-2B, "480V Hot Bus Transfer," Revision 6 and Abnormal Operating Procedure BD-AOP-32, "Loss of 4160 Volt or 480 Volt Bus Power," Revision 2. When questioned by the team, the licensee stated that there was no analysis or test which verified the diesel generator capability to properly supply corresponding loads during an emergency start situation while the plant is in a shutdown condition at less than a 300°F reactor coolant system temperature. The engineering staff indicated that diesel generator loading during shutdown is manually controlled by operator actions to maintain load less than the Technical Data Book operational curves. The team noted that during plant shutdown conditions the loads that would normally be supplied by the diesel generator are reduced. The team concluded that adequate administrative controls were in place for reliable operation of the diesel generator, but also noted that under the most limiting accident scenario conditions, the design limitations of the diesel generator required heightened operator attention to assure diesel generator availability.

1.2.2 Design Changes

The team reviewed Modification MR-FC-92-044 which, in part, replaced the master emergency key switch (183MES/D2) for Diesel Generator No. 2 with a nonkey switch as part of a simplification of the engineered safeguards system. Licensee personnel indicated that the design modification would prevent inadvertent operation since the switch is located behind a cover which is normally closed and latched. The team noted that the Diesel Generator No. 2 breaker control switch had the key stored in place to enable generator breaker closure from local Control Panel AI-133B. The team verified key control was maintained in accordance with Standing Order SO-O-26, "Plant Keys." The team observed that the post-modification testing verified proper operation for all the master emergency switch functions including isolation of diesel generator control from the control room; generation of a backup loss-of-voltage trip for the 4160 Vac bus; control of the 480 Vac breaker trip logic; transfer of diesel breaker control; control room annunciation; initiation of motor control center load shed; damper control; and opening of associated 4160 Vac breakers. The licensee identified in Nuclear Safety Review Group Assessment SRG-94-122, that updated safety analysis report Sections 7.3.6 and 7.6.4 described key switches and, thus, did not reflect the nonkey switch modification for Diesel Generator No. 2. The team noted that the post-modification review was still in progress, and that completion of the updated safety analysis report and design basis documents revisions normally occur after receipt of the modification review report by design engineering. The team concluded that this design modification was technically acceptable and was installed properly.

The team noted that no temporary modifications were presently installed in the diesel generator systems. The team also noted during a system walkdown that the air supply to the Diesel Generator No. 1 air intake dampers was isolated and tagged. This kept the dampers in their failed open position. The purpose of this tag out was to prevent an icing condition from rendering the dampers inoperable if left in the closed position. Control of these dampers was by the licensee's tag out procedure. The licensee was planning to install a design modification for these dampers in September 1995. The team concluded that this action was adequate.

1.2.3 Other System Attributes

The team identified an instrument air filter regulator that did not appear on instrument air Drawing 11405-M-264. This filter regulator was located in the supply line for the intake air dampers for the respective diesel generators. Subsequent to this observation, the team noted that an in-line filter regulator did appear on Drawing EM-871, but was not labeled with any component identification number. The licensee wrote Engineering Change Notice 95-302 to correct this minor discrepancy.

The team noted that Engineering Assistance Request 91-128 was written to address the need for a configuration check on the diesel generator damper drawings for both diesel generators. However, the response was rejected by system engineering because it only resolved the issue for Diesel Generator No. 2 and did not address the dampers for Diesel Generator No. 1 as originally requested. As the result of this rejection, the engineering assistance request was left open and had been left open for approximately 4 years.

As previously discussed in Section 1.1.4, the team concluded that the licensee's administrative controls were lacking, in that engineering assistance requests had not been resolved in a timely manner. This issue is discussed further in Section 2.4 of this report.

1.3 <u>High Pressure Safety Injection System</u>

1.3.1 System Operation and Testing

1.3.1.1 Surveillance Testing

The team verified that selected testing requirements were incorporated into surveillance procedures. To accomplish this verification, the team reviewed Surveillance Test Procedures OP-ST-SI-3007, "High Pressure Safety Injection Pump and Check Valve Test" and OP-ST-SI-3008, "Safety Injection and Containment Spray Pump Inservice Test and Valve Exercise Test," and their associated test results.

Procedure OP-ST-SI-3007 was prepared to meet the Technical Specification requirement that surveillance tests are performed every refueling outage to demonstrate the operation of the safety injection system. The test measured differential pressure, flow, and vibration for the three safety injection pumps. The team also reviewed the test results from the refueling outages that occurred on April 1992, November 1993, and April 1995.
Procedure OP-ST-SI-3008 was prepared to meet the Technical Specification inservice test requirement that the active components, pumps, and valves, are tested every three months to verify that the pumps are in satisfactory running condition. This test measured the differential pressure and vibration for the high pressure safety injection pumps, the low pressure safety injection pumps, and the containment spray pumps. The team reviewed the test results from January 1993 to the most recent test results.

The team noted that the vibration velocity measurements recorded for the low pressure safety injection pumps and containment spray pumps were higher than the values for the high pressure safety injection pumps. The licensee had prepared Engineering Assistance Request 94-014, dated July 15, 1994, to address this problem. As the result of this engineering assistance request, the licensee determined that they could not meet the inservice test requirements of the 1989 edition of ASME Section XI for vibration measurements for the low pressure safety injection pumps and the containment spray pumps when operated on minimum recirculation flow. The licensee requested and received NRC relief from these requirements on September 26, 1994. The licensee had proposed to test the low pressure safety injection and containment spray pumps quarterly using adjusted vibration velocity limits and to adhere to the code-specified limits during full flow tests when in a refueling outage. The team concluded that the licensee's surveillance tests were satisfactorily performed in accordance with Technical Specification and ASME Section XI requirements as modified by the licensee's relief request.

1.3.1.2 System Physical Condition

The team walked down the high pressure safety injection system using Operating Instruction OI-SI-1, "Safety Injection Normal," and Drawing E-23866-210-130, "Composite Flow Diagram Safety Injection and Containment Spray System." During these walkdowns, the team noted a broken handle on Valve HCV-306-B2 and boric acid crystals on Valves SI-111 and HCV-347, which was indicative of valve packing leaks. The licensee generated maintenance work orders to repair the valves. In addition, the team noted that a maintenance work request tag documented oil leaks on high pressure safety injection Pump SI-2A. The licensee also investigated this work request and determined that the tag no longer applied; therefore, the tag was removed.

The team also noted that the isolation valve for the air supply to Valves HCV-385 and 386 was listed in the operating procedure; however, the team could not locate the valve during the walkdown. The licensee investigated the missing valve and determined that the valve tag number had been changed to IA-3029 in 1989 by Engineering Change Notice 89-046. This change notice had revised the tag numbers of a number of instrument air valves. Operating Instruction OI-SI-01 was revised on October 14, 1991, to incorporate the tag number changes made under Engineering Change Notice 89-046. However, the procedure change failed to revise the tag for Valve IA-HCV-385/386-B2 to reflect the new tag number. The team reviewed the operating procedure checklist performed on April 8, 1995, and found that the valve Tag IA-HCV-385/386-B2 had been lined out and Tag IA-3029 had been written in and initialed by the person performing the checklist. The licensee took actions to correct Operating Instruction OI-SI-01.

1.3.2 System Reliability

1.3.2.1 Trending High Pressure Safety Injection Pumps

The team discussed trending with the inservice test engineer and determined that differential pressure, motor current, and vibration velocities were trended for the quarterly tests. In addition, the team determined that the licensee also trended flow, differential pressure, and vibration velocities during the full flow refueling outage tests for the high pressure safety injection pumps, low pressure safety injection pumps, and the containment spray pumps. The team noted that trending had begun in August 1993 and continued to the present. In addition, the licensee had trended the pump reference values, which included the high alert value and the high action value. The team also noted that the test data remained below the high alert and high action values. The team concluded that the test values had remained reasonably consistent, which indicated that the pumps were not degrading.

1.3.2.2 System Engineering Notebook

The team reviewed the safety injection system notebook. This notebook contained a list of tasks and their status, which included the due date, a listing of the engineering assistance requests initiated since 1989, a list of the engineering change notices, and a list of modifications. The work order histories for major components were also included in the system notebook. The team concluded that the safety injection system notebook provided good information that allowed the system engineer to be cognizant of the system status and any system problems, and provided information to management so that they were aware of the system status.

1.3.2.3 System Report Cards

The team reviewed the safety injection system report cards for the period October 1991 through July 1995. The report cards contained an assessment of the overall system performance and the repair backlog, system and equipment reliability concerns, major activities which occurred, and a list of significant items awaiting resolution. The team concluded that the contents of the system report cards provided management with an understanding of the health of this system.

1.3.3 Design Changes

1.3.3.1 Design Basis Review

As part of the review of Design Basis Document SDBD-SI-HP-132, the team requested a copy of the procedures that were used to verify the safety injection refueling water tank chemistry specified in Section 4.1.1.3 of the design basis document. The licensee stated that the water chemistry specified in the design basis document had been taken from a Combustion Engineering specification and was not being maintained, that this chemistry specification was not necessary, and that the specification was going to be deleted from the design basis document.

The team also noted that Section 4.1.2.3 and Attachment 4 of the design basis document, and updated safety analysis report Table 6.2-2 all listed the high pressure safety injection pump design temperature as 350°F. The team also noted that Bingham Pump Company Drawing A-37372X, Revision A, which was a drawing of the pump nameplate, listed the pump design temperature as 300°F. After these inaccuracies were identified by the team, the licensee initiated a review of the design parameters contained within this attachment.

The team also noted that Section 6.2.1 of the updated safety analysis report documented that the minimum available net-positive suction head for the high pressure safety injection pump was 41.51 feet. However, Table 6.2-2 in the updated safety analysis report and the high pressure safety injection design basis document listed the minimum available net positive suction head as 18 feet. The licensee stated that the calculated value of 41.51 feet was the correct value and that they would revise the updated safety analysis report and design basis document to reflect the correct values and delete Table 6.2-2 from the updated safety analysis report. The team considered these findings to be further examples of minor inaccuracies and discrepancies between the design basis documents and updated safety analysis report which is discussed in Section 2.3.1 of this report.

1.3.3.2 System Modifications and Calculations

The team reviewed one implemented temporary modification, four design modifications, which had not been implemented, and three calculations listed in Attachment 3 to this report.

Engineering Evaluation and Assistance Request (a predecessor to the modification request) MR-FC-84-143, dated August 22, 1984, was initiated by radiation protection personnel as a result of excessive personnel exposure problems identified while cleaning the safety injection refueling water storage tank during the 1984 refueling outage. The modification request asked for the addition of a drain line to aid in complete drain down of the tank. The response to the 1984 modification request was a memorandum dated March 22, 1993, which cancelled Request MR-FC-84-143. The request was cancelled because additional drain lines had already been installed during the 1992 refueling outage.

The team concluded that the reviewed calculations and modifications were adequate with the exception of an untimely response to Modification Request MR-FC-84-143. The team considered the response to this request to be untimely because it increased the potential of personnel exposure and contradicted the licensee's ALARA goals.

1.3.4 Other System Attributes

The team reviewed a number of maintenance work orders, incident reports, and engineering assistance requests. The team concluded that the maintenance work orders and the incident reports were adequate.

The team reviewed Engineering Assistance Request 93-021, dated February 1, 1993. This engineering assistance request was written to identify that the minimum temperature specified in the Technical Specifications for the safety injection refueling water tank was not the most limiting temperature. The engineering assistance request requested that calculations be reviewed to determine if other conditions, such as pressurized thermal shock of the reactor vessel, required a more limiting temperature. The engineering assistance request was closed out July 31, 1995. Attached to the engineering assistance request was a letter from ASEA Brown Boveri, dated June 29, 1993, which contained the response necessary to answer the engineering assistance request. While the team considered the response in 1993 to be timely, the 2-year period to formally close out the engineering assistance request seemed unreasonably long.

The team reviewed Engineering Assistance Request 91-113, dated October 23, 1991. The engineering assistance request was written to identify that the high pressure injection flow could be diverted from the core to the safety injection tanks by back leakage through the check valves. While this engineering assistance request was appropriately resolved, it was not resolved until 3 years later on July 13, 1995.

Based upon these reviews the team concluded that the licensee's administrative controls for the processing of engineering assistance requests were lacking in that no provision had been made to ensure timely closure. The team's concern regarding the timeliness of the closure of engineering assistance requests is further discussed in Section 2.4 of this report.

2 PRODUCTION ENGINEERING (37550)

2.1 Management Expectations

The team noted that the licensee had established management expectations for both the design and system engineers. However, the expectations for system engineers appeared to be more general than those for the design engineers.

The management expectations for design engineering were clearly defined and included:

- Identification of conditions before plant safety and reliability was affected;
- An absence of challenges to plant safety systems due to design engineering activities;
- Implementation of the Maintenance Rule;
- Completion goals for the 1996 refueling outage;
- Limiting outstanding plant modifications at the end of 1995 to a maximum of 68;
- Limiting outstanding engineering assistance requests at the end of 1995 to a maximum of 140; and,
- Improving the reliability of major plant systems.

The team noted that management expectations for the system engineers were not as clearly defined. For example, the expectation concerning system walkdowns did not define the frequency, level of detail, or define a method of documenting these walkdowns. In addition, the expectation did not define the contents of the system notebook and there was no requirement for system engineering notebooks to have the installation specifications so that an understanding of the original installation would be readily available. In response to this concern, the licensee indicated that they did not intend to define specific expectations for system engineers. The licensee's reasoning was that the lack of specific expectations allowed flexibility in judging the effectiveness of system engineers in the performance of their assigned duties.

The team concluded that the licensee had provided clear guidance to design engineers regarding management expectations. Although these expectations were not as clearly communicated to the system engineers, the team concluded they were effective based upon the team's observation of the system ownership and strong knowledge that these engineers had of their assigned systems.

2.2 System Engineering

2.2.1 System Status Awareness

To evaluate the system engineer's understanding of their systems, the team conducted walkdowns or physical inspections with selected system engineers. These walkdowns included the 125 Vdc power system, the diesel generator system, and the high pressure safety injection system. In all cases, the team concluded that the system engineers were aware of the status and were knowledgeable of the design and operation of their assigned systems. For example, when the team found checklist information inconsistent with plant hardware, the diesel generator system engineer readily understood the problem and how it occurred. Additionally, the team observed that the system engineers performed routine walkdowns of their systems at a periodicity that depended upon plant conditions and events. The team concluded that the system engineers demonstrated a sound knowledge and awareness of system problems and deficiencies and demonstrated a positive sense of ownership and responsibility.

The team noted that the system engineers did not have a list of the deficiency tags on their system, nor did they have ready access to such information. The licensee utilized these deficiency tags on plant equipment to identify a problem with the equipment. These tags were generally written for minor equipment problems that did not affect equipment operation (e.g., minor valve packing leakage). When more significant problems were identified, an incident report was written which were available on the plant computer system. In response to this concern, the licensee indicated that system engineers did not need a list of these deficiency tags, because they were expected to be aware of deficiencies on their systems based upon their frequent system walkdowns. The team did not identify any examples that contradicted the licensee's position.

2.2.2 System Risk Awareness

The team noted that the probabilistic risk assessment for plant systems was delineated in probabilistic risk assessment system notebooks. These notebooks provided risk insights and their effects upon a system's operation. However, the team noted that the diesel generator system engineer was not aware of the probabilistic risk assessment significance of the system's components and that the probabilistic risk assessment system notebooks had not been maintained.

Specifically, the probabilistic risk assessment system notebook on diesel generators had not been revised since June 15, 1992. Upon further review, the team noted that none of the system notebooks were maintained. The team also noted that the probablistic review assessment staff was aware of the condition, planned to revise the notebooks by the end of 1996, and limited the distribution of the current system notebooks. The team noted that much of the licensee's staff effort was involved with the recent submittal of the individual plant examination of external events; therefore they did not have the resources to maintain the system notebooks current.

2.2.3 System Report Cards

The team reviewed a sample of recently issued system report cards. The system engineers had issued these reports in a timely manner in accordance with the licensee's established guidance. However, half of the engineers interviewed felt that the report cards were not effective due to the fact that they had no indication that anyone other than their immediate supervisor were reviewing these reports and that many of the issues identified in the report cards remained open for extended periods of time. In response to this concern, the licensee indicated that, although they were reviewing the system report card data, they were not providing feedback to the system engineers regarding these reviews. The licensee stated that they would review this issue and establish a method of providing feedback to the system engineers regarding the results of their report card reviews.

2.3 Design Engineering

2.3.1 Design Input Information

As previously identified during the system reviews, the team noted a number of inaccuracies in the design input information. Specifically, there were several inaccuracies identified between the design basis document and the updated safety analysis report. Although none of these discrepancies had any affect on the operability of equipment, the licensee took action to resolve these discrepancies as they were identified. In addition, the team noted that the licensee was generally aware of inaccuracies within the design basis documents and had established a program as described in Quality Procedure QP-13, "Design Basis Document Control," to identify and correct the inaccuracies in the design basis document. The team reviewed this program and concluded that many inaccuracies had been corrected, that the program appeared to be effective in correcting the design basis document inaccuracies, and that there was good progress in reducing the number of these inaccuracies.

2.3.2 Engineering Change Process

The licensee performs plant design changes using permanent plant modifications, temporary plant modifications, or engineering change notices. As documented in Sections 1 and 4 of this report, the team reviewed a number of plant design changes to determine the effectiveness of this process.

As previously discussed, the diesel generator cooling was modified (Section 1.2.1.3) by changing the number of radiator fan blades from 8 to 12, changing the blade pitch, and changing the fan blade-to-fan housing clearances. This modification was accomplished in accordance with Engineering Change Notice 91-306. During the review of this modification, the team noted that this modification was performed as a substitute replacement item in accordance with Production Engineering Procedure PED-GEI-60, "Substitute Replacement Item Evaluations," in lieu of a permanent modification. The substitute replacement item process fit between the "like-for-like" replacement process and the permanent modification process. The "like-for-like" replacement process does not require a 10 CFR 50.59 safety evaluation for the design phase because parts are replaced with their identical copies, whereas the permanent modification process requires a safety evaluation because an actual system change is being performed that may affect the plant's design basis. The substitute replacement process allowed replacement of parts that were not "like-for-like," but were sufficiently similar such that the plant's design basis was not affected. This process required a comparison analysis to be made to assure that the replacement part did not affect the plant's design basis. In addition, this process did not require the performance of a 10 CFR 50.59 safety evaluation for the design phase, but did require a safety evaluation for the installation and testing phases. The substitute replacement item process made the plant change process simpler and less time consuming and, therefore, appeared to have become the process of choice for the licensee in the performance of plant modifications.

The team's review of this engineering change notice indicated that the change made to the diesel generator radiators was considerably more than a simple part replacement evolution. For example, this change involved a change to the plant's updated safety analysis report, which indicated that the change did involve a change to the plant's design basis. Despite the procedure requirements, the team also noted that a 10 CFR 50.59 safety evaluation was performed as a part of this substitute replacement item process and that this safety evaluation identified the fact that a updated safety analysis report change was needed.

As the result of this finding, the team expanded their review of engineering change notices that involved the use of the substitute replacement item process and determined that these engineering change notices were performed without the required safety evaluations for the installation and testing phases that were required by Procedure PED-GEI-60.

- ECN 93-152, Diesel Generator Starting Air Drain Lines and Valves;
- ECN 93-237, Replace HCV-265 Valve Stem with Longer Stem;
- ECN 93-379, Wide Range Nuclear Instrumentation Channel D Power Supply Replacement;
- ECN 93-488, Substitute Replacement for Disc in (valve) MS-292;
- ECN 93-600, Alternate Materials for High Pressure Safety Injection Isolation Valves;

- ECN 93-631, Emergency Feedwater Storage Tank Level Indication; and,
- ECN 95-129, Repair Parts for (valve) HCV-2908-0.

The team also noted that since there was no safety evaluation performed, these modifications did not receive the plant review committee review that would be required for other plant modifications prior to implementation.

Title 10 CFR 50.59 requires that the licensee maintain records of changes to a facility that includes a written safety evaluation. This safety evaluation must provide a basis to determine whether the change involves a change to the plant's design basis. The use of substitute replacement items to perform plant modifications either with a safety evaluation that resulted in a change to the plant's design basis or were performed without a safety evaluation was not in accordance with the requirements of Procedure PED-GEI-60 and is considered to be a violation of 10 CFR 50.59. (285/9511-01).

2.3.3 Design Change Packages and Calculations

2.3.3.1 Calculations

The team reviewed a sample of ten calculations and verified the consistency of the licensee's assumptions and clarity of logic. In general, the calculations were considered to be satisfactory. However, the team did identify an inconsistency between calculations. For example, Calculation FC05916 established a maximum operating temperature for Diesel Generator No. 1 as 107°F and for Diesel Generator No. 2 as 114°F. The licensee based these operating temperatures on recent hot weather testing results. However, Calculation FC05491 for the diesel generator air starting system established a maximum operating temperature for both diesel generators of 110°F. Engineering based this temperature on the most limiting condition for the system which was the melting point of solder joints in the air lines. Calculation FC05916 also assumed that the Diesel Generator No. 2 room would heat up 16°F in 1 hour when the diesel generator was running. When the team applied the room heatup assumption to Calculation FC05491, it appeared that the maximum operating temperature of the diesel generators was limited to a maximum ambient temperature of 94°F (110°F less 16°F). The licensee documented this finding in Incident Report 950569 and performed an operability evaluation. The licensee's evaluation concluded that there was no operability concern. The team reviewed the licensee's evaluation and concurred that the assumption that solder joints would fail at 110°F was unduly conservative.

During the review of calculations the team also identified an error in the design basis document for the diesel generators. The diesel generator diesel basis document documented that there were two independent air starting systems for each diesel generator. The team's review indicated that the primary and secondary air start systems shared a common water jacket low pressure switch, PS-13, and several control relays. The team also noted that the updated safety analysis report described the air start systems as duplicate systems and that the system training manual documented that a single failure in the air starting system would not prevent the diesel generators from starting.

When informed of this design basis document error by the team, the licensee initiated actions to revise both the design basis document and the training manual. The team considered this finding to be another example of minor inaccuracies and discrepancies between the design basis documents and updated safety analysis report which is discussed in Section 2.3.1 of this report.

2.4 Engineering Backlogs

The team reviewed the engineering backlogs of Modification Requests, engineering change notice, Incident Reports, and engineering assistance requests. During the last 2 years, the licensee's backlog reduction efforts included the following:

- Completion of 5 modifications with 23 modifications in process or preparation;
- Issued 517 document change engineering change notices of which 68 remained open;
- Issued 209 facility change engineering change notices of which 152 remained open;
- Issued 470 substitute replacement item engineering change notices of which 246 were open; and,
- Issued 869 incident reports of which 82 remained open.

The team noted that while the backlogs for engineering change notice related to facility changes and substitute replacement items were higher than the other backlogs, there was a decreasing trend. The team considered these results to be indicative of the licensee's efforts to maintain a low engineering backlog.

As previously discussed in this report, the team found that the licensee had not closed all engineering assistance requests in a timely manner. To determine the extent of this problem, the team selected several additional engineering assistance requests for review. This review identified several additional examples where engineering assistance request responses, reviews, or formal closure were not timely.

For example, EAR 91-071, which involved the diesel generator's low temperature operability limits, was issued on July 19, 1991 and received a response on December 13, 1991. The system engineer did not review this response, which left the engineering assistance request open. Further review of this issue by the team identified an error in the diesel generator lubricating oil low temperature alarm setpoint. Specifically, the December 13 response identified that the vendor recommended a temperature alarm setpoint of 85°F. The team noted that the lubricating oil temperature alarm setpoint was currently set at 80°F. The purpose of the 85°F temperature alarm was to alert operators that the diesel generator lubricating oil was cooling down. A cooldown below 85°F had the potential of affecting the diesel generator fast start capability. The lubricating oil temperature was maintained by the diesel generator water jacket keep-warm system. This system maintained the jacket water temperature at between 100°F and 110°F. A review of the vendor's design documentation by the team indicated that by maintaining the jacket water temperature in this range, the lubricating oil temperature would not decrease below the 85°F limit, therefore, it appeared that lubricating oil alarm setpoint error did not impact diesel generator operation. As the result of this finding, the system engineer issued a procedure change request for each diesel (IC-ST-DG-0005 and IC-ST-DG-0045) that changed the lubricating oil temperature alarm setpoint to 85°F.

As the result of the team's concern, the licensee issued an incident report to determine the reasons that these engineering assistance requests had not been resolved in a more timely manner. As of the close of this inspection, this incident report had not yet been resolved.

In order to evaluate the licensee's controls for processing engineering assistance requests, the team reviewed Standing Order SO-G-82, "Engineering Assistance Requests." Based upon a review of this procedure, the team concluded that the licensee's administrative controls for the processing of engineering assistance requests were lacking, in that the procedure did not have provisions to ensure timely closure of engineering assistance requests. The team had reviewed 50 engineering assistance requests and identified problems with 34 of these requests. However, the team did not identify any examples where untimely responses adversely affected equipment operation. Nevertheless, additional effort is needed to strengthen the administrative controls for engineering assistance requests.

2.5 <u>Training</u>

During the inspection, the licensee indicated that they were planning to implement a new training and qualification program effort. These new guidelines provided for a core training schedule and a position specific training schedule which resulted in more flexible personnel assignments. The team reviewed the current training requirements found in the applicable training program master plans. This review included the training and qualification requirements and the associated records. The team noted that the licensee had established requirements for each position (e.g., system engineer or design engineer). The team noted that the system engineers had received all of their required training. The team considered this training to be a contributor to the high level of knowledge that these engineers exhibited toward their systems. However, the team also noted that the training for design engineers was lagging especially in the electrical and reactor engineering areas.

2.6 <u>Review of Engineering Response to Specific Issues</u>

To determine whether engineering was adequately identifying and addressing emerging technical issues, the team reviewed the actions that engineering had taken on two specific issues, the response to a pipe erosion event that occurred at the Millstone Nuclear Power Station on August 8, 1995, and the development of an air operated valve diagnostic testing program.

On August 8, 1995, the Millstone Nuclear Station experienced a pipe rupture in a main steam heater drain recirculation line. This line was not a part of the Millstone licensee's pipe erosion/corrosion program because this line was not used during full-power operation. However, this line was used during transient plant conditions, such as plant startup and shutdown. The team determined that the licensee was aware of the event and had already determined that their erosion/corrosion program encompassed similar piping conditions at the Fort Calhoun Station. Due to the licensee's use of "modeling" in the licensee's erosion/corrosion computer program (CHECKWORKS), the licensee had identified plant piping that was subjected to flow conditions only on an intermittent basis and had included such piping in their monitoring program. Furthermore, the team noted that the licensee had already examined portions of this piping during their last plant outage. The team considered the licensee's response to be indicative of a proactive response and their erosion/corrosion program to be a strength.

Due to the high number of air operated valves at the Fort Calhoun Station, the team considered the establishment of an air operated valve diagnostic testing program to be an important safety enhancement. Approximately 2 to 3 years ago, the licensee acquired an AIRCET valve diagnostic system that could be used to perform such diagnostic testing on air operated valves. While such a system was available at the site, the licensee had only recently begun developing a program to perform such diagnostic testing. At the time of this inspection, the licensee had developed calibration procedures for the test equipment and had completed training of instrumentation and control technicians to perform the calibrations. The licensee was in the process of developing actual testing procedures; therefore, the diagnostic process had not been implemented. Additionally, the team noted that the extent of the licensee's test program was to only perform valve troubleshooting and not to perform periodic in-service testing on the valves. The team was concerned, that, based upon the high reliance of the Fort Calhoun Station on air operated valves, the licensee's efforts to develop a diagnostic program that the licensee intended to maintain only a reactive program. The licensee acknowledged this observation, but concluded that their efforts were acceptable.

3 NUCLEAR SAFETY REVIEW GROUP (37550)

The nuclear safety review group was the licensee's independent safety engineering group. The purpose of this group was to provide an independent review of selected plant activities related to nuclear safety, investigate plant events, develop recommendations to preclude incidents involving nuclear safety, and advise licensee management and/or the licensee's offsite review committee on the overall quality of nuclear safety. To accomplish this mission, the nuclear safety review group reviewed in-house and industry operating experience information and 10 CFR 50.59 evaluations as assigned by the off-site review committee. The nuclear safety review group also performed, through either management direction or self-initiation, root cause analyses, special investigations, special reviews or assessments. The nuclear safety review group also assisted as team members on special licensee directed self-assessment teams and quality assurance audit teams, and conducted periodic plant tours.

The team evaluated the effectiveness of the nuclear safety review group by reviewing selected nuclear safety review group reports, interviewing nuclear safety review group members, and reviewing the implementation of the corrective actions recommended by this group. This review encompassed 12 nuclear safety review group assessment reports and the corrective action recommendations from these reports. These assessment reports were found to be thorough, detailed, and complete. In most cases the reports had numerous corrective action recommendations. The team noted that these recommendations properly addressed the weaknesses identified during the assessments. The team then reviewed selected recommendations to determine whether these recommendations were being implemented in a timely manner. Of the eight multiple part recommendations reviewed, all but one recommendation were completed on-time. The one recommendation that was not completed had its completion date extended. The team reviewed the reason for the date extension and considered the reason to be to be appropriate. Based upon these reviews, the team concluded that the activities of the nuclear safety review group were effective, were identifying substantial issues, and were successful in accomplishing their goals. The nuclear safety review group was considered to be a strength.

4 SAFETY EVALUATION PROGRAM, 10 CFR 50.59 (37001)

4.1 Screenings and Evaluations

The team reviewed the 10 CFR 50.59 nuclear safety evaluations for plant modifications, temporary modifications, engineering change notices, and procedure changes performed in accordance with Quality Procedure NOD-QP-3, "10 CFR 50.59 Safety Evaluations." The purpose of this review was to determine if these safety evaluations conformed with the 10 CFR 50.59 safety evaluation requirements. The team reviewed six modifications, three temporary modifications, eleven engineering change notices, and three procedure changes. Procedure NOD-QP-3 provided the guidance for completion of safety evaluation forms. This was accomplished by answering questions in Sections 9.0 through 10.0 of the safety evaluation forms.

4.2 Modification Safety Evaluations

The inspection team performed a detailed review of three modification requests (FC-94-012, dated March 8, 1995; FC-90-070 dated July 1, 1993; and FC-92-044 dated August 12, 1994) to determine whether the licensee was adequately implementing the provisions of 10 CFR 50.59. In each example, the team found minor administrative discrepancies during the performance of the safety evaluations involving incomplete answers on the safety evaluation form or confusing statements. None of these examples adversely affected the performance of the safety evaluation.

Nevertheless, these errors indicated a lack of attention to detail by personnel performing the reviews. This lack of attention to detail in the performance of 10 CFR 50.59 safety evaluations was previously identified in NRC Inspection Report 285/93-08 and was considered as a noncited violation. Subsequent to this finding, the licensee issued Corrective Action Report 93-209 to identify and implement corrective actions to increase personnel awareness in their performance of such safety evaluations. In addition, in February 1994, the nuclear safety review group conducted an audit (94-151) of this issue. This audit identified that the corrective actions implemented by Corrective Action Report 93-209 were not effective and that similar problems with the performance of these safety evaluations were still occurring. The team concluded that additional effort is needed to improve the administrative performance of safety evaluations.

5 FOLLOWUP ON RELATED ENGINEERING ISSUES (92903)

5.1 Service Water Self Assessment

During the period of October 24 through November 11, 1994, the licensee conducted a self assessment of their service water system. The results of this self assessment were reviewed by the NRC and documented in NRC Inspection Report 50-285/94-04. NRC Inspection Report 50-285/94-04 identified issues that needed further licensee consideration. To determine the response of the engineering organization to these issues, the team reviewed the status of the following selected issues:

- Effect of a component cooling pipe break in the auxiliary building.
- Leakage from the raw water system to the component cooling system through the system interface valves.
- Calculational weaknesses regarding raw water flow to the component cooling water heat exchangers.
- Weaknesses in the heat exchanger heat transfer capability test program.
- Identification of the limiting single failures for the maximum component cooling water return temperature, and
- Deficiencies in the air check valve testing methodology.

The team's review indicated that four of these six issues had been resolved and closed. The remaining two issues, the heat exchanger transfer capability testing and raw water to component cooling water interface valve leakage were still open. The team noted that resolution for these two issues was being actively pursued and tracked by engineering.

5.2 Violation Closeout

(Closed) Violation 285/9415-01: Failure to specify appropriate post-modification test requirements

Background

This violation identified that the licensee failed to provide written instructions for the conduct and documentation of post-modification vibration testing. This violation occurred after installation of a flow orifice in the raw water line to the component cooling heater heat exchanger in accordance with Temporary Modification 94-007.

Followup

The licensee's corrective actions included revising Standing Order M-101 and Maintenance Department Instruction No. 7 and the training of engineers and work planners on the revised procedures. The revisions to the procedures strengthened the review process for changes to maintenance work orders. The licensee had training for engineers and planners on the revised procedures.

The team reviewed the latest revisions of Standing Order M-101 and Maintenance Department Instruction No. 7 to verify that the changes identified as a part of the licensee's corrective actions were implemented. The team also verified, by review of training records, that all applicable engineers and work planners had received the appropriate training. Additionally the team reviewed a sampling of maintenance work orders and did not identify any further problems with post-maintenance testing activities.

Conclusion

The team concluded that the licensee completed all the corrective actions documented in their response to this violation and considered the violation to be closed.

ATTACHMENT 1

Persons Contacted and Exit Meeting

1 PERSONS CONTACTED

1.1 Licensee Personnel

J. Adams, Design Engineer
R. Andrews, Division Manager, Nuclear Services
C. Boughter, Supervisor, Special Services Engineering
J. Brown, Shift Supervisor, Operations
G. Cavanaugh, Station Licensing
R. Conner, Assistant Plant Manager
G. Cook, Supervisor, Station Licensing
J. Connolley, Lead Test and Performance Engineer
W. Gates, Vice President, Nuclear
J. Gasper, Manager, Training
D. Gorence, Lead System Engineer, Secondary Systems
A. Gurtis, Senior Quality Assurance Engineer
L. Kusek, Manager, Nuclear Safety Review Group
D. Latkin, Nuclear Safety Review Group Specialist
J. O'Conner, Manager, Electrical Design Engineering
W. Orr, Manager, Quality Assurance and Quality Control
T. Patterson, Division Manager, Nuclear Operations
R. Phelps, Acting Division Manager, Production Engineering
J. Ressler, Nuclear Design Engineer
A. Richard, Manager, Mechanical Design Engineering
J. Skiles, Acting Manager, Design Engineering
M. Tesar, Manager, Corrective Action Group
D. Trausch, Manager, Licensing and Industry Affairs

1.2 Southern California Edison Personnel

D. Axline, Licensing

1.3 NRC Personnel

S. Bloom, Fort Calhoun Station Project Manager, Nuclear Reactor Regulation
P. Gage, Reactor Inspector, Engineering Branch, Region IV
P. Goldberg, Reactor Inspector, Engineering Branch, Region IV
T. Gwynn, Director, Division of Reactor Safety, Region IV
W. McNeill, Reactor Inspector, Engineering Branch, Region IV
R. Mullikin, Reactor Inspector, Engineering Branch, Region IV

The above personnel attended the exit meeting. In addition to these personnel, the team contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on July 25, 1995. During this meeting, the team reviewed the scope and findings of the report. The licensee did not express a position on the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the team.

ATTACHMENT 2

INSPECTION FINDING INDEX

- Violation (285/9511-01) was opened.
- Violation (285/9415-01) was closed.

ATTACHMENT 3

LIST OF DOCUMENTS REVIEWED

PLANT DRAWINGS

- Drawing 11405-E-8, Sheets 1 & 2, 125VDC Miscellaneous Power Distribution Diagram, Revisions 47 and 4
- Drawing 136B2570, Sheets 8 & 9, Elementary Diagram Instrument Bus Scheme, Revision 13
- Drawing 2C6288, D.C. Distribution Schematic EE-8F, Revision 1
- Drawing 2C6289, D.C. Distribution Schematic EE-8G, Revision 1
- Drawing 531-227-61, Overall Schematic Inverter INV. 103-1-102, Revision 4
- Drawing 531-227-75, Sheet 1, Static Inverter Installation Drawing INV 103-1-102, Revision 3
- Drawing 642-213-40, Inverter Circuit Breaker Assembly, Revision E
- Drawing 6511031, Overall Inverter Schematic, Revision 0
- Drawing 7511031, Sheet 1, Inverter Installation Drawing INV. 752-1-101F, Revision 1
- Drawing D-55-1610S, Schematic 125 VDC Battery Charger, Revision 2
- Drawing E-23866-210-130, Composite Flow Diagram Safety Injection and Containment Spray System, Revision 5
- Figure 8.1-1, Simplified One Line Diagram Plant - Electrical System, Revision 91

DESIGN BASIS DOCUMENTS

- SDBD-DG-112, Emergency Diesel Generators, Revision 12
- SDBD-EE-202, DC Distribution, Revision 8
- SDBD-SI-HP-132, High Pressure Safety Injection, Revision 6

SYSTEM NOTEBOOKS

- 649-20-12, Electric Power System Analysis Notebook, Revision 2
- Safety Injection System Notebook

SYSTEMS TRAINING MANUALS

- Electrical Distribution System, Volume 14
- Emergency Core Cooling, Volume 15, Revision 9
- Emergency Diesel Generators, Volume 16, Revision 8

CALCULATIONS

- FC05372, Allowable Time for Operating HPSI and LPSI Pumps in a Dead-Head Mode, Revision 0
- FC03382, Diesel Generator LOCA Loads, Revision 8
- FC05384, Minimum Pump Performance Curves for HPSI, LPSI, and CS Pumps, Revision A
- FC05491, Design and Operation Conditions for the Diesel Generator Starting Air System, Revision 1

- FC05602, Amount of Hydrogen Produced by the Station Batteries in the Battery Rooms
- FC05603, DC Short Circuit Current Calculation of Various Points of DC Bus 1 and DC Bus 2
- FC05690, Battery Load Profile and Capacity Calculation
- FC05827, 125 VDC System Voltage Drop Study
- FC05863, Diesel Generator Rooms 63, 64, 65 Hydraulic Demand Calculation, Revision 1
- FC05910, 125 VDC Short Circuit/Coordination Study
- FC05916, Operating Temperature Limits for DG-1 and DG-2, Revision 2
- FC06282, Tank Curves for Diesel Fuel Oil Storage Tank (FO-1), Auxiliary Boiler Fuel Oil Tank (FO-10), Fuel Oil Storage Tank (FO-32) [buried].and Fuel Oil Day Tank (FO-33), Revision 0
- FC06289, Diesel Generator Fuel Oil Storage Tanks Level TLU Calculation, Revision 0
- FC06298, Diesel Generator Load Indication TLU and EOP Impact, Revision 0
- FC06296, Total Loop Uncertainty Calculation for HPSI Flow Indication, Revision 0

PROCEDURES

- EM-ST-EE-0002, Monthly Surveillance Test for Station Battery No. 1 (EE-8A), June and July, 1995
- EM-ST-EE-0002, Monthly Surveillance Test for Station Battery No. 2 (EE-8B), June and July, 1995
- EM-ST-EE-0003, Quarterly Surveillance Test for Station Battery No. 1 (EE-8A), March and June 1995
- EM-ST-EE-0004, Quarterly Surveillance Test for Station Battery No. 2 (EE-8B), March and June 1995
- EM-ST-EE-0009, Monthly Surveillance Test for Station Battery Chargers, June and July, 1995
- PED-GEI-60, Substitute Replacement Item Evaluations, Revision 4, June 23, 1995
- IC-ST-SA-3001, Starting Air Compressors Discharge Check Valves Exercise Test, Revision 7
- Maintenance Department Instruction MDI-7, Maintenance Planner's Instructions, Revision 17
- OI-ES-2, Engineered Safeguards Controls - Cold Shutdown Operation, Revision 3
- OI-EE-2B, 480 Volt Hot Bus Transfer, Revision 6
- OP-3A, Plant Shutdown, Revision 6
- OP-FT-DG-0001, 183 Master Electrical Switch Functional Test Revision 1
- OP-ST-DG-0001, Diesel Generator 1 Check, Revision 12
- OP-ST-ESF-0001, Diesel Auto Start Initiating Circuit Check, Revision 7
- OP-ST-FO-3002, 2 Fuel Oil System Pump Inservice Test, Revision 8
- OP-ST-SI-3007, High Pressure Safety Injection Pump and Check Valve Test, Revision 8
- OP-ST-SI-3008, Safety Injection and Containment Spray Pump Inservice Test and Valve Exercise Test, Revision 18
- OI-DG-1, Diesel Generator No. 1 (DG 1) Normal Operations, Revision 15

- OI-DG-2, Diesel Generator No. 2 (DG 2) Normal Operations, Revision 18
- OI-EE-3, 125 VDC System Normal Operation, Revision 10
- OI-SI-1, Safety Injection Normal, Revision 24
- Preventive Maintenance Procedure SE-PM-EX-1600, Infrared Thermographic Surveys, Revision 1
- NOD-QP-01, Engineering Assistance Requests, Revision 5
- NOD-QP-02, Configuration Change Control, Revision 18
- NOD-QP-03, 10 CFR 50.59 Safety Evaluations, Revision 14, November 1, 1994
- NOD-QP-13, Design Basis Document Control, Revision 3
- NOD-QP-14, Use of Engineering Judgement, Revision 2
- SP-CP-08-OPLS-TD, Calibration of Timing Relays, Revision 2
- SO-G-7, Operating Manual, Revision 29
- SO-G-21, Modification Control, Revision 57
- SO-O-25, Temporary Modification Control, Revision 48, November 3, 1994
- SO-O-26, Plant Keys, Revision 26
- SO-G-30, Procedure Changes and Generation, Revision 62
- SO-G-82, Engineering Assistance Requests, Revision 2
- SO-M-101, Maintenance Work Control, Revision 33
- SEI-03, Acoustic Monitoring and Trending of Check Valves, Revision 1
- SEI-08, Performance Monitoring and Trending, Revision 2
- SEI-11, Trend Monitoring and Analysis, Revision 1
- SEI-13, Preventive Maintenance Program, Revision 4
- SEI-14, Surveillance Testing, Revision 3
- SEI-19, System Review, Trending and Reporting, Revision 5 with Procedure Change Notices 1 and 2
- SEI-25, Project Instructions, Revision 2
- SEI-28, Program Instructions, Revision 2
- TBD-AOP-32, Loss of 4160 Volt or 480 Volt Bus Power, Revision 2

INCIDENT REPORTS

- 930068, Valve opened when system pressure was applied
- 930116, Trouble alarm on Inverter 2
- 930144, Downward trend in specific gravity of 2 battery cells
- 930224, Unable to place pump switch in pull to lock
- 930261, Manufacturing cold solder joint fuse failure
- 930271, Large amount of water found in valve diaphragm housing
- 930294, Broken disc in isolation valve
- 930295, Valve disc replaced without proper paperwork
- 930339, Hydraulic snubbers failed
- 940078, Two Battery 2 cells with broken electrolyte withdrawal tubes
- 940114, Administrative issue
- 940159, Measure inverter total harmonic distortion
- 940327, HPSI discharge valve would not close
- 940409, Inverter B trouble alarm received in control room
- 940411, Failure to notify quality control during maintenance
- 950016, DC grounds received during maintenance
- 950035, Trouble alarm received for Inverter 2
- 950038, Trouble alarm received for Inverter 2
- 950046, Missing jumper on new DC circuit board

- 950061, Control room alarms received during maintenance due to personnel error
- 950065, Crack found on valve body to bonnet weld
- 950080, Label discrepancy on drawing
- 950108, Request for setpoint change
- 950118, Pump discharge valve found in open position
- 950177, Incorrect studs found
- 950351, Air intensifier would not reseat
- 950402, Trouble alarm received for Inverter D
- 950456, Inverter bypass transformer trouble
- 950532, Diesel Generator Ambient Temperature Limit, Revision 0
- 950550, Diesel generator loading curve, August 11, 1995

ENGINEERING ASSISTANCE REQUESTS

- 89-005, Fault current values
- 89-040, ST-ISI-NG-4 Design vise System Pressure
- 89-045, EFWST Tank Curves
- 90-003, Mod Request for Limitorque Operator
- 90-008, Evaluate NSRG concerns in document review
- 90-084, Resolve nuclear instrument inaccuracy
- 90-085, Develop additional graphs in OI-RC-12
- 90-093, Develop Discriminator threshold, read, and adjust procedure
- 90-094, Calibration upgrade for transmitter A, B, C, D./PT-120
- 90-103, Discrepancy between vendor for wide range log NLW-3
- 90-120, Develop procedure for power ratio calculator drawer
- 90-157, RPS Setpoint Margin Requirements
- 90-165, EOP-04 Step 3.18 steam generator tube rupture revision
- 91-023, Annunciator for EFWST always in alarm
- 91-034, Double sided foam tape in electrical panels
- 91-071, Diesel Generator Low Temperature Operability Limits
- 91-113, HPSI injection flow diverted to safety injection tanks
- 91-116, Frequent failure of valve
- 91-128, Diesel generator damper drawings
- 91-133, Potential Stem Thrust Problems with HCV-1384
- 91-136, Bypass of SIAS function during refueling outage
- 91-148, CCW heat exchanger RW isolation valves
- 92-009, Main Feedwater Valve Leakage Impact on MSLB Analysis
- 92-012, Steam Generator Level Correction Curves
- 92-073, Feed Rate of Steam Generator Following a Loss of Feedwater, Revision 1
- 92-076, Born Dilution of the RCS-Post SGTR
- 92-081, CIAS signal to HCV-881 and HCV-882
- 92-082, CIAS signal to HCV-883A and HCV-884A
- 92-088, Conduit fittings
- 92-117, CIAS signal to HCV-2983
- 92-145, Analysis to qualify CS pump suction lines
- 92-145, CS Pumps for SDC
- 92-146, Cooldown of a Steam Generator after a Tube Rupture, Revision 1
- 92-222, CET validation method
- 93-021, Limiting temperature for SIRWT
- 93-033, Containment purge flow indicators

- 93-042. Nonconservative low temperature overpressure setpoint
- 93-058. Fuse information
- 93-083. Engineering analysis to support various electric configurations
- 93-100. Reactor coolant pumps lube oil reservoir level alarm setpoints
- 93-118. Investacation of permanent reactivity computer
- 93-124. Test switch wiring/drawing error
- 93-131. Evaluation of continuous audio count rate speaker requirement
- 93-161. PIC-1555 indicator divisions
- 93-169. Resolution of out-of-tolerance instrument power supplies
- 93-180. PDIL and PPDIL value limits
- 94-014. Code exemption relief request
- 94-019. Contingency actions for loss of DC bus circuit breakers
- 94-023. Low Range Flow Indications for Feedwater Feedring
- 94-034. Update of drawing and verification of work performed
- 94-038. Engineering evaluation of hanger
- 94-045. Review and approval of qualification report
- 94-050. Evaluation of HGA Failures
- 94-054. Total harmonic distortion reduction
- 94-058. Low Steam Generator Pressure Pretrip Setpoint
- 94-065. The Tale of the Missing Rods
- 94-080. Steam Generator and Reactor Coolant Pump Snubber Replacement Struts
- 94-135. Provide a study area for operations personnel
- 94-142. Additional Fuel Oil Storage
- 95-006. Battery charger high voltage
- 95-044. Fire Pump Operability
- 95-068. Evaluate design conditions for air operated valve

MODIFICATIONS

- ECN-91-306. Emergency Diesel Generator Radiator Fan Replacement. November 1993
- ECN-92-225. Change SIRWT Low Temperature Alarm
- ECN-92-471. Diesel Generator Pressure Switch Replacement
- ECN-93-076. Inverter Input Circuit Breaker Replacement
- ECN-93-086. Revision of Temporary Modification TM-92-066 to Permanent Status. August 1994
- ECN-93-130. NonSafety-Related Inverters DC Input Breaker Replacement
- ECN-93-152. DG Starting Air Drain Lines and Valves. August 1994
- ECN-93-237. Replace HCV-265 Valve Stem With Longer Stem. March 1994
- ECN-93-379. Wide Range Nuclear Instrumentation Channel D Power Supply Replacement. September 1993
- ECN-93-488. Substitute Replacement for Disc in MS-292. April 1994
- ECN-93-577. Flow Orifice in CCW Inlet Lines to RCP Thermal Barrier. January 1994
- ECN-93-600. Alternate Materials for HPSI Isolation Valves. June 1995
- ECN-93-631. Emergency Feedwater Storage Tank Level Indication. August 1994
- ECN-94-048. Sizing of RW/CCW Heat Exchanger Orifice Plates. August 1994
- ECN-94-404. DG Engine Heater Contactor Replacement
- ECN-95-120. Repair Parts For HCV-2908-O. July 1995
- ECN-95-263. Waterproof HCV-2987 Junction Box

- ECN-95-302, DG inlet air damper instrument air drawing
- MR-FC-84-143, Safety Injection Refueling Water (SIRWT) Tank Cleaning
- MR-FC-87-030, Containment Pressure Control Switch Replacement, May 2, 1994
- MR-FC-90-031, Diesel Generator 10 Second Start Recorder
- MR-FC-90-070, ILRT Vent Pass, July 1, 1993
- MR-FC-92-001, SIRWT Filtration
- MR-FC-92-035, Shutdown Cooling High Flow Alarm, August 12, 1994
- MR-FC-92-044, Simplification of the Engineered Safeguards System, Revision 10
- MR-FC-93-015, Modification to PORV Loop Seal Configuration, October 23, 1993
- MR-FC-94-002, Replace MCC Breakers on DC Buses
- MR-FC-94-012, Secondary Chemical Feed and Straight Injection and Tote Relocation Modification, March 8, 1995
- TM-93-005, Local indication for BAST CH-11A and CH-11B, January 13, 1993
- TM-93-080, Installation of Voltmeters on the CRDM Ground Detection System, November 24, 1993
- TM-94-039, Install a Second Pilot Valve in Control System for Intensifier
- TM-95-007, Removal of Turbine Building Crane (HE-3) North-South Travel Limit Switch Arms, February 22, 1995
- TM-95-035, Temporary Use of TE-121H as Input to RPS Channel D

MAINTENANCE WORK ORDERS

- 922345, DG-1 Hot Weather Testing, Revision 1
- 922870, DG-2 Hot Weather Testing, Revision 1
- 923695, SI valve had water on cap and boric acid crystals
- 930434, Valve would not operate due to leaking intensifier
- 930970, Inspect seat and disc of valve
- 932107, 932108, 932603, 932605, Routine troubleshooting for DC bus grounds
- 932449, Intensifier pressure low
- 933059, Valve would not close
- 933131, 940595, 941133, 950892, 951067, Troubleshoot causes of DC bus ground alarms received
- 933149, Valve would not open on ESF signal
- 933802, Corroded actuator parts
- 940092, Valve operator had to be reset
- 940166, Replaced parts in relief valve
- 940569, Union on water line had a leak
- 941859, Pump leaking some oil
- 942086, Intensifier not maintaining pressure
- 942458, Pump leaking some oil
- 950184, Valve oiler could not be adjusted
- 950638, Intensifier stroking once every 30 seconds
- 950750, Boric acid accumulation around valve
- 951334, Intensifier rebuilt
- 951790, Intensifier continually stroking

NUCLEAR SAFETY REVIEW GROUP ASSESSMENTS

- SRG-93-137, Assessment of Configuration Control, July 1993
- SRG-93-198, Assessment of the Fuse Control Program, October 1993
- SRG-93-249, Steam Generator Assessment Checklist, October 1993
- SRG-93-297, Perform an Operator Work Around Assessment Starting on November 1 Through December 6, December 1993
- SRG-94-039, Service Water Systems, March 1994
- SRG-94-078, Special Review, Independent Verification, June 1994
- SRG-94-092, Special Review of Service Water Systems Actions Items List, July 1994
- SRG-94-122, Assist in the Performance of an NSRG Group Assessment of the Overall PED Modification Process, October 1994
- SRG-94-122, NSRG Modification Process Assessment, Revision 0
- SRG-94-131, Quality Assurance Plan/Procedure Review, QAP 7.2, Material Identification and Control, Revision 3, September 1994
- SRG-94-140, Fast Transfer Logic for New Off-Site Power Lines
- SRG-94-151, Effectiveness of Corrective Actions for CAR 93-209, November 1994
- SRG-94-162, Nuclear Safety Review Group Assessment of Operator Work Arounds, December 1994
- SRG-95-018, Oversight of the Test Engineering Group, March 1995
- SRG-95-036, Follow-up Assessment of Operator Work Arounds
- SRG-95-038, Nuclear Safety Review Group Assessment of PED-QP-19 and Generic Letter 91-18 Implementation, April 1995
- SRG-95-086, Diesel Generator Hot Weather Operation Issue, Revision 0

OTHER DOCUMENTS

- Engineering Analysis EA-FC-91-034, AOP-32 480 Volt Island Bus Transfer, Revision 1
- Procedure Change 38794, Diesel Generator Loading Curve (TDB-III.26.A), January 20, 1993
- Procedure Change 44302, Diesel Generator Loading Curve (TDB-III.26.A), July 20, 1995
- Procedure Change 44386, DG-1 and DG-2 Output Power Rating (TDB-III.26.A), August 22, 1995
- Operations Memorandum 95-12, DG Temperature Limits, August 8, 1995
- Vendor Manual TD P319.0020, Three Phase Thyristor Controlled Constant Potential Battery Chargers
- Vendor Manual TM C173.0010, C & D (Charter Power Systems) Batteries