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EXECUTIVE SUMMARY

From January 27 through March 14, 1997, the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (NRR), Special Inspection Branch, conducted a design inspection at Units 1 and 2 of the Joseph M. Farley Nuclear Plant. Specifically, this inspection included visits to the licensee's engineering offices and the plant site on February 3-7, February 17-28, and March 10-14, 1997. The inspection team consisted of a team leader from the NRR and four engineers from the Stone & Webster Engineering Corporation.

As the focus of this design inspection, the team selected the Unit 1 auxiliary feedwater (AFW) system, the Unit 2 component cooling water (CCW) system, and their support systems, because of the importance of these systems in mitigating accidents at Farley. The purpose of the inspection was to evaluate the capability of the selected systems to perform safety functions required by their design bases, as well as the adherence of the systems to their respective design and licensing bases, and the consistency of the as-built configuration and system operations with the final safety analysis report (FSAR). To achieve this purpose, the team followed the engineering design and configuration control section of Inspection Procedure 93801. The team also selected and reviewed relevant portions of the FSAR, design-basis documents, Technical Specifications (TS), drawings, calculations, modification packages, procedures, and other plant-related documentation.

Overall, the inspection team determined that the selected systems are capable of performing their intended safety functions and have adequate design margins. The interaction and communication between the plant, home office engineering, and engineering contractors were well managed. Moreover, the engineering staff exhibited adequate knowledge of the systems evaluated and provided excellent support to the inspection team. The team also found that the Farley design and licensing bases have been adequately implemented in all but a few instances, and the quality of calculations and design changes was generally good. Although we consider the safety system self-assessments for AFW and CCW systems performed by the licensee to be a positive initiative, the assessments did not identify and correct many of the issues raised by the team. In addition, the system design documents reviewed by the team adequately supported the design, with the exception of the items discussed in the following paragraphs.

The licensee's evaluations of plant modifications, conducted in accordance with 10 CFR 50.59, were generally adequate. However, one 10 CFR 50.59 safety evaluation performed as a result of a licensee self-initiated safety system assessment associated with an FSAR change deleting the requirement for tornado missile protection for several condensate storage tank (CST) piping connections failed to identify the existence of a potential unreviewed safety question. As a result of the team's questions, the licensee reported this issue, in accordance with 10 CFR 50.72, on February 27, 1997, and implemented interim corrective actions to maintain the operability of the system until the issue can be resolved. In a similar instance, the 10 CFR 50.59 evaluations for a surveillance test procedure and FSAR revision did not identify that a

technical specifications change was required as a result of changes to an Auxiliary Building battery profile and associated battery voltage requirements.

The team also observed inadequate tornado missile protection of the turbine-driven AFW (TDAFW) pump vent stack and the CST level transmitters, and their associated electrical conduits and cables. The as-built plant configuration did not conform to the design and licensing bases requirements concerning tornado missile protection. In addition, the exhaust silencers for the diesel generators (including the station blackout diesels) were only protected against horizontal tornado missiles, although the equipment is also susceptible to vertical and other non-horizontal missiles. The NRR staff will review this issue associated with the diesel generators to determine whether the tornado missile protection in the Farley Unit 1 and 2 design and licensing bases included missile spectra other than horizontal missiles.

The TDAFW pump discharge check valves were not tested in the reverse direction, and the surveillance test procedure acceptance criterion for forward flow testing of certain AFW check valves was incorrect. In addition, the corrective action for a notice of violation for a similar issue did not thoroughly address the check valve reverse flow testing deficiency. Another testing issue included the lack of a service test for the Class 1E TDAFW battery.

The team identified certain design control weaknesses for calculations. Superseded calculations were not always identified on the calculation index, and design-basis calculations were not always updated when design conditions changed. The licensee implemented controls for new calculations that would prevent such occurrences in the future. However, no action was taken to correct deficiencies with existing calculations.

The team's other concerns included : the design-basis differential pressures for several motor-operated containment isolation valves were incorrect; the fluid temperatures used in the CCW piping stress analyses did not use the maximum operating temperature; the setpoint calculation for the CST low-level alarms did not consider drift and deadband errors; process and documentation discrepancies in the modification to coat portions of the CCW heat exchangers with epoxy and stabilize/plug tubes, as well as the lack of specific documentation demonstrating the adequacy of a fire protection seal; and the installed configurations of the TDAFW battery rack and the CST level transmitter were not in accordance with design requirements.

The team identified discrepancies between the FSAR and other documents, such as functional system descriptions (FSDs), procedures, calculations, and drawings. In some cases, the licensee had not updated the FSAR or performed safety evaluations to assess the possibility of unreviewed safety questions.

The licensee's corrective measures effectively resolved the immediate concerns identified by the team, and the team did not have any unresolved operability concerns.

Report Details

III. Engineering

E1 Conduct of Engineering

E1.1 Inspection Objectives and Methodology

The primary objective of the design inspection at the Farley Nuclear Plant was to evaluate the capability of selected systems to perform their safety functions and adherence to their design and licensing bases, and the consistency of the as-built configuration and system operations with the final safety analysis report (FSAR). The team selected the Unit 1 auxiliary feedwater (AFW) system, the Unit 2 component cooling water (CCW) system, and their support systems, because of the importance of these systems in mitigating accidents at Farley. The inspection focused on the engineered safeguards functions of these systems and their interfaces with other systems. For guidance in performing the inspection, the team followed the applicable engineering design and configuration control portions of Inspection Procedure (IP) 93801, "Safety System Functional Inspection."

Appendix A identifies the open items resulting from this inspection, while Appendices B, C, and D identify the exit meeting attendees, the list of acronyms used, and the list of documents reviewed, respectively.

E1.2 Auxiliary Feedwater System

E1.2.1 System Description and Safety Functions

The AFW system consisted of two motor-driven pumps, one turbine-driven pump, and the associated piping, valves, instruments, and controls. The system was designed to use water from the condensate storage tank (CST) or the service water (SW) system backup supply to provide high-pressure feedwater to the steam generators through the main feedwater (MFW) system. Each of the motor-driven AFW (MDAFW) pumps was sized to supply the steam generators with 100% of the required feedwater flow for a normal safe cooldown of the reactor coolant system (RCS). The turbine-driven AFW (TDAFW) pump was sized to supply the steam generators with 200 percent of the required feedwater flow for a normal safe cooldown of the RCS. Under accident conditions, both MDAFW pumps (assuming a failure of TDAFW) were required to provide the necessary feedwater flow.

The function of the AFW system was to supply high-pressure feedwater to the steam generators during plant startup, cooldown, and emergency conditions when the normal feedwater supply is not available. The system was designed as an engineered safety feature to provide redundant means of removing decay and sensible heat from the reactor coolant system via the steam generators during emergency conditions. In addition, the system was designed to meet the single failure criteria so that no single failure will prevent the supply of sufficient feedwater to at least two of the three steam generators. The AFW

system design was contingent on providing sufficient flow to prevent loss of pressurizer vapor space during a feedwater line break with a loss of offsite power.

E1.2.2 Mechanical Design Review

E1.2.2.1 Condensate Storage Tank (CST)

a. Inspection Scope

The mechanical design review of the CST included a detailed evaluation of the related equipment specification, drawings, manufacturer's information, and applicable calculations, in order to assess consistency with the design and licensing bases.

b. Observations and Findings

1. Structure of the CST

The CST was designed to serve as the normal source of water for the AFW system. The total CST volume was approximately 500,000 gallons. The lower 150,000 gallons was reserved for the decay heat removal and cooldown requirements of the AFW system, and this portion of the CST was designed to withstand the forces resulting from the impact of a design-basis tornado-generated missile without loss of pressure boundary integrity.

The team reviewed the CST manufacturer's drawings U161693D, Revision A0, and U161703B, Revision 2, and verified that the volumes of both the entire CST and the lower portion of the CST were correct. The team also verified the tank nozzle locations and the tornado missile criteria shown on drawing U161693D, which indicated that the lower portion of the CST was fabricated from 1-inch thick steel plate. Specifically, the team reviewed the CST equipment specification, SS-1111-4, Revision 2, and manufacturer-provided CST design calculation, Charge No. 72-4859-Memphis Engineering. As a result of this review, the team questioned the basis of the equation used in the manufacturer's CST design calculation. Specifically, the manufacturer used a ballistics equation intended to determine the required thickness of a steel plate to prevent perforation. The manufacturer did not provide a basis for using this equation to demonstrate the capability of the tank to withstand the impact of a design-basis tornado-generated missile without a loss of pressure boundary integrity. Moreover, the equation was not based on a specific grade of steel.

In response to the team's concerns, the licensee performed an independent verification calculation (SM-97-S19108-001) to assess the adequacy of the CST shell thickness for the most limiting design-basis tornado-generated missile. This analysis verified that the maximum membrane stress from the missile would be less than the allowable stress for the tank material. As a result, the licensee's independent calculation confirmed that the CST was capable of withstanding the impact of a design-basis tornado-generated missile without a loss of pressure boundary integrity.

The team also reviewed calculation 1.10, "Condensate Storage vs. Decay Heat," Revision 1, and calculation BM-95-0961-001, "Verification of CST Sizing Basis," Revision 1. On the basis of this review, the team verified that the CST volume was adequate to perform the required functions during normal and accident conditions.

2. Unprotected CST Connections

On October 21, 1994, during a self-initiated safety system assessment (SSSA) of the AFW system, the assessment team discovered that the following connections on the lower portion of the CSTs in both Unit 1 and Unit 2 were not missile protected:

- drain connections on the Unit 1 and Unit 2 CSTs
- sensing lines for the level transmitters on the Unit 1 and Unit 2 CSTs
- 6-inch diameter nozzle and isolation valve on the Unit 2 CST
- 6-inch diameter vacuum degasifier suction line on the Unit 1 CST

To resolve this issue, the licensee issued Incident Report 1-94-299, Licensee Event Report (LER 94-005), and an FSAR change and associated 10 CFR 50.59 Safety Evaluation. According to LER-94-005-00, dated November 18, 1994, probabilistic risk assessment (PRA) techniques indicated that the increase in the probability of an accident resulting from the lack of missile protection on the subject connections was negligible. The LER further stated that the licensee had revised the FSAR to include the PRA results and to delete the requirement for missile protection of the subject CST connections. The PRA concluded that the impact frequency with which a tornado missile could strike exposed CST piping was on the order of 1.0×10^{-8} per year. An estimate of the resulting core damage frequency (CDF) was on the order of 1.0×10^{-9} per year. The licensee also documented the PRA in calculation REES-F-94-014, "Farley - Estimate of Core Damage Frequency from a Tornado Missile Striking Exposed CST Piping," Revision 0. The team noted that NRC Inspection Report 95-20, dated January 23, 1996, documented the related LER review.

Based on the 10 CFR 50.59 safety evaluation associated with the FSAR change, the licensee concluded that the lack of missile protection on the CST tank connections did not place the plant outside its design bases. Further, the licensee concluded that the FSAR change could be implemented without prior approval from the NRC because it did not involve an unreviewed safety question (USQ) or a change to the Technical Specifications (TS). The team noted that NRC Inspection Report 96-07, dated September 27, 1996, documented a review of this safety evaluation.

The team also recognized that the licensee's conclusion regarding the 10 CFR 50.59 safety evaluation was founded on Regulatory Guide (RG) 1.70, Section 2.2.3.1, which states that design-basis events external to the nuclear plant are defined as accidents that have a probability of occurrence of about 10^{-7} per year or greater with potential consequences serious enough to affect the safety of the plant. The licensee's safety evaluation compared the calculated

impact frequency with which a tornado missile could strike exposed CST piping (approximately 1.0×10^{-8} per year) to the RG 1.70 value of 1.0×10^{-7} per year and concluded that this postulated tornado event was not required to be analyzed as an "accident" in the FSAR. However, the team concluded that this comparison was not appropriate because the RG 1.70 value of 1.0×10^{-7} per year applied to the probability of occurrence of an external event, such as an explosion or a toxic gas release, that could affect the plant as a whole, not the probability of a tornado missile striking a specific target in the plant.

FSAR Section 3.5.4 states that Category I equipment and piping outside containment are either housed in Category I structures or buried underground. FSAR Table 3.2-1 (Footnote 24) supplements that general statement by addressing the tornado missile design of the CST. In addition, FSAR Section 9.2.6.3 addresses the potential of tornado missiles striking the CST. At the time that this condition was discovered on October 21, 1994, FSAR Sections 9.2.6.3 and 9.2.6.6 stated that the lower section of the CST was designed to withstand tornado missiles.

Criterion II of Appendix A to 10 CFR Part 50, Appendix A, requires that structures, systems, and components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as tornados without loss of capability to perform their safety functions. The Farley licensing bases included this criterion.

Consequently, the licensee's 10 CFR 50.59 safety evaluation dated November 17, 1994, included a question (question number 6) asking, "May the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR?" The licensee answered this question "No," in accord with the presumption that a CST failure caused by a missile was less likely than other malfunctions considered by the FSAR. However, as discussed above, it was not appropriate to compare the frequency of a tornado missile strike on specific equipment with the 1.0×10^{-7} per year criterion for the occurrence of external design-basis events. In addition, Question 6 in the safety evaluation addressed the possibility of an equipment malfunction; it did not address the probability compared to other malfunctions considered by the FSAR. The team therefore stated that the use of a PRA to determine the frequency of a tornado missile strike on specific equipment was not an appropriate justification to determine that a USQ did not exist when the as-built plant configuration did not conform to the FSAR.

The 10 CFR 50.59 safety evaluation referred to the NRC Standard Review Plan (SRP), Section 3.5.1.4, "Missiles Generated by Natural Phenomena," which states that the methodology used to identify appropriate design-basis missiles generated by natural phenomena shall be consistent with the acceptance criteria defined for the evaluation of potential accidents from external sources in SRP Section 2.2.3. On the basis of this cross-reference, the licensee's safety evaluation cited SRP Section 2.2.3, "Evaluation of Potential Accidents," and RG 1.70, Section 2.2.3.1, to establish the "1.0×10^{-7} per year" criterion. However, both of these references addressed potential accidents involving hazardous materials or activities related to nearby industrial, military, and transportation facilities in the vicinity of the

plant. The cross-reference from SRP Section 3.5.1.4 to SRP Section 2.2.3 addressed the identification of appropriate design-basis missiles generated by natural phenomena, not the frequency of a tornado missile strike on specific equipment. Consequently, the team observed that the use of RG 1.70, Section 2.2.3.1, as a basis for the "1.0×10^{-7} per year" criterion was not appropriate because RG 1.70 addresses potential accidents involving hazardous materials or activities in the vicinity of the plant, not tornado-generated missiles.

The licensee stated that both deterministic and probabilistic analyses were performed in support of the safety evaluation. However, the licensee did not document the deterministic analysis in the safety evaluation. The licensee also stated that both the LER and safety evaluation had previously been reviewed and found to be acceptable by the NRC. In addition, the licensee stated that it was not their normal practice to use PRA techniques as the primary justification for safety evaluations.

After reviewing other 10 CFR 50.59 safety evaluations and discussing the matter with the licensee, the team determined that the inappropriate use of PRA techniques was not a widespread problem. Nonetheless, the team identified a potential USQ because the licensee effectively changed the plant from that described in the FSAR. The installed condition of the safety-related CST tank did not conform to the original design bases documented in the FSAR, and the facility change was made without prior approval from the NRC. In addition, the team determined that the licensee's corrective action for self-initiated safety system assessment (SSSA) observation was not adequate to resolve this deficiency (unprotected CST piping) and did not meet the requirements specified in Criterion XVI of Appendix B to 10 CFR Part 50.

As a result of this condition, a tornado missile could damage some CST connections, thereby resulting in a loss of inventory from the safety-related tank and affecting the operability of the entire AFW system. Tank damage resulting from a tornado missile would be considered a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR. As defined in 10 CFR 50.59, an activity that creates the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR is considered a USQ. Further, 10 CFR 50.59(c) requires that the licensee submit an application for amendment of its license pursuant to 10 CFR 50.90 when changes are made which involve a USQ. In addition to Criterion XVI of Appendix B to 10 CFR Part 50, the requirements of 10 CFR 50.59 were apparently not met. This issue was identified as Unresolved Item 50-348; 50-364/97-201-01.

In response to this item, the licensee initiated Occurrence Report Number 1-97-047 and issued a non-emergency notification LER in accordance with 10 CFR 50.72(6)(1)(ii)(b) during the inspection. The Occurrence Report documented interim actions required to support the operability of the tank. These actions included verification that the CST connection valves were closed and tagged and revision of the abnormal operating procedure to carefully monitor the CST level during adverse weather conditions and to take action if the CST develops leakage. The licensee committed to implement the action required to bring the plant condition into conformance with the licensing bases.

3. CST Level Transmitter Enclosure

During the walkdown of the Unit 1 CST, the team observed that the safety-related CST level transmitters and enclosures (fabricated from 1/4-inch steel plate), as well as the associated cables and conduits, were outside, and routed above ground around the tank perimeter. This equipment appeared to be susceptible to tornado-generated missiles.

FSAR Sections 3.2.1.5, and 9.2.6.1 and Table 3.2-1 state that the AFW system instrument and control (I&C) system equipment and CST equipment are Category I, respectively. FSAR Section 3.5.4 states that Category I equipment and piping outside containment are either housed in Category I structures or buried underground. FSAR Table 3.2-1 (Footnote 24) supplements this general statement by addressing the tornado missile design of the CST. In addition, FSAR Section 9.2.6.3 addresses the potential of tornado missiles striking the CST. Criterion II of Appendix A to 10 CFR Part 50 requires that SSCs important to safety be designed to withstand the effec , of natural phenomena such as tornados without loss of capability to perform their safety functions. The Farley licensing bases included this criterion.

The licensee responded that neither the level transmitter enclosures nor the ridged conduit were designed to provide tornado missile protection. Further, the licensee stated that the operators would be informed of the loss of these instruments by level indicators LI-4132A or LI-4132B failing high or low. Consequently, the licensee concluded that the functional impact or consequences of a tornado-generated missile destroying the level transmitter tubing connections would be similar to a missile destroying the conduits that contain the transmitter control circuits. These transmitter circuits were intended for indication and did not perform an automatic control function. The licensee also stated that the AFW pump suction pressure transmitter (PT-3211A) provided a diverse means for determining CST level. AFW flow transmitter (FT-3402) and steam generator level transmitter (LT-477) also provided indications to ensure that CST inventory was available. On that basis, the licensee concluded that the level transmitters were not essential to safely shut down the plant during events involving missiles.

Nonetheless, the installed condition of the safety-related CST tank level transmitters, and the associated cables and conduits, did not conform to the Farley design and licensing bases. As a result, these instruments could be damaged or destroyed by a tornado missile. In addition to Criterion II of Appendix A to 10 CFR Part 50, the requirements of 10 CFR Part 50, as specified in Appendix B, Criteria III and V which state that applicable regulatory requirements and design basis are correctly translated into drawings, procedures and instructions and that activities affecting quality be prescribed by and performed in accordance with instructions, procedures or drawings were apparently not met. This issue was identified as Unresolved Item 50-348; 50-364/97-201-02.

To address this item, the licensee issued ABN 97-0-1043. Other issues related to tornado missile protection of safety-related equipment are addressed in Section E1.2.6 of this report.

4. Other CST Issues

The team identified a conflict involving the CST-related AFW Operator License Training Objectives, OPS-52102H, dated June 21, 1992. Specifically, Question/Answer 5, stated that the CST low-low level alarms were set to inform the operator when the AFW had 20 minutes of normal supply left, with three AFW pumps operating at full capacity. However, the team observed that this stated basis was not consistent with the discussion of the CST low-low level alarm setpoint in the AFW functional system description (FSD), A-181010, Revision 1, Section 3.21.6.4, and calculation SM-87-1-4380-001, Revision 0. Specifically, both the FSD and the calculation indicated that the setpoint was dependent on the TDAFW pump operating at 700 gpm for 20 minutes.

The licensee confirmed this dependency on the basis of calculation SM-87-1-4380-001, "Setpoint Calculation for Level Alarm Replacement," Revision 0. The licensee stated that the level alarm setpoint, as documented in this calculation was appropriate to inform the operator when the AFW had 20 minutes of normal supply left. Further, the licensee cited plant procedures indicating that the AFW system supply would be aligned to the SW system when the setpoint was reached. However, the team observed that OPS-52102H was not correct and did not provide the information required for the operators to understand the basis of the setpoint. The licensee issued REA 97-1410 to revise OPS-52102H.

c. Conclusions

Overall, the CST was capable of performing its required functions during both normal and accident conditions. With the exception that some safety-related equipment was susceptible to tornado generated missile damage, the mechanical aspects of the CST were consistent with the design and licensing bases. However, the team noted one instance in which the licensee's safety evaluation for an FSAR change regarding unprotected CST piping failed to identify a potential USQ and corrective action for the SSSA observation regarding this issue was not adequate.

E1.2.2.2 AFW System Performance

a. Inspection Scope

The mechanical design review of AFW system performance included a detailed evaluation of the related system drawings, specifications, manufacturer's data, and applicable calculations, in order to assess consistency with the design and licensing bases.

b. Observations and Findings

The team reviewed calculation 29.01, "Auxiliary Feedwater Pumps Minimum Flow Evaluation," Revision 1, and verified that the minimum flow lines for both the MDAFW and TDAFW pumps had adequate capacity. The team also reviewed calculation 11.13, "As-Built NPSH for Auxiliary Feedwater Pumps," Revision 0,

and verified that, even with the CST empty, net positive suction head (NPSH) exceeded that required for the pump runout flow experienced during the initial period of a steam or feedwater pipe break event.

Next, the team reviewed calculation 40.02, "Verification of Auxiliary Feedwater Flow Basis," Revision 3, to verify the system flow capacity for various normal and accident conditions supported by the AFW system. This calculation showed that all of the AFW system minimum flow requirements were met by the minimum available AFW pump performance, considering up to 5% degraded performance. The calculation also showed that the system did not exceed the maximum allowable AFW flow limit for main steam line break (MSLB) and accidental depressurization events, with 5 % enhanced pump performance. The team found these results acceptable.

The team also observed that calculation 25.3, "Overpressurization of Auxiliary Feedwater Piping During Overspeed Testing," Revision 0, addressed the maximum AFW system pressure resulting from the TDAFW pump operating at 125% of its rated speed. By contrast, the licensee indicated that the current overspeed trip setpoint for the TDAFW pump was 115 % of the rated speed, and verified that the system pressure corresponding to 115 % was acceptable for the pump, piping, valves, and flanges in the system. Section E1.5 of this report, "Control of Calculations," addresses the issue that the licensee had not revised calculation 25.3 to reflect the current setpoint.

c. Conclusions

The team found that the mechanical aspects of the AFW system were consistent with the design-basis requirements and the system was capable of performing the required functions during both normal and accident conditions. In addition, the team found that the current mechanical calculations related to system performance were of high quality, with inputs, assumptions, methods, and results clearly stated and well documented.

E1.2.2.3 AFW System Valve Operation

a. Inspection Scope

The mechanical design review of AFW system valve operation included a detailed evaluation of drawings, specifications, manufacturer's data, operating procedures, and applicable calculations associated with various AFW system motor-operated, air-operated, manual, and check valves. Section E1.2.2.4 of this report addresses the licensee's testing of the AFW system valves..

b. Observations and Findings

The team reviewed calculation 23.5, "FNP Auxiliary Feedwater MOVs (IE Bulletin 85-03)," Revision 1, to verify that the licensee used the appropriate maximum differential pressure value for each motor-operated valve (MOV) in the AFW system. In addition, the team verified that the differential pressure values for each valve represented the most limiting system operating conditions, and that the calculated values were consistent with the Unit 1 MOV Design-Basis Document, U418109, Revision A.

The team then reviewed the AFW System Process & Instrumentation Diagram (P&ID), D-175007, Revision 22, and the Main Steam (MS) and Auxiliary Steam System P&ID, D-175033, Revision 18, to verify the normal position of each valve in the AFW system and the steam supply to the AFW pump turbine drive. The team also found that Surveillance Test Procedure FNP-1-STP-22.5, "Auxiliary Feedwater System Flow Path Verification," Revision 17, was consistent with the P&IDs. These normal valve positions were appropriate for the function of the valves.

Procedure FNP-1-AOP-6.0, "Loss of Instrument Air," Revision 20, included a list of air-operated valves. In reviewing this procedure, the team noted that Table 1 (page 16 of 29) included valves 1-AFW-FCV-3212A, 1-AFW-FCV-3212B, and 1-AFW-FCV-3218, which were removed by Production Change Notice (PCN) 1-5582 and are no longer installed in the plant. In response to the team's question, the licensee stated that a temporary change notice (TCN) was issued to remove these valves from the procedure, and FNP-1-AOP-6.0, Revision 21, was issued on February 19, 1997, to incorporate the required change.

During the walkdown of Unit 1, the team observed that electrical components associated with the safety-related AFW discharge valves (HV-3227A, B, C; and HV-3228A, B, C) were located in the main steam valve room approximately 4 feet above the floor. This elevation was above the stated flood elevation for the room, but was not consistent with the FSAR. The licensee contended that the location of the AFW valves was acceptable, and issued ABN 97-0-1043 to correct the FSAR. Section E1.2.7 of this report discusses this FSAR discrepancy in greater detail.

c. Conclusions

The team found that AFW system valves were capable of performing the required functions during both normal and accident conditions. With the exception of deficiencies in an abnormal operating procedure and in the FSAR, the mechanical aspects of the AFW system valve operation were consistent with the design-basis requirements.

E1.2.2.4 AFW System Testing

a. Inspection Scope

The team evaluated the licensee's surveillance test procedures (STPs) to verify that the licensee had appropriately tested the AFW pumps and all of the required system valves.

b. Observations and Findings

The team reviewed the results of the following STPs completed during 1995 and 1996:

- FNP-1-STP-22.1, "1A Auxiliary Feedwater Pump Quarterly Inservice Test," Revision 23

- FNP-1-STP-22.8, "Auxiliary Feedwater Inservice Valve Exercise Test," Revision 13

- FNP-1-STP-22.16, "Turbine-Driven Auxiliary Feedwater Pump Quarterly Inservice Test (TAVE>/=547 F)," Revision 25

- FNP-1-STP-22.21, "Turbine-Driven Auxiliary Feedwater Pump Automatic Valve Test," Revision 5

These test results were all satisfactory, with data within the acceptance criteria for the procedure.

The team also reviewed one example of a surveillance test failure and the associated corrective action. On February 29, 1996, the "B" motor-driven AFW pump discharge check valve failed to pass a reverse flow closure operability test (STP-22.24). This valve had also failed the reverse flow closure operability test on December 1, 1992. The licensee's subsequent evaluation revealed that the root cause for both failures was an accumulation of sediment on the check valve seating surfaces. As a result, the licensee issued Work Order (WO) 96001338 to inspect and clean the valve during the Unit 1 RF15 refueling outage, which began in March 1997. The licensee stated that if the sediment accumulation was deemed to be significant by mechanical maintenance engineering, a preventative maintenance task would be established to inspect and clean the valve during every other refueling outage. The team determined that the licensee's root cause evaluation and corrective action were appropriate and sufficient. No other similar conditions were identified during the inspection.

In a similar fashion, the team's review revealed that the licensee had not tested TDAFW pump discharge check valves V003 and V002D, F, and H in the reverse direction. (Valves V002A and 002B, located in the discharge lines from the MDAFW pumps, were tested in the reverse direction.) Check valve V003 was located in the discharge line from the TDAFW pump, while valves V002D, F, and H were located in the supply lines from the TDAFW pump discharge header to each feedwater line. The AFW FSD, A-181010, Revision 1, Section 5.27.1, stated that the basic function of valve V003 was to prevent reverse flow into the TDAFW pump when the pump was idle.

Either check valve V003 or check valves V002D, F, and H were required to perform a safety function in the closed position. Failure of these valves to close could potentially result in failure of the MDAFW pumps to perform their safety function. In the event that one or both of the MDAFW pumps were operating and the TDAFW pump was not operating, failure of the TDAFW pump discharge check valves to close could result in some flow from the MDAFW pumps failing to reach the steam generators. That flow from the MDAFW pumps could return to the CST though the TDAFW pump minimum flow line, or could be discharged through the TDAFW pump suction relief valve, QV068C. In addition, overpressurization of the TDAFW pump suction piping could occur if the reverse flow though the check valves exceeded the capacity of the TDAFW pump suction relief valve, QV068C. The team noted that 10 CFR 50.55a(f) and TS Section 4.0.5 require inservice testing (IST) of valves in accordance with Section XI and applicable Addenda of the Boiler and Pressure Vessel (B&PV) Code

promulgated by the American Society of Mechanical Engineers (ASME). In particular, the purpose of this testing was to verify the operational readiness of valves that must function to ensure plant safety.

Reverse flow testing of check valves was addressed in NRC Information Notice (IN) 88-70, dated August 29, 1988. In a memorandum dated September 12, 1989, responding to this IN, the licensee listed the AFW system valves that were being subjected to reverse flow testing at the time. The memorandum did not address the lack of reverse flow testing of the TDAFW pump discharge check valves. In addition, during an inspection conducted on October 2-6, and 16-20, 1989, the NRC identified a violation involving failure to test the MDAFW pump discharge check valves, V002A and V002B, in the reverse direction. Subsequent to the 1989 inspection, the licensee expanded the IST program to include reverse flow testing of the MDAFW pump discharge check valves, V002A and V002B, but did not address the TDAFW pump discharge check valves. The licensee subsequently issued REA 97-1410 to correct this discrepancy.

The team noted that TS Section 4.05 requires inservice testing of ASME Code Classes 1, 2, and 3 pumps and valves in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. The licensee is committed to inservice testing in accordance with the 1983 Edition of the Code and Summer 1983 Addenda. Section XI, Subsection IWV-3522 requires valves whose function is to prevent reverse flow to be tested in a manner that proves that the disk travels to the seat promptly on cessation or reversal of flow. The lack of adequate inservice testing was identified as Unresolved Item 50-348; 50-364/97-201-03.

In addressing this item, the licensee stated that either check valve V003 or check valves V002D, F, and H were required to perform a safety function in the closed position. Only one isolation barrier was required in the TDAFW pump injection line to prevent diversion of flow from an operating MDAFW pump. The licensee also stated that the results of performance testing during each unit's last refueling outage indicated that check valves V002D, F, and H were sufficiently closed to allow the MDAFW pumps to perform their safety function. In addition, to verify operability, the licensee disassembled and inspected the V003 check valves for both Unit 1 (Work Order No. 97001743) and Unit 2 (Work Order No. 97001744) on March 7, 1997. These activities provided an acceptable confidence level that the valves were operable. Further, the licensee stated that check valves V002D, F, and H would be identified in the IST program and plan documents as having a safety function to close, and Surveillance Test Procedure FNP-1/2-STP-22.29 would be issued to test check valves V002D, F, and H in the reverse direction. In addition, the licensee issued FNP Occurrence Report 1-97-048 on March 3, 1997, to document this condition and to ensure that all corrective measures would be implemented correctly and in a timely manner.

The team also reviewed Surveillance Test Procedure FNP-1-STP-22.13, "Turbine-Driven Auxiliary Feedwater Pump Check Valves Flow Verification," Revision 14, dated May 7, 1996. The purpose of this procedure was to verify the capability of check valves in the TDAFW pump system to pass the full design flow. (Section 2.2 of the STP defined an acceptance criterion of 625 gpm flow for check valves V006 and V003.) The surveillance test involved aligning the

TDAFW pump to the steam generators and verifying that the pump flow was at least 625 gpm, measured by either FE-3218 or FI-3403A. These flow instruments were located in the pump suction piping and measure the total pump flow.

Check valve V006 was located in the pump suction line and would be required to pass the full pump flow during the test. However, valve V003 was located in the pump discharge line downstream from the open pump minimum flow recirculation line. Therefore, valve V003 would pass less than the full pump flow.

The difference between the measured pump flow and the valve flow was the recirculation line flow. According to Calculation 29.01, "Auxiliary Feedwater Pumps Minimum Flow Evaluation," Revision 0, the recirculation line flow was approximately 95 gpm with the pump running at 3960 rpm. Therefore, the flow through check valve V003 would be on the order of 530 gpm when the pump was operated at 625 gpm. The current acceptance criterion of 625 gpm, presented in Section 2.2 of FNP-1(2)-STP-22.13, was not consistent with the actual test flow. Criterion V of Appendix B to 10 CFR Part 50 requires that activities affecting quality be prescribed by and performed in accordance with instructions, procedures or drawings which include appropriate acceptance criteria for determining the activity is satisfactorily accomplished. The team identified the inadequate acceptance criterion for check valve as Unresolved Item 50-348; 50-364/97-201-04.

Amendment 112 to TS 3/4.7.1.2 states that the TDAFW pump is capable of delivering a total flow of 450 gpm. The check valve test flow of approximately 530 gpm was significantly greater than the TS requirement of 450 gpm. Therefore, the team did not have any operability concerns. In responding to this item, the licensee stated that U1-TCN-14A and U2-TCN-12A would be issued to correct Section 2.2 of FNP-1/2-STP-22.13 for the forward flow test.

c. Conclusions

The team concluded that the licensee's AFW system testing was generally sufficient to verify that the pumps and valves were capable of performing their required functions. However, the team noted weaknesses in the check valve testing. In one instance, the licensee's corrective actions in response to a previously identified violation did not thoroughly address the check valve testing deficiency.

E1.2.2.5 <u>AFW System Modifications</u>

a. Inspection Scope

In this portion of the mechanical design review, the team selected 7 AFW system modifications for detailed evaluation. In particular, the team assessed each selected modification in detail to verify that the problem identification and justification for the modification were clearly stated; to determine if the subject was generic and potentially applicable to other systems; to determine the impact on the design and licensing bases; to verify that the safety evaluation, if required, was correct and consistent with the

applicable procedures; to verify that the post-modification testing was complete and appropriate; and to verify that the required plant documentation was updated to reflect the modification.

b. Observations and Findings

The team had no concerns regarding the modifications reviewed. The majority of the modifications reviewed were minor in nature, addressing such issues as replacement of control valve actuators, addition of as-built information to P&IDs, acceptance of check valve replacement parts of a different material, acceptance of replacement mechanical seals for the AFW pumps, and replacement of valve trim. The system modification packages were of high quality. The problem identification and justification for each modification were clearly stated, the safety evaluations were correct and consistent with the applicable procedures, post-modification testing was complete and appropriate, and the required plant documentation was updated to reflect the modifications. None of the modifications were generic in nature or applicable to other systems, and none of the modifications significantly impacted the design or licensing bases of the system.

The most significant modifications reviewed addressed reducing the overspeed trip setpoint for the TDAFW pump and replacing a section of carbon steel service water supply piping with stainless steel. The team reviewed modifications in detail and found them to be acceptable.

c. Conclusions

The team concluded that the licensee's AFW system modifications were consistent with the design-basis requirements and did not adversely affect the ability of the system to perform its required functions during both normal and accident conditions.

E1.2.3 Electrical Design Review

E1.2.3.1 Turbine-Driven AFW (TDAFW) Uninterruptable Power System (UPS)

a. Inspection Scope

In this portion of the electrical design review, the team evaluated the TDAFW batteries (Units 1 and 2) to assess their ability to supply adequate electrical capacity, as well as their conformance to the FSAR, installation documents, and maintenance and testing programs.

b. Observations and Findings

Calculation 07597-E-106, "Battery Capacity Calculation for TDAFP-UPS," Revision 1, provided the duty cycle analysis for those emergency loads required for AFW operation when battery charger support was not available. Calculation E-106 determined that adequate battery capacity from the TDAFW Class 1E battery was available to supply the AFW electrical loads.

Review of the maintenance and testing activities for the TDAFW batteries indicated that the licensee had performed weekly and quarterly inspections, yearly battery equalization, 18-month general battery cleaning, and 54-month UPS battery performance testing. However, the licensee did not perform battery service testing to verify the ability of the battery to meet the design duty cycle specified in calculation 07597-E-106, as identified in Sections 5.3 and 6.6 of the "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," of Standard 450-1980 promulgated by the Institute of Electrical and Electronics Engineers (IEEE). The team noted, however, that other Class 1E batteries in the plant were tested in accordance with IEEE 450 recommendations. FSAR Section 8.3.3.2 states that all TDAFW UPS components are designed to conform with Class 1E electrical system design criteria. FSAR Sections 8.1.4 and 8.3.2.1.5 identify the Class 1E battery testing requirements.

The team did not have any immediate safety concern, since the licensee's maintenance and testing (mentioned above) provided reasonable assurance that the battery could support the assigned loads. A test program with an appropriate written procedure was not identified and performed in accordance with written procedures incorporating the requirements specified in Criterion XI of Appendix B to 10 CFR Part 50, and TS Section 6.8.1.a. The team therefore identified this issue as Unresolved Item 50-348; 50-364/97-201-05. The licensee issued REA 97-1410 to evaluate this item, and stated that battery service testing will be performed in refueling outage RF14 for Unit 1 and RF12 for Unit 2.

The team compared the installed configuration of the TDAFW UPS system to FSAR Section 8.3.3, Design Change Packages (DCPs) 96-1-9008-0-001 and 96-2-9009-0-001, Production Change Notices (PCNs) 81-917 and 81-2125, and applicable design drawings. The installed configuration agreed with the design documentation of the previously mentioned DCPs, PCNs, and engineering drawings. The six output fuses for the UPS rectifier had been changed from 3 amperes to 5 amperes. Section E1.2.7 of this report discusses an FSAR discrepancy associated with this change. Adequate engineering was performed to ensure that the equipment was protected and coordinated.

Review of the TDAFW battery installation for Unit 2 indicated that various structural and electrical components were not installed in accordance with the manufacturer's drawings and instructions (7597-20-E10.9-29, Revision 1, U265645A, Revision 2, U263212, Revision D). The following deficiencies were noted:

- Five polystyrene spacers were installed between the battery cell and the end rail where none were required.

- Structural steel bracing in the rear of the rack did not agree with the drawing.

- Bolts were missing from the upper-and lower-tier tie rod brackets.

- Silicon bronze bolting hardware was utilized at the cable terminations in lieu of stainless steel hardware as shown on vendor drawing U265645A, Revision 2.

- The intercell battery connections were torqued to 75 in-lbs instead of the required 125 in-lbs specified in the battery manufacturer's instruction manual U263212, Revision D.

- The battery rack steel rails and tie rods exhibited corrosion.

The licensee reviewed the installed structural configuration of the TDAFW battery rack and determined by calculation that the corrosion, additional polystyrene cell spacers, and structural member configuration had not compromised the seismic design of the battery installation. On the basis of discussions with the battery manufacturer concerning the utilization of silicon bronze fastener material in lieu of stainless steel and the lower torque value of 75 in-lbs, the licensee concluded that the capability of the battery to perform as designed had not been compromised.

The licensee issued Deficiency Report #537766 for the battery rack corrosion; REA 97-1408 to reconcile the differences between the installation and the design drawings; and REA 97-1410 to revise TDAFW battery maintenance procedures and appropriate documentation to clarify acceptable fastener material. The team determined that the TDAFW battery structural and electrical component installations were not accomplished in accordance with drawings, procedures or instructions as required by Criterion V of Appendix B to 10 CFR Part 50. This issue was therefore identified as Unresolved Item 50-364/97-201-06.

c. Conclusions

Even though the team identified discrepancies with the structural and electrical aspects of the TDAFW battery installation, the team concluded that the TDAFW UPS system was capable of performing the design-basis functions of providing electrical power during the various plant operating modes for which it was designed.

E1.2.3.2 <u>AFW Electrical Loads</u>

a. Inspection Scope

In this portion of the electrical design review, the team evaluated the electrical loads required for the AFW system to perform its design functions under both normal and accident conditions. This evaluation addressed cable sizing, protective coordination, electrical bus loading, and dc battery loading calculations.

b. Observations and Findings

To ensure that the AFW loads had been identified, the team reviewed calculations SE-94-0470-001, "Unit 1 As-Built Load Study Update," Revision 1; SE-94-0470-004, "As-Built Load Study Summary Calculation," Revision 1; E-42,

"Steady State Diesel Loadings Calculation for LOSP, SI and SBO," Revision 8; E-35.1.A, "Setting of Protective Relays for FNP Unit 1 4.16 kV Auxiliary Power System," Revision 2; and E-95, "Battery Capacity Calculation for LOSP and LOSP+LOCA Situations and Limiting Battery Load Profile," Revision 7. All major electrical loads had been accounted for within these calculations for normal and accident conditions. The frequency and voltage ratings of the AFW electrical loads were compatible with the electrical design analyses. The team determined that the methodology and assumptions used were appropriate.

Calculation SE-94-0-0378-001, "Instantaneous Trip Settings for MCCBs in the MOV Setpoint Document," Revision 0, was reviewed. The team observed that the source of the overload relay time vs. current curve used for MOV-3406 was not documented. Additionally, the curve was used by extrapolation beyond its published data. The licensee contacted the overload relay manufacturer during the inspection and received information which supported the extrapolation used within the calculation. The licensee stated that a copy of the information from the manufacturer was filed for future reference. No additional concerns were identified with this calculation.

Cable sizing for the 4 kV Motor-Driven AFW (MDAFW) pump motors was reviewed. The review compared the actual cable data for the MDAFW pump motors against the "Electric Power Cable Sizing Guide," Revision 0. This review concluded that cable sizing for the MDAFW pump motors was adequate and was in conformance with the Farley cable sizing criteria.

c. Conclusions

The electrical design for components that perform the normal and emergency functions of the AFW system supported the design-basis functions of the system. The electrical distribution system provided independent, redundant, safety-related (Class 1E) power to the AFW electrical loads.

E1.2.3.3 Electrical AFW Modifications

a. Inspection Scope

In this portion of the electrical design review, the team evaluated electrical modifications to the AFW system and verified that the changes were in conformance with the AFW and electrical design bases.

b. Observations and Findings

The following modifications associated with the AFW system were reviewed by the team: ABN 96-0-0986, "Add Circuits to TDAFW Pump Control Panel"; PCN 89-1-6106, "AFW HFA Relay Drawing Change per 88-0-4980"; and PCN 89-1-6354, "Lighting Around TDAFW UPS Batteries." The document review for the referenced modifications indicated that appropriate procedures were followed in their preparation and that proper design review and verification was employed. 10 CFR 50.59 evaluations associated with the modifications were technically complete and prepared in conformance with administrative procedures. The team did not identify any concerns with these modifications.

c. Conclusions

The modifications reviewed by the team were designed in accordance with the system design bases.

E1.2.4 Instrumentation & Controls (I&C) Design Review

E1.2.4.1 Condensate Storage Tank

a. Inspection Scope

In this portion of the I&C design review, the team evaluated the licensee's determination of the CST low-level alarms.

b. Observations and Findings

The purpose of calculation SM-87-4380-001, Revision 0, "Condensate Storage Tank Low-Level Alarm Switches," was to prove that the condensate storage tank low-level alarms, Q1P11LA4133A and Q1P11LA4133B, would allow at least 20 minutes for operator action. It also verified that a "Moore Industries" switch could be used to replace an existing "Rosemount" switch to provide the alarms.

The calculation was reviewed for adequacy and setpoint uncertainties. The team observed that drift error was not addressed in the calculation. The I&C design criteria described this variable as being highly important in selecting setpoint tolerances. Rosemount defined the drift error for the Model 1152DP transmitter used in this loop as ± 0.2% upper range limit (URL). Although the use of this drift error did not add significantly to the loop error, it should have been addressed in the calculation. The team found that the available level was conservative and a drift of 0.2 % URL (less than one inch in 30 months) was very minimal.

The team also observed that the calculation provided a deadband specification for the replacement switch which was analyzed in this calculation for acceptability. The total instrument tolerance calculated by the square root of the sum of the squares method did not include the deadband of 1% of span. In "I&C Design Criteria for Joseph M. Farley Nuclear Plant-Units 1 and 2," page 15, Steps 12.6.1 and 12.6.2 Southern Company Services, Inc. noted that deadband should be assessed. The team found that it was not clear as to the circumstances to use it (i.e., when to use deadband in uncertainty calculations for a particular type of switch).

The licensee issued REA 97-1407 to review this uncertainty calculation and clarify the application of deadband as a factor in scaling calculations. The review of uncertainty calculation to address drift and deadband errors and clarify the application of deadband as a factor in scaling calculations was identified as Inspection Follow-up Item 50-348/97-201-07.

c. Conclusions

The absence of drift and deadband in the setpoint uncertainty calculation did not significantly affect the trip setpoint. The design basis was maintained. The set point had adequate margin to alert the operators for switchover to SW.

E1.2.4.2 Instrument Loop Uncertainty Calculations

a. Inspection Scope

In this portion of the I&C design review, the team evaluated the consistency and adequacy of the licensee's setpoint bases and uncertainty calculations for various instrument loops in the AFW system.

b. Observations and Findings

The licensee was asked to provide various design-basis documents defining analytical design limits and setpoint uncertainty for selected instrument loops in the AFW system. Only one calculation was provided, SM-87-1-4380-001, "Setpoint Calculation for Level Alarm Replacement," Revision 0. This was a calculation developed before the current setpoint program (GO-M-1, Revision 3) was initiated. The team found that it did not contain a complete instrument loop uncertainty calculation nor did it consider drift and deadband errors. The licensee could not produce additional instrument calculations to review.

The lack of calculations on some instrument loops made it difficult to judge whether the methodology that went into the development of the original setpoints conformed to the current setpoint program. The licensee developed a preliminary uncertainty calculation for the AFW system, Loop F3402, "Motor-Driven AFW Pump Suction Flow" (high-flow setpoint), to confirm the validity of the original margin between the setpoint and the analytical limit. This calculation showed that there was adequate margin in the alarm setpoint, considering loop uncertainties for a high-flow condition in the suction of the motor-driven AFW pump. The preliminary calculation for Loop F3402 was adequate and consistent with Farley's current setpoint program, GO-M-1, Revision 3. The methodology in the uncertainty calculation was also consistent with the I&C design criteria. The terminology in the calculation was adequately defined and included appropriate supporting references.

c. Conclusions

The calculation provided confidence that there was built-in conservatism and margin with the data found in surveillance test procedures. The setpoint program was adequate and consistent with industry standards and consistent with Westinghouse methodology.

E1.2.4.3 Instrument Setpoint Uncertainty Program

a. Inspection Scope

In this portion of the I&C design review, the team evaluated GO-M-1,"Designer Interface Document," Revision 3, which provided design guidance for the Farley instrument setpoint uncertainty program.

b. Observations and Findings

The licensee's setpoint program was generally consistent with recommendations from the Instrument Society of America (ISA), and the setpoint calculation methodologies followed the ISA standards ISA-S67.04 Part I and II. The licensee had incorporated the above methodologies in I&C design criteria document. The random uncertainties were combined using the square root-sum-of-squares, and non-random uncertainties were combined algebraically.

The team's review of design interface document GO-M-1 showed that when design changes impacted setpoint calculations that had been performed by Westinghouse in emergency operating/response procedures (EOPs/ERPs), the changes had to be evaluated by Westinghouse. Examples included uncertainties for considerations of sensor and instrument rack drift, accuracy and environmental effects, indicator accuracy, and calibration accuracy. The team found that the flow chart in Attachment 1, page 69, of GO-M-1, which depicted the instrument setpoint control, lacked the flow path for sending the change to Westinghouse for evaluation. The licensee stated that GO-M-1 would be reviewed to determine if changes were appropriate and issued REA 97-1410 to revise the I&C design criteria.

c. Conclusions

The team found that design document adequately described the method to update Westinghouse design changes, but the flow chart did not adequately reflect the text of the document. The setpoint uncertainty program was adequate.

E1.2.4.4 Modifications and Other Reviews

a. Inspection Scope

In this portion of the I&C design review, the team evaluated 12 AFW system design modifications and associated 10 CFR 50.59 safety evaluations to assess their adequacy and consistency with the relevant design bases. In addition, the team reviewed ATWS (anticipated transients without scram) mitigating system actuation circuitry (AMSAC) logic design, AFW actuation circuits design, and post accident monitoring instrumentation in accordance with Regulatory Guide (RG) 1.97, "Instrumentation for light-water-cooled nuclear power plants to assess plant and environs conditions during and following an accident," Revision 2 design.

b. Observations and Findings

The team found eleven modifications adequate. The quality of the modification documentation was generally good and consistent with the design bases. The team found the associated 10 CFR 50.59 evaluations acceptable. However, one discrepancy with PCN 84-1-2518, "Auxiliary Feedwater Check Valve Temperature Monitoring System," Revision 3, was found. This modification installed temperature elements on either side of various MDAFW and TDAFW pump discharge check valves.

The orientation of temperature elements TE-2293K and TE-2293L on P&ID D-175007, Revision 22, were reversed from the orientation in the PCN. The licensee issued ABN-97-0-1053 to revise the P&ID.

The team verified that the AMSAC logic design for the MDAFW pumps was consistent with design requirements. The team also verified that appropriate isolation devices were installed between Non-1E AMSAC circuits and 1E turbine-driven auxiliary feedwater trip circuits, and between non-1E and the 1E auxiliary feedwater start circuits. The steam generator level output signal to AMSAC (loops L485 and L496) was isolated properly with design requirements. AFW automatic initiation/isolation logic design was also consistent with the guidance provided in NUREG 0578, "TMI-1 Lesson Learned Tasks Force Status Report and Short Term Recommendations." The team determined that both the MDAFW and the TDAFW pump would automatically start on a low-low steam generator level as sensed by the AMSAC.

The team reviewed Farley's computerized Scaling Manual Unit 1, Section 10, "Steam Generator Level Control & Protection. The team found that this document which calculated the low-low level Reactor Trip was consistent with design requirements. However, the team did not review the Westinghouse uncertainty calculation to evaluate loop error as part of the square root-of-sum-of-squares (SRSS) methodology. The automatic actuation of the AFW system met the requirement of NUREG 0578.

In addition, the team reviewed the Regulatory Guide 1.97 Category 1 compliance checklists for the condensate storage tank level transmitters LT 515A/516B, Auxiliary Feedwater Flow transmitters, FT 3229A, B, and C. Regulatory Guide 1.97 stated that the indication for the AFW flow was considered a Category 2, type D variable with a required scale range of 0 to 110% of the design flow. The team verified that the scale for the AFW indicators met this guidance. The team found the checklists and instrument design consistent with RG 1.97 guidance.

c. Conclusions

The team concluded that the licensee's design modifications maintained the plant's design bases and no concerns were found with the safety evaluations. The licensee's design for AMSAC, post-accident monitoring instrumentation and AFW system initiation met the design-bases requirements.

E1.2.5 System Interface Design Review

E1.2.5.1 Service Water (SW) System

a. Inspection Scope

In this portion of the system interface design review, the team evaluated the SW system P&ID to verify that the interface between the SW system and the AFW system was consistent with the AFW system design bases. The team also reviewed Production Change Request (PCR) 92-1-8234, which replaced the carbon steel SW piping to the AFW system with stainless steel.

b. Observations and Findings

A backup source of water for the AFW pumps was provided from the safety-related portion of the SW system. The SW supply was isolated from the normal suction piping by two closed MOVs. These valves could be operated remote manually from the control room or by using the manual handwheel at the valve. The SW system also supplied cooling water for the motor-driven AFW pump room coolers.

The team had no concerns related to the SW system interface. A review of the SW system P&IDs, D-175003, Sheet 1, Revision 32, and D-175003, Sheet 2, Revision 27, indicated that 8-inch, safety-related SW system lines were provided from redundant SW system headers to the AFW system and to the AFW pump room coolers. This system configuration was consistent with the AFW design bases.

The team questioned the potential for galvanic corrosion at the dissimilar weld interface between the stainless steel piping added by PCR 92-1-8234, Revision 1 and the existing carbon steel valve body. In response, the licensee had evaluated this condition and determined that galvanic corrosion was not likely to be significant. The licensee also stated that the last examination of the equivalent weld on Unit 2, performed in April 1996, did not show any significant degradation. The team found this condition acceptable.

c. Conclusions

The interface between the SW system and the AFW system was consistent with the AFW system design bases.

E1.2.5.2 Instrument Air (IA) System

a. Inspection Scope

In this portion of the system interface review, the team evaluated the AFW system P&ID and the main steam (MS) and auxiliary steam system P&IDs to verify the normal and failure positions of each air-operated valve in the AFW system and the steam supply to the AFW pump turbine drive. The team also reviewed the service air system P&ID, D-175035, Sheet 2, Revision 6.

b. Observations and Findings

The team had no concerns related to the IA system interface. The IA system provided air for the air-operated valves in the AFW system and the steam supply to the AFW pump turbine drive. A review of the AFW system P&ID, D-175007, Revision 22, and the MS and auxiliary steam system P&ID, D-175033, Revision 18, indicated that each of the safety-related air-operated valves was designed to fail in a safe position upon loss of air or was provided with an air reservoir to allow continued operation of the valve. This design was consistent with the AFW design bases.

c. Conclusions

The interface between the IA system and the AFW system was consistent with the AFW system design bases.

E1.2.5.3 Main Steam (MS) System

a. Inspection Scope

In this portion of the system interface review, the team evaluated the AFW system P&ID and the MS and auxiliary steam system P&ID to verify that the interface between the MS system and the AFW system was consistent with the AFW system design bases.

b. Observations and Findings

The MS system provided a steam supply to the TDAFW pump driver. The interface between the systems was upstream from the safety-related MS isolation valves in the MS valve room. A review of the MS and auxiliary steam system P&ID, D-175033, Revision 18, indicated that two redundant steam supply lines were provided from redundant steam generators to the AFW pump driver. This system configuration was consistent with the AFW design bases.

The team questioned if NRC Information Notice 91-75, "Static Head Corrections Mistakenly Not Included In Pressure Transmitter Calibration Procedures," was taken into account in Farley's steam generator level control scaling documents. The response to this information notice discussed Vogtle Plant's discovery that a static head correction of approximately 25 psig had not been applied during the calibration of the pressurizer pressure transmitters. The same concern could apply to steam generator level measurement loops. The team reviewed Farley's Computerized Scaling Manual for Unit 1, Section 10, "Steam Generator Level Control and Protection." The document was reviewed for adequacy and consistency between the setpoint program and industry standards. The team found that the Westinghouse Computerized Scaling Manual was consistent with the methodology of the I&C design criteria document and industry standard ISA-S67.04, Part II. The scaling manual was adequate and setpoint studies were consistent with the requirements of the I&C design criteria.

c. Conclusions

The interface between the MS system and the AFW system was consistent with the AFW system design bases.

E1.2.6 System Walkdown

a. Inspection Scope

During the course of the design inspection, the team walked down all of the accessible portions of the AFW system, portions of interfacing systems, and the control room. Consistency of calibration intervals with licensing documents and calibration data were also checked during the walkdown. Verification of the as-built designs was conducted and compliance to RG 1.97 was observed.

b. Observations and Findings

1. TDAFW Pump Vent Stack

During the team's walkdown of Unit 1, it was observed that the safety-related TDAFW pump vent stack was installed outside, on the roof of the Auxiliary Building. This vent was not protected from tornado-generated missiles. FSAR Section 3.5.4 states that Category I equipment and piping outside containment are either housed in Category I structures or buried underground. FSAR Sections 6.5.1, 3.2.1.3, 3.2.1.5 and Table 3.2-1 state that AFW system equipment and piping are Category 1. FSAR Table 3.2-1 addresses the tornado missile protection of the AFW pumps. This stack could be damaged by a tornado missile, which could restrict the steam flow path from the turbine drive and adversely affect the AFW pump operability. Criterion II of Appendix A to 10 CFR Part 50 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as tornados without loss of capability to perform their safety functions. The Farley licensing bases included this criterion. The installed condition of the safety-related TDAFW pump vent stack did not conform to Farley design and licensing basis requirements. The team noted that the requirements of Criteria III and V of Appendix B to 10 CFR Part 50 which state that applicable regulatory requirements and design basis are correctly translated into drawings, procedures and instructions and that activities affecting quality be prescribed by and performed in accordance with instructions, procedures or drawings were apparently not met. This issue was identified as Unresolved Item 50-348; 50-364/97-201-08.

The licensee stated that the AFW system could withstand the loss of the TDAFW pump in conjunction with a loss of offsite power (LOSP) and a single failure of one MDAFW pump and that the remaining MDAFW pump would have sufficient capacity to safely cool down the plant. The licensee identified the following preliminary action plan to address this issue:

The licensee prepared a draft safety evaluation, which reflected the use of one MDAFW pump to shutdown the plant in the event that the vent stack was damaged. The licensee stated that this approach could result in a potential USQ, requiring prior NRC approval.

An analysis will be prepared to evaluate the effects of the applicable tornado missiles striking the exposed vent stack. This analysis would determine if the TDAFW pump would remain operable after the event. If this analysis shows that the TDAFW pump would remain operable, a safety evaluation will be prepared to revise the FSAR.

If neither of the above items are favorable, then appropriate modifications to the vent stack would be implemented to prevent missile damage.

The licensee issued ABN-97-0-1043 to address this item.

2. Diesel Generator Exhaust Silencers

The team also observed that the safety-related emergency diesel generators and the station blackout diesel generator exhaust silencers for both units were installed outside, on the roof of the diesel generator building. This equipment appeared to be protected from horizontal tornado-generated missiles by the building walls. However, the equipment was susceptible to vertical missiles and other non-horizontal missiles.

FSAR Section 3.5.4 states that Category I equipment and piping outside containment are either housed in Category I structures or buried underground. FSAR Table 3.2-1 addresses the tornado missile protection of the emergency diesel generators. FSAR Table 3.2-1 did not address the exhaust silencers associated with the generators. FSAR Section 3.5.2.1 discusses the types of missiles considered for missile protection. However, the missile spectra were not clearly defined. The team asked if the plant's design bases distinguished between horizontal and vertical/non-horizontal tornado missiles, and if the current design was consistent with the design bases.

The licensee stated that, although the FSAR did not distinguish between horizontal and vertical/non-horizontal missiles, a horizontal missile was evident as the design basis for the plant. However, the licensee could not provide any documentation clearly supporting their statement. This issue will be further reviewed by the NRC to determine if the tornado missile protection in the Farley design and licensing bases included missile spectra other than horizontal missiles. This issue was identified as Unresolved Item 50-348; 50-364/97-201-09. The licensee issued REA 97-1409 to address this item.

3. AFW Flow Control Valves

During a Unit 1 AFW system walkdown on February 6, 1997, the team observed that one of the safety-related, air-operated AFW flow control valves (HV-3227B) had been tagged with a Deficiency Report (DR) 547310 indicating that screws were missing. The team observed that the missing screws were required to hold a solenoid valve to a bracket on the valve actuator. The solenoid valve was temporarily attached to the actuator with tie wrap. The

licensee stated that the DR had been initiated on January 10, 1997 (approximately 3 weeks prior to start of inspection). Following questioning by the team, the condition was corrected on February 6, 1997, via Work Order Number 547310. The team questioned the timeliness of the repair for this safety-related valve.

The licensee stated that DR 547310 had been electronically entered into the DR system when the condition was discovered. This DR had not yet been scheduled for repair as of February 6, 1997. The licensee reviewed the condition of the valve and concluded that the valve had been operable before the repair. Valve HV-3227B was in the supply line from the MDAFW pumps to the "B" steam generator. The AFW flow control valves were normally maintained in a fully open position. These valves were required to be open during an accident to perform their safety function. A failure of the solenoid would result in the valve failing to the open position. In addition, redundant MOVs were located in series with HV-3227B.

The licensee stated that the remaining items in the DR system were reviewed and it was verified that no other potentially significant items related to safety-related equipment existed. After DRs were entered into the DR system they were reviewed and prioritized by a dispatcher. The dispatcher was responsible for identifying any potentially significant items. The team observed that the potential significance of DR 547310 was not identified during the review.

4. <u>CST Level Transmitters</u>

The team inspected the installation of the CST level transmitter Q1P11LT516 and the associated freeze protection. The inspection revealed that the inside walls of the freeze protection enclosure were partially insulated and in need of repair. Drawing A-170256 Sheet 61, Revision 1 showed the sides of the freeze protection enclosure insulated. Eight inches of process tubing at the transmitter housing and transmitter housing were missing the required heat tracing.

The team also observed that the heat tracing power cable appeared not to be securely fastened along the run to the thermostat mounted on the freeze protection enclosure. The cable was observed to be attached to the CST foundation by aluminum bonding tape and not run in conduit. This method was approved by plant drawing B-172374. The review of licensee's actions to address freeze protection installation issue was identified as Inspection Follow-up Item 50-348/97-201-10. The licensee issued Work Request (WR) 97001089 to correct this problem.

In addition, the team verified that Q1P11LT515, LT516 instrumentation installation met the guidance provided in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 2.

5. Other Issues Reviewed

The team observed that cable channel 1VBDDA0A was not connected to its lower support above MDAFW pump motor 1B. Cable 1VBDG10P was routed through 1VBDDA0A to the 1B MDAFW pump motor. The licensee reviewed the installation and concluded that the power cable would be able to perform its intended function in the configuration observed since the seismic accelerations in this area were small. The licensee initiated deficiency report 537759 to correct the channel attachment.

During the inspection of the Unit 1 TDAFW pump, a flexible conduit on the turbine skid was noted as being disconnected from its connector at its junction box. The cable within the flexible conduit fed the manual trip solenoid on the turbine. The licensee evaluated the situation and concluded that the operation of the TDAFW pump was not compromised since the circuitry was used in the manual tripping of the pump. DR 537762 was initiated to repair the disconnected flexible conduit.

c. Conclusions

In general, the AFW system design observed during the walkdown was consistent with the design-basis requirements. However, the team identified that in some cases the design and licensing bases requirements were not fully implemented as required by regulations. The team also identified that further NRC staff review was required to determine if the tornado missile protection in the Farley design and licensing bases included missile spectra other than horizontal missiles

E1.2.7 FSAR, FSD and Other Reviews

a. Inspection Scope

During the Farley design inspection, the team reviewed the FSAR sections and FSDs related to the AFW system and the associated electrical and I&C systems. In addition, the team reviewed licensee's self-initiated safety system assessment (SSSA) for AFW system.

b. Observations and Findings

The team identified the following discrepancies in the FSAR:

1. FSAR Section 6.5.2.2.4 stated, "All valves in the AFW flow path from the condensate storage tank to the steam generators are normally open, with the exception of the fail open AFW flow control valves." This statement implied that the AFW flow control valves were not normally open. In accordance with the plant's operating procedures, these valves were normally maintained in the open position. The FSAR statement was not correct.

2. The team observed that electrical components associated with the safety-related AFW discharge valves HV-3227A, B, C; and HV-3228A, B, C were located in the main steam valve room approximately 4 feet above the floor (elevation 131 feet - 0 inch). The control valves were not above 133 feet, 3 inches as stated in the FSAR. This elevation was above the stated flood elevation for the room; but was not consistent with the FSAR. In addition, the team observed that the limit switches associated with the main steam isolation valves (MSIVs) were located less than one foot above the floor, which was below the stated flood level.

   FSAR Section 3K.4.1.2.7(F) stated that the lowest safety-related equipment in the main steam room was the atmospheric relief valves located at elevation 133 feet, 3 inches. The floor elevation was 127 feet, 0 inch. The AFW flow control valves and the TDAFW pump steam supply valves were also located in the main steam room. FSAR Section 3K.4.1.2.7(F) also stated that the flood elevation in the room would be 3 feet, 5 inches above the 127 feet, 0 inch elevation before the AFW pumps were isolated.

   The licensee stated that the MSIVs were environmentally qualified for submergence and that the location of the auxiliary feedwater valves was acceptable.

3. Section 8.3.3.2 did not reflect the installed fusing for the TDAFW UPS system. The FSAR stated that the rectifier output fusing for the TDAFW UPS was 3 amperes when PCN 81-2125 and DCPs 96-1-9008 and 96-2-9009 changed the fuse size to 5 amperes.

The above discrepancies had not been corrected and the FSAR updated to ensure that the information included in the FSAR contained the latest material as required by 10 CFR 50.71(e). This issue was identified as Unresolved Item 50-348; 50-364/97-201-11. The licensee issued ABN 97-0-1043 to correct and clarify the FSAR.

The team also identified the following discrepancies in AFW FSD A-181010, Revision 1:

1. Section 3.10.2.3 stated that the overspeed trip for the TDAFW pump would prevent pressurization of the system in excess of the design pressure rating. During a TDAFW pump overspeed condition, the system pressure would exceed the design pressure rating, but would be within allowable limits for infrequent operation, as specified in Section III of the ASME B&PV Code.

2. Table T-2 identified the power supply for SV3234B as panel B instead of panel D and did not show the power supply for FISL-3212A as Panel A.

The licensee issued ABN 97-0-1043 to revise the FSD.

The licensee has performed 13 SSSAs to date to determine the system functional performance. A review of the AFW SSSA identified that the licensee's assessment identified many good issues. The open issues resulted from this assessment was satisfactorily resolved except in one instance. Section E1.2.2.1 addresses this issue in greater detail. Although, the assessment identified many issues, licensee's assessment did not identify all of the issues discussed in this report.

c. Conclusions

The team concluded that, in several instances, licensee did not revise the FSAR in accordance with the requirements of 10 CFR 50.71(e). The team also identified several discrepancies during the FSD review. These weaknesses either have been, or are currently being, evaluated by the licensee. Although the safety system self-assessments performed by the licensee failed to identify many of the issues raised by the team, these assessments led to the identification and correction of many other issues.

E1.3 Component Cooling Water (CCW) System

E1.3.1 System Description and Safety Functions

The CCW system was a closed cooling water system which transferred heat to the (SW) system from components which process radioactive fluid. The system was designed to function during normal operation, plant cooldown, refueling, and accident and post-accident conditions. Those portions of the system which cool safety-related components were redundant and safety-related.

The system consisted of three CCW pumps, three CCW heat exchangers, a two-section surge tank, interconnecting piping, and instrumentation. Two CCW pumps, heat exchangers, and both sections of the surge tank were dedicated to two separate trains. The third pump and heat exchanger were capable of being aligned to either train. Manual isolation valves were used to separate the two redundant trains. One pump and one heat exchanger were normally operated to provide cooling for various components in the Auxiliary Building and containment, with another pump on standby. A standby heat exchanger was available should it become necessary to isolate the operating heat exchanger. The third pump and a heat exchanger were isolated from the operating components and on standby to cool engineered safeguards system components if needed. Two CCW pumps and heat exchangers were normally used during cooldown.

During emergency operation, the non-essential loads were shed and the dedicated train was automatically started. Although both trains were used for accident conditions, only one train was required.

E1.3.2 Mechanical Design Review

E1.3.2.1 CCW System Performance

a. Inspection Scope

In this portion of the mechanical design review of the CCW system, the team evaluated the ability of the system to cool safety-related components during both normal operation and accident conditions.

b. Observations and Findings

1. CCW Heat Removal Capability

Calculation 37.7, "Component Cooling Water (CCW) Flow Balance," Revision 0, documented a hydraulic computer model of the system and determined the flows through components cooled by CCW for various modes of operation. The model was calibrated in order to reproduce the test data obtained during normal condition and cooldown condition validation testing. The modes of operation considered were normal, cooldown, loss-of-coolant accident (LOCA) injection, and LOCA recirculation. Single failures and a pipe break in the nonsafety-related portion of the system were considered for each mode as appropriate to cover limiting conditions. This calculation used the pump vendor head-flow curves as input. The team reviewed calculation 37.7 and determined that the methodology and assumptions used were appropriate. Validation for the "BALANCE" computer program was included in the calculation.

Calculation 37.4, "CCW System Heat Exchanger Models and Heat Removal Capacity Calculation," Revision 0, developed models for the heat exchangers cooled by the CCW system. These models were integrated with the flow balance models developed in calculation 37.7 to evaluate the heat removal capacity and the temperatures in the CCW system for various modes of operation. The modes included were normal cooldown, abnormal cooldown (cooldown in one unit, LOSP, and loss of an electrical bus in the cooldown unit), LOCA injection, and LOCA recirculation.

Heat loads for each of these modes were used to calculate CCW temperatures at the outlet of the CCW heat exchangers, which were then used to calculate process temperatures of the components cooled by CCW. The maximum calculated process inlet temperature for the reactor coolant pump (RCP) bearing oil coolers was above the design allowable during a cooldown mode with the loss of an electrical bus. The nuclear steam supply system (NSSS) vendor stated that RCP operation during the cooldown was not a system requirement and that shutting down the RCP before the design limits were reached would be acceptable.

RCP seal water return temperature was also calculated to be higher than the published design allowable during the abnormal cooldown mode. An evaluation by the NSSS vendor determined that the seal water temperatures were acceptable considering the lower reactor coolant temperature when elevated CCW temperatures would be experienced and the component design limits. The ability to cool down the RCS to 200°F in 36 hours was verified. Calculation

37.4 also verified that the design basis SW flow rate of 10,000 gpm to the CCW heat exchanger was adequate. The team reviewed calculation 37.4 and found it adequate. Validation of the "BALANCE - Heat Exchanger Performance Utility" computer program used was included in the calculation.

The above two calculations were dependent on the CCW pump performance curve supplied by the pump vendor. Calculation BM-95-0776-001, "CCW System Evaluation Using Degrade CCW Pump Curves," Revision 0, evaluated the performance of the CCW system with the CCW pumps degraded approximately 10%. This pump performance was referred to as the minimum analyzed pump curve and was used with the in-service testing program. This calculation used the flow model developed for Unit 1 in calculation 37.8 as it resulted in slightly lower flow rates than the Unit 2 calculation did and thus would produce conservative results for the Unit 2 CCW heat exchangers. The thermal model from calculation 37.4 was used to predict process and CCW fluid temperatures using CCW flows calculated from the minimum analyzed pump curves during operating modes of two train cooldown, abnormal cooldown, LOCA injection, and LOCA recirculation. The calculated process and CCW temperatures were evaluated as acceptable. The team reviewed this calculation and found it adequate.

2. Net Positive Suction Head (NPSH)

The team reviewed calculation 34.5, "Component Cooling Water System NPSH (ES No. 90-1820)," Revision 0. This calculation concluded that substantial margin existed between the NPSH required and the NPSH available for the CCW pumps. The calculation stated that the flow through the pipe from the CCW pump suction header to the CCW surge tank would be negligible under normal operation and that there was no large leak from the system requiring significant flow from the tank. Therefore, no head loss was assumed in this pipe. However, calculation 39.3, "CCW Surge Tank Analytical Limit for Level Setpoint," Revision 0, calculated a flow of 801 gpm through this pipe for the period from a pipe break occurring in the nonsafety-related portion of the CCW system and automatic isolation of the break (10 seconds from the low-low level setpoint in the CCW surge tank). This flow resulted in a very small reduction of the NPSH available of less than one %. The team concluded that there was adequate NPSH available to the CCW pumps in all modes of system operation.

3. Single Failure Design

The team reviewed the capability of the system to accommodate a single failure in conjunction with postulated plant accidents. The mechanical design included adequate valving to enable the two trains of the system to be separated for combinations of operating and standby CCW pumps and heat exchangers. The team reviewed the applicable portions of the following operating procedures for the CCW system and SW system, and determined that the system was properly operated to ensure the ability to accommodate a single failure.

- FNP-2-SOP-23.0A, "System Operating Procedure Checklist, Component Cooling Water System," Revision 5

- FNP-2-SOP-23.0, "System Operating Procedure, Component Cooling Water System," Revision 39

- FNP-2-SOP-24.0, "System Operating Procedure, Service Water System," Revision 31

4. Flooding of CCW Components

The team reviewed the potential for flooding of the room which contained the CCW pumps and heat exchangers. Bechtel letter AP-17096, dated Dec. 22, 1989, "CCW Heat Exchanger Room Flooding," reported the results of a flooding evaluation and concluded that there was no concern for flooding in this room since there were no credible line breaks or other sources of flooding present. The team confirmed that the design-basis criteria used to postulate pipe breaks was appropriate and conducted a walkdown of the room and adjacent areas which further confirmed the results of the evaluation.

5. Pipe Stress and Support Analyses

The team reviewed the CCW fluid temperatures used in the pipe stress and support analyses. The licensee had discovered a discrepancy in a Unit 1 stress calculation during analyses performed in support of a snubber reduction program. The normal operating temperature was used instead of the higher temperatures that could occur during a design-basis accident. The licensee reviewed the stress calculations for the CCW systems for both units and determined that the maximum operating temperature was generally higher than the temperature used for the stress analyses. The licensee stated that, as substantiated by an initial assessment, the CCW system for both units met the ASME Code allowable limits.

The team reviewed the maximum operating temperatures used in the assessment of the Unit 2 CCW system and noted some inconsistencies in the data. The licensee reviewed the maximum operating temperature data for both units and corrected the data as required. The corrected data did not change the licensee's conclusion that the systems met the ASME code allowable.

The licensee established a screening criteria to determine actions required for the existing pipe stress calculations. The licensee determined that negligible impact on the results of the existing analysis would exist if the difference between the current analysis temperature and the maximum operating temperature was less than 20°F. Calculation Change Notices were being prepared to document the revised temperature for each affected calculation so that the corrected temperature would be used if the calculation was revised in the future. The licensee stated that pipe stress calculations for which the difference between the current analysis and maximum operating temperatures was greater that 20°F would be revised and the support loads recalculated to ensure that design allowable values were met.

The licensee issued Deficiency Notice 97-001, "Incorrect Dimensions and Temperatures Used in CCW Pipe Stress Analysis," to document the conclusions reached and recommended that a Root Cause Evaluation Team be formed to determine the root cause and the broadness of this event. The licensee also issued REA 97-1407 to revise the Unit 1 and 2 stress calculations. The adequacy of licensee's root cause evaluation and the corrective actions to revise the affected calculations were identified as Unresolved Item 50-348; 50-364/97-201-12.

c. Conclusions

The team concluded that the CCW system was capable of performing the design-basis functions of cooling safety-related equipment during the various plant operating modes for which it was designed. The design margin has been reduced because of the discrepancy in the temperatures used in the pipe stress and support analyses.

E1.3.2.2 CCW Surge Tank

a. Inspection Scope

In this portion of the mechanical design review of the CCW system, the team evaluated the surge tank to determine if it provided an adequate water supply until a pipe break in the nonsafety-related portion of the system can be isolated. In addition, the team evaluated whether the surge tank was provided with adequate pressure relief and vacuum breaker capability to maintain the required tank operating pressure within design limits.

b. Observations and Findings

The team reviewed calculation 39.3, "CCW Surge Tank Analytical Limit for Level Setpoint," Revision 0. This calculation determined the tank level for the low-low level setpoint which would result in the tank becoming empty in the event of a pipe break in the nonsafety-related portion of the system. The CCW pumps had adequate NPSH available when the tank was just empty. The minimum level setpoint determined by this calculation was conservative when compared to the actual setting. The calculation used the results of calculation 38.6, "Determine Flow Rate Through Pipe Break in the CCW System," Revision 0, as input for the maximum flow rate out of the surge tank. The team reviewed this calculation also and concluded both calculations were adequate.

The surge tank was protected from overpressure by relief valve PSV-3029, which was designed to limit the internal tank pressure to 14 psig when subjected to the flow which could result from a rupture of a reactor coolant pump thermal barrier cooling coil. The team verified that this flow was less than the 300 gpm for which the relief valve was designed. The relief valve discharge was piped to the floor drain tank in the Waste Processing System. The team reviewed the portion of calculation 12.19, "Nuclear Relief Valves Sizing," Revision 1, that dealt with PSV 3029 and walked down the routing of the discharge piping. The team verified that the back pressure that could exist on the relief valve at its maximum capacity was less than that assumed in the valve sizing calculation and that the design input to the calculation was

correct. The sources of makeup to the surge tank were the Demineralized Water and Reactor Makeup Water systems. The team verified that neither system was capable of providing a flow to the surge tank in excess of the relieving capability of PSV 3029.

The tank was protected against external pressure conditions, which could exist if the tank vent was isolated during an outflow, by redundant vacuum breaker valves which would open at less than 1 psid.

c. Conclusions

The surge tank provided adequate water volume for the system and was adequately protected against pressures outside of its design bases.

E1.3.2.3 CCW System Containment Isolation

a. Inspection Scope

In this portion of the mechanical design review of the CCW system, the team evaluated the licensee's containment isolation provisions for the CCW pipes that penetrate containment. In particular, the team focused on correct implementation of the design bases for such penetrations.

b. Observations and Findings

The CCW system penetrated containment for the supply and return of cooling for the reactor coolant pumps (RCPs) and the excess letdown and reactor coolant drain tank heat exchangers. Motor-operated, air-operated, and check valves were used as containment isolation valves. The team verified that the containment penetration arrangements were designed to accommodate a single active mechanical or electrical failure and that containment isolation capability was not lost as a result of a failure of the nonsafety-related IA system. The team verified that the containment penetrations and the piping inside containment were adequately protected against the effects of overpressure that could occur during LOCA containment temperatures.

The licensee's response to NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated January 27, 1997, was reviewed. This response stated that the only CCW penetration requiring review for potential overpressure was the return line from the RCP thermal barrier (penetration 43), and that valve HV-3184 will unseat and relieve pressure in the penetration to a section of piping protected by a relief valve before the penetration piping allowable pressure was exceeded. The team reviewed the valve specification sheet, A-J-300-13, "CCW from RCP Thermal Barrier" and memoranda REA 96-1286 and ENG 51-1 dated January 14, 1997, which documented that the valve lift pressure was less than the piping design pressure. The team found that this documentation supported the conclusion that penetration 43 was adequately protected against overpressure.

The team reviewed the design-basis differential pressures in the MOV Design-Basis Document, U 418109, Revision A, for the motor-operated containment isolation valves, MOV 3046, 3052, and 3182. The design-basis differential pressure was defined in U 418109 as the maximum differential pressure that the valve could have to open or close against. These valves were located in the piping at penetration numbers 42 and 44, which were the CCW supply to the RCPs and the CCW return from the RCP oil coolers. FSAR Table 6.2-31 identified these penetrations as Type II. FSAR Section 6.2.4.1 defined Type II penetrations as serving those lines that connect directly to the containment atmosphere.

The CCW lines which run through penetrations 42 and 44 could become directly connected to the containment atmosphere as a result of the dynamic effects of a LOCA inside containment. Therefore, the design-basis differential pressure for the containment isolation valves should consider the maximum post-LOCA containment pressure that could exist when the valves operate. The design-basis differential pressures in U 418109 were 24 psid for MOVs 3046 and 3182, and 9 psid for MOV 3052. The licensee stated that the maximum pressures could be 52 psid for MOV 3052 and 27 psid for MOVs 3046 and 3182 when the effect of containment pressure was considered.

The licensee stated that the thrust setpoints for these valves were such that they could close at differential pressures of a minimum of 2.75 times greater than the higher design-basis differential pressures. (This resulted from the confidence band methodology used by the licensee to determine the closing thrust requirements.) The team noted that the requirements defined in Appendix B to 10 CFR Part 50, Criterion III, regarding verifying and checking the adequacy of the design were not followed for determination of the design-bases differential pressure for the CCW motor-operated containment isolation valves. This issue was identified as Unresolved Item 50-348; 50-364/97-201-13. The licensee issued ABN 97-0-1044 to revise the MOV Design-Basis Document and related documentation.

c. Conclusions

Containment isolation of the CCW lines penetrating the containment generally met the design-basis requirements and would adequately perform its safety function. However, the team identified that in one instance the licensee did not properly verify or check the adequacy of the design-basis differential pressures used for MOVs.

E1.3.2.4 <u>CCW System Testing</u>

a. Inspection Scope

In this portion of the mechanical design review, the team evaluated the licensee's testing of CCW system components. In particular, the team's objective in conducting this evaluation was to verify that IST program included components intended to perform a safety function (as required by Section XI of the ASME B&PV Code), that the testing adequately verified the design function of the components, and that TS requirements for testing were appropriately implemented.

b. Observations and Findings

TS 4.7.3 requires periodic verification of accessible valves in the flow paths servicing safety-related equipment that are not locked, sealed, or otherwise secured at least every 31 days. The team reviewed surveillance procedure FNP-2-STP-23.7, "Component Cooling Water Flow Path Verification Test," Revision 12, and confirmed that it implemented the TS surveillance requirement for flow path testing. TS 4.7.3 also requires periodic verification during shutdown that each automatic valve servicing safety-related equipment actuates to its correct position on a safety injection test signal. The team reviewed surveillance procedure FNP-2-STP-40.0, "Safety Injection with Loss of Offsite Power Test," Revision 25, and confirmed that it implemented this TS surveillance requirement for CCW valves.

TS 4.0.5 requires testing of ASME Code Class 1, 2, and 3 pumps and valves in accordance with Section XI of the ASME B&PV Code. This code required testing of valves and pumps that perform a safety function. The team verified that the required CCW pumps and valves were tested by a review of the following surveillance procedures:

- FNP-2-STP-23.8, "Component Cooling Water Valve Inservice Test," Revision 17

- FNP-2-STP-23.12, "CCW to RCP Thermal Barrier Check Valves Reverse Flow Test," Revision 2

- FNP-2-STP-23.1, "2A Component Cooling Pump Quarterly Inservice Test," Revision 14 (This procedure included testing of the pump 2A discharge check valves and was typical for the three CCW pumps.)

The licensee also performed performance testing for the CCW heat exchangers. Of these tests, the team reviewed FNP-0-ETP-4379, "Performance Test for Unit 1 and 2 Component Cooling Water Heat Exchangers," Revision 5, and determined that it yielded appropriate results founded on acceptance criteria consistent with the design-basis calculations.

Inservice testing of the CCW pumps was performed in accordance with FNP-2-STP-23.1, "2A Component Cooling Water Pump Quarterly Inservice Test," Revision 14, and similar procedures for the other two pumps. The procedure was performed with the system in a normal alignment which required the pump minimum flow recirculation lines be in service. Data for pump differential pressure and flow measured downstream from the recirculation line were recorded and compared to a table of acceptable differential pressures and corresponding flows in the procedure.

Calculation BM-95-0776-001, "CCW System Evaluation Using Degraded CCW Pump Curve," Revision 0, documented the acceptability of the CCW system performance with the pumps degraded approximately 10% from the vendor test curves and stated that this degraded pump curve will be used for comparing test data to verify the performance capability of the CCW pumps. However, the table of acceptable performance values in procedure FNP-2-STP-23.1 was different than the degraded pump curve used in the calculation.

The team noted that the acceptance values in the procedure considered the pump recirculation lines in service and the degraded curve in the calculation was for a condition with the recirculation lines closed and questioned how the acceptable test values in the procedure accounted for the design-basis pump degradation. The licensee stated that a special test was performed, FNP-2-ETP-4416, which obtained flow and pressure data with the recirculation line closed at a single operating point. These data were compared to the design-basis degraded pump curve. If the margin in pressure was greater than 10%, the 90% acceptance data in the IST procedure was deemed appropriate; if less than 10%, adjustments were made to the IST acceptance data so that it would be within the design-basis degraded pump curve. (The 90% value was an ASME XI standard reference point.) These data required adjustment for only pumps 1A and 1B. The team found that the above process resulted in acceptable test acceptance criteria for the CCW pumps. This process was not completely documented.

Calculation BM-95-0776-001 was performed under REA 95-0776. The response to this REA was issued on May 22, 1996, by Bechtel letter AP-21413. A safety evaluation and a proposed ABN for revising the FSAR, affected FSDs, and affected drawings were attached to this letter. The licensee stated that the IST procedure will be revised to include the degraded curve after the safety evaluation was approved by the Plant Operations Review Committee (PORC). The licensee stated that an ABN will be issued to revise the appropriate documents once PORC approval has been obtained. The review of IST procedure and applicable safety evaluation was identified as Inspection Follow-up Item 50-348; 50-364/97-201-14.

c. Conclusion

The team found that surveillance requirements in the TS for safety-related pumps and valves had been appropriately implemented by procedures and that the CCW heat exchangers were tested to verify the performance required by the design bases.

E1.3.2.5 <u>CCW System Modifications</u>

a. Inspection Scope

In this portion of the mechanical design review, the team evaluated 10 CCW design modifications to assess their consistency with the design and licensing bases. In particular, this review considered the licensee's 10 CFR 50.59 evaluations, post-modification testing, and sample of the affected documents that required updates to reflect the given design change.

b. Observations and Findings

The team determined that the 10 CFR 50.59 evaluations for the modifications reviewed were appropriate and changes to required design and licensing basis documentation were identified and implemented. The design bases for the component(s) affected by the change were correctly identified. Except for the instance discussed below, appropriate post-modification testing was performed.

Design Change Package (DCP) 96-0-9012-2-006, "Process Coating for CCW Heat Exchangers," provided direction for modification of the CCW heat exchangers by application of an epoxy coating (Plastocor) to the tubesheets, channel head, channel cover, channel head shell relief line, approximately 12 inches into the service water inlet and outlet lines, and 12 inches of the inlet end of the tubes. The DCP, along with REA 96-1211, also provided direction for plugging and stabilizing tubes. Procedure FNP-0-ETP-4418, "CCW Heat Exchanger Epoxy Coating Application," Revision 1, implemented the epoxy coating and Work Orders 96001476, 96001477, and 96001478 installed the stabilizing rods in the tubes. The team reviewed calculations SM-96-9012-002, "Effects of Plastocor Coating on CCW Hx's Thermal Performance," Revision 0, which determined an equivalent length for a tube partially coated with epoxy, and SM-96-9012-004, "Effects of Plastocor Coating on CCW Hx's Thermal Performance," Revision 0, and identified no concerns with these calculations.

The team identified the following discrepancies with this modification:

1. Commercial grade dedication of the Plastocor epoxy material included a pull-off test of some samples at the vendor facility. The application procedure used for these coating samples was not documented. Such documentation would provide a method to ensure that the commercial grade dedication tests were representative of the modification performed at Farley. The licensee obtained documentation from the vendor during the inspection that the application method used for the samples was the same as that used for the actual work at Farley. Additionally, the licensee stated that Plastocor personnel performed the application in both locations.

2. No post-modification testing to ensure design flow capability had been maintained was required by the procedure used to apply the epoxy coating nor in DCP 96-9012-2-006. The team noted that an earlier version of the DCP stated that post-modification flow testing would be performed and this statement was deleted in a subsequent revision to the DCP. Although the application method appeared to limit the epoxy to the intended surfaces, unexpected/unknown occurrences could cause the epoxy to hinder flow through the tubes. The epoxy was applied to the first 12 inches of each tube and this area inspected in-process for correct application. The work procedure did not require any inspection of the remaining portion of the tubes (tube length is approximately 29 feet), nor any flow testing.

   The licensee stated that the heat exchangers were returned to service with no problems indicated, including no abnormal flow indication. The team noted that reduction in flow greater than that which would be equivalent to the tube plugging limit might not be detected on installed flow instruments as the reduction could be within the instrument error band. The only post-modification tests recorded on Work Authorizations for the heat exchangers, numbers 123723, 121390, and 123724, were leak tests. The team was concerned that the design-basis flow capability of the heat exchangers was not verified after the modification was completed and before the equipment was returned to service.

Farley Support Procedure GO-M-1,"Designer Interface Document," Revision 3, states that design input requirements are in accordance with Section 3.2 of American National Standards Institute (ANSI) Standard N45.2.11, which states that design input shall include test requirements including in-plant tests and the conditions under which they will be performed. Farley Nuclear Plant Administrative Procedure FNP-0-AP-8, "Design Modification Control," Revision 22, Section 9.1.5, requires identification of tests which will be required to verify acceptable performance of plant components and systems added, modified or affected by incorporation of the design change into the plant.

In addition, the Farley Nuclear Plant Administrative Procedure FNP-0-AP-70, "Conduct of Operations Plant Modifications and Maintenance Support," Revision 4, Section 4.19, requires that the functional test procedure shall include as a minimum, sufficient information and steps to adequately test and document the proper function of the equipment added or modified by the design change.

The design control measures for the design change did not ensure that design requirements were correctly translated into instructions procedures, or drawings and accomplished in accordance with drawings, procedures or instructions as required by Criteria III and V of Appendix B to 10 CFR Part 50. The lack of adequate post-modification testing was identified as Unresolved Item 50-348;50-364/97-201-15.

3. The team noted several discrepancies in calculation SC-96-1211-002, "CCW Heat Exchanger Maintenance Repairs," Revision 1. This calculation documented the seismic and mechanical acceptability of the modification.

Section 2.1 determined a net increase in weight of the heat exchanger of 13.85% and compared this increase to a 15% increase stated to be acceptable by the vendor. However, the 13.85% incorrectly subtracted the weight of the water displaced by the modification, as the 15% allowable limit was defined on the basis of a dry heat exchanger. The licensee stated that the correct weight increase to be used for comparison was 14.86%.

Section 2.3 evaluated the heat exchanger shell and supports and used 14.09% as the increase in weight between the tubesheets following the modification. This value should have been 14.86% as discussed above.

Section 2.4 concluded that the small increase in weight associated with the modification would not have a significant impact on the seismic response of the system, and therefore the foundation anchorage of the heat exchangers would remain adequate. This conclusion was an engineering judgement and the reduction in margin for the foundation anchorage was not documented against the foundation base calculation, U-405165. The licensee reviewed the anchorage design at the request of the team and stated that margin existed in the anchorage with the added weight of the modification considered.

Section 2.5 referenced a data sheet from the plug vendor, EST, containing design information for the Perma Pop-a-Plugs used to plug the tubes. This data sheet was neither attached to the calculation nor specifically identified therein.

None of these discrepancies changed the results of the safety analysis or conclusions of the calculation. The licensee issued REA 97-1407 to revise the calculation. The review of revised calculation addressing the above discrepancies was identified as Inspection Follow-up Item 50-348; 50-364/97-201-16.

4. Minor administrative errors were made in preparation of the DCP. The Design Input Record indicated that the heat exchangers were not referenced in or identified by any design-basis documents; contrary to the existence of several design-basis calculations concerning the heat exchangers. The DCP calculation record omits calculation SM-96-9012-004 and has the wrong title for calculation SC-96-1211-002.

c. Conclusions

The modifications reviewed maintained the design and licensing bases of the system and contained appropriate safety evaluations. The modifications were properly implemented and documented except for the documentation, calculation, and testing discrepancies identified concerning the CCW heat exchanger modification.

E1.3.2.6 Other Related CCW System Review

a. Inspection Scope

In this portion of the mechanical design review, the team evaluated LERs, NRC Notices of Violation, CCW System SSSA observations, and licensing commitments for appropriate follow-up action. Licensee practices for updating drawings were reviewed, as well as valve lineup procedures.

b. Observations and Findings

The team identified no concerns with licensee actions with regard to the LERs, Notices of Violation, and licensing commitments reviewed. The licensee's safety system self-assessment identified many good issues. Follow-up actions noted in these documents were verified. The issues were resolved satisfactorily except in one instance. This is discussed in the following paragraph. Although the safety system self-assessments performed by the licensee failed to identify and correct many of the issues raised by the team, these assessments led to the identification and correction of many other issues.

Procedures FNP-2-SOP-23.0A, "Component Cooling Water System," Revision 5; FNP-2-SOP-2.1A, "Chemical and Volume Control System," Revision 8; and FNP-2-SOP-1.1A, "Reactor Coolant System," Revision 6, were checklists for the normal positions of valves and circuit breakers. The team identified numerous differences between the P&IDs for the system (D-205002 Sheet 1, Revision 21;

Sheet 2, Revision 10; and Sheet 3, Revision 2) and procedures FNP-2-SOP-23.0A and FNP-2-SOP-2.1A concerning the existence of caps on vent and drain lines. The team noted that item 5 of SSSA observation CCW-CM-01 was related to this item in that it discussed a discrepancy between FSAR Section 9.2.2.3 and the checklists in FNP-2-SOP-23.0 that was apparently not resolved. The SSSA was issued on April 19, 1990. The licensee issued ABN 97-0-1044 to update the P&IDs, and stated that the SOP valve checklists which included CCW valves would be updated to match the drawings and that field verification would be performed when the SOP checklists were implemented. The requirements specified in Criterion XVI of Appendix B to 10 CFR Part 50 which state that conditions adverse to quality such as malfunctions and deviations be promptly identified and corrected were apparently not met. This issue was identified as Unresolved Item 50-348; 50-364/97-201-17.

The team questioned if a controlled list of drawings that were maintained as-built existed or how such drawings were otherwise identified. The licensee stated that drawings which were not maintained as-built were not necessarily marked accordingly and no procedure existed to identify which drawings were as-built. The team was concerned that design changes and/or decisions could use drawings which did not reflect the as-built plant. The team did not identify any issues because of the lack of this information. The licensee stated that only three categories of drawings were not as-built (conduit layout for Class 1E structures, telephone backboard, and bills-of-material). The licensee had identified this problem before the inspection, and had started developing a list of all currently maintained drawings as well as those no longer updated for distribution to project personnel. The licensee issued REA 97-1410 to finish developing this list.

c. Conclusions

Licensee follow-up actions for the LERs, Notices of Violation, and licensing commitments reviewed were satisfactory. Weaknesses were identified in the control of as-built drawings and corrective action for SSSA observation concerning the existence of caps on vent and drain lines.

E1.3.3 Electrical Design Review

E1.3.3.1 CCW Electrical Loads

a. Inspection Scope

In this portion of the electrical design review, the team evaluated the electrical loads required for the CCW system to perform its intended design functions under both normal and accident conditions. In particular, this evaluation addressed cable sizing, protective coordination, electrical bus loading, and dc battery loading calculations.

b. Observations and Findings

To ensure that the CCW loads had been properly identified, the team reviewed calculations SE-94-0470-007, "Farley Unit 2 As-Built Load Study Update," Revision 0; SE-94-0470-005, "As-Built Load Study Summary Calculation,"

Revision 1; E-42, "Steady State Diesel Loadings Calculation for LOSP, SI and SBO," Revision 8; E-35.2.A, "Setting of Protective Relays for FNP Unit 2 4.16 kV Auxiliary Power System," Revision 0; and E-95, "Battery Capacity Calculation for LOSP and LOSP+LOCA Situations and Limiting Battery Load Profile," Revision 7. All major electrical loads were accounted for within these calculations for normal and accident conditions. The frequency and voltage ratings of the CCW equipment were compatible with the electrical design analyses. The team observed that the methodology and assumptions used were appropriate.

Cable sizing for the 4 kV CCW pump motors was reviewed. The review compared the actual cable data for the CCW pump motors against the "Electric Power Cable Sizing Guide," Revision 0. The review determined that cable sizing for the CCW pump motors was adequate and in conformance with the plant cable sizing criteria. Protective coordination for the 4 kV CCW motors was reviewed and found to be in accordance with plant drawings and setting sheets. Calculation E-95 was reviewed by the team to ensure that the CCW dc loading was evaluated. All required loads were identified in the calculation.

c. Conclusions

The electrical design for components that perform the normal and emergency functions of the CCW system was adequate. The Electrical Distribution System provided independent, redundant, safety-related (Class 1E) power to the CCW electrical loads.

E1.3.3.2 <u>CCW Swing Pump Operation</u>

a. Inspection Scope

In this portion of the electrical design review, the team evaluated the ability of the train B CCW pump to be aligned to either electrical redundant power supply (train A or train B) in accordance with the engineering design bases.

b. Observations and Findings

The team reviewed FSD A181000, "Functional System Description Component Cooling Water System," Revision 6, and logic diagrams B-205810 (sheets 22,23,100, and 101). CCW pump motors C and A were dedicated to 4 kV Train A and Train B respectively. CCW pump B motor could be aligned either to Train A or Train B class 1E 4 kV electrical buses. A review of the logic drawings referenced and a plant walkdown of the installation revealed that a cross-connection of the two electrical trains (A and B) could not be achieved. Keylock operated disconnect switches were installed and interlocked to prevent cross-connection of the two electrical trains.

c. Conclusions

The team concluded that adequate physical controls as well as electrical interlocks existed in the design of the swing CCW pump B to ensure that the design-basis separation of the redundant class 1E Trains A and B would be maintained.

E1.3.3.3 Electrical Modifications

a. Inspection Scope

In this portion of the electrical design review, the team evaluated electrical modifications to the CCW system and verified that the changes were in conformance with the CCW and electrical design bases.

b. Observations and Findings

The following modifications associated with the CCW system were reviewed by the team: ABN 96-0-0930, "Change CCW Pump Room Equipment Maximum Temperature," and Procurement Deviation Evaluation (PDE) 93-0-0029, "Replacement Motor for CCW System."

PDE 93-0-0029 evaluated motor replacements for Limitorque MOVs Q2P17MOV3185A and B. The evaluation ensured that the original vendor (Limitorque) certified that the replacement motors met the environmental, electrical, and seismic design as originally specified. The appropriate documentation was specified in the purchase requisition (x253631) and the motors purchased met all design criteria of the original motors.

ABN 96-0-0930 updated FSDs A-181000 and A-181004 to reflect increased room temperatures when various room coolers were out of service for maintenance. REA 95-0873 documented the results of an engineering evaluation which allowed temperature increases for the motor control center (MCC) room for MCC 1B and 2A, battery charger A, B, and C rooms, and the CCW pump room. The team reviewed REA 95-0873 and verified that adequate justification existed for the equipment to operate in these elevated temperatures during normal and accident conditions. Documentation was available to support the engineering evaluation. A 10 CFR 50.59 evaluation was performed which identified changes to the FSAR. The appropriate sections of the FSDs incorporated ABN 96-0-0930.

c. Conclusions

The electrical modifications reviewed by the team for CCW were adequate. The design bases as delineated within FSDs and the FSAR were maintained.

E1.3.4 Instrumentation & Controls (I&C) Design Review

E1.3.4.1 CCW Surge Tank Level Setpoint

a. Inspection Scope

In this portion of the I&C design review of the CCW system, the team evaluated calculation CS-L3027C, "Instrument Loop Uncertainty and Setpoint Calculation," Revision 0. In particular, the team assessed the adequacy and consistency in relation to the setpoint program, GO-M-1.

b. Observations and Findings

This uncertainty calculation was performed for Unit 1 to evaluate the existing low-low level setpoint of 20 inches above the tank bottom. The results also applied to Unit 2. The calculation evaluated CCW surge tank low-low level actuation requirements and verified appropriate stroke times for CCW isolation valves Q2P17HV-3096A and B. The evaluation was required to ensure adequate NPSH for the on-service train CCW pump without a need for make-up water being supplied to the CCW surge tank. The team found the calculation adequate and consistent with Farley's setpoint uncertainty program.

c. Conclusions

The team determined that low-low setpoint in this calculation was correct and the methodology of the calculation was consistent with Farley's setpoint program.

E1.3.4.2 Instrument Loop Uncertainty Calculations

a. Inspection Scope

In this portion of the I&C design review, the team evaluated the consistency and adequacy of the licensee's setpoint bases and uncertainty calculations for various instrument loops in the CCW system.

b. Observations and Findings

The team observed that nearly all of the CCW instrument loops lacked calculations documenting uncertainty. The lack of calculations made it difficult to judge whether the methodology that went into the development of the original setpoints conformed to the current setpoint program. The licensee developed three preliminary uncertainty calculations to show that original setpoints and margins demonstrated the conservatism included in the design of the Farley systems.

Loop F3045, "CCW Return Flow from the RCP Thermal Barrier" (high-flow setpoint), calculation documented that there was adequate margin in the setpoint considering the loop uncertainties. The licensee also noted that the ± 15 gpm activation range (the setpoint tolerance) could be reduced to conform to a more conservative calibration tolerance consistent with the capabilities of the instrument. This reduction would have a slight effect on the

percentage of flow equation on page 3 of the calculation, but would have minimal affect on the total error. The slight effect on the flow equation would yield more margin. The licensee stated that the final version of Loop F3045 would correct these deficiencies.

Loop L3027, "CCW surge tank level setpoints" calculation documented that the low and low-low level CCW surge tank setpoints would perform their intended function considering the loop uncertainties by alarming and providing actuating interlocks to Q2P17HV-3096A/B. Loop P3184, "CCW return pressure from RCP thermal barrier" (high-pressure setpoint), calculation documented that the pressure switch was set to maintain maximum sensitivity to a thermal barrier cooling coil failure considering the loop uncertainties, but high enough to prevent spurious actuation as a consequence of normal process pressure changes. The team had no concerns with these two calculations.

c. Conclusions

The calculations proved to be adequate and consistent with Farley's current setpoint program. The methodology was consistent with the I&C design criteria and industry standards. The calculations had built-in conservatism and margin particularly with the associated surveillance test procedure data sheets. Future setpoint changes can be adequately calculated and process limits can be reconstituted and used for these changes using the current setpoint program.

E1.3.4.3 Setpoint Indexes and Other Reviews

a. Inspection Scope

In this portion of the I&C design review, the team evaluated the consistency of setpoint index B-205968 and the field instrument setpoint index from the Westinghouse balance of plant (BOP) manual (Farley drawings U-262166 and U-262167). In addition, the team also evaluated licensee's compliance with Regulatory Guide (RG) 1.97, "Instrumentation for light-water-cooled nuclear power plants to assess plant and environs conditions during and following an accident," Revision 2.

b. Observations and Findings

The team observed that there were discrepancies in setpoint data between the setpoint index and the Westinghouse BOP manual field instrument index found on drawings U-262166 and U-262167, Revision 3. The team found that the column titled "setpoint unit" for level instrument LSLL 3027C on drawing U-262166 noted 27 inches. Similarly for LSLL 3027D the "setpoint unit" was 27 inches on drawing U-262167. Both of these setpoints should be 20 inches, as stated in the setpoint index. The team found that the low-low setpoint of 20 inches had been used correctly in the surge tank level and calibration range calculation (CS-L3027C). Farley procedures FNP-2-IMP-210.6, Revision 8, "Component Cooling Water Surge Tank Level Loop calibration" Q2P17LT3027C and FNP-2-IMP-210.7, Revision 8, "Component Cooling Water Surge Tank Level Loop Calibration" Q2P17LT3027D were also reviewed for consistency. Each associated data sheet in each maintenance procedure correctly noted 20 inches tank level. The licensee issued ABN 97-0-1044 to correct the Westinghouse manual.

The team reviewed the RG 1.97 Category 1 compliance checklists for the CCW heat exchanger inlet flow instruments, FT 3043A, B, and C and the component cooling water heat exchanger discharge temperature instruments, TE 3042A, TE 3042B, and TE 3042C. RG 1.97 stated that the indication for the CCW flow was considered a Category 2, type D variable with a required scale range of 0 to 110% of the design flow. The team verified that the scales for the CCW flow indicators met this requirement. The range of the temperature elements (32°F to 200°F) was sufficiently sensitive in the normal operating range. The team found the checklists and instrument design were consistent with RG 1.97 guidance.

c. Conclusions

The team found that the setpoint indexes were consistent and in accordance with the design basis except for one minor discrepancy which did not adversely affect safe plant operation. The post accident monitoring instrumentation met the RG.1.97 guidance.

E1.3.5 System Interface Design Review

E1.3.5.1 Service Water (SW) System

a. Inspection Scope

In this portion of the system interface design review, the team evaluated the ability of the SW system to supply cooling water to the CCW heat exchangers. In particular, the team reviewed FSAR Section 9.2.1, "Station Cooling Water System," and Section 9.2.5, "Ultimate Heat Sink"; the P&IDs; portions of FNP-2-SOP-24.0, "Service Water System," Revision 31; and calculation SM-ES-89-1500-007, "Bounding Service Water Inlet Temperature Profile," Revision 0.

b. Observations and Findings

The team found the documents reviewed satisfactory and did not identify any concerns with the interface between the SW and CCW systems. The SW system design for single failure was consistent with that of the CCW system and SW design temperatures and flow rates were appropriately utilized in the CCW design.

c. Conclusions

The team concluded that the design of the interface between the SW and CCW systems was satisfactory and that the SW system operation adequately supported CCW system operation.

E1.3.5.2 Instrument Air (IA) System

a. Inspection Scope

In this portion of the design review, the team evaluated the interfaces between the IA system and the CCW air-operated valves. In particular, the team reviewed FSAR Section 9.3.1, "Compressed Air System," data sheets for the

CCW air-operated valves from control valve specification SS-1102-36, and FNP-2-AOP-6, "Loss of Instrument Air," Revision 15.

b. Observations and Findings

The IA system was not safety-related and the team determined that loss of the system would not prevent the CCW system and its components from performing their safety functions. The team did not identify any concerns with the interface between the IA and CCW systems.

c. Conclusions

The team concluded that the design of the interface between the IA and CCW systems was satisfactory.

E1.3.5.3 Auxiliary Building Ventilation (ABV) System

a. Inspection Scope

In this portion of the system interface design review, the team evaluated the ability of the ABV system to provide acceptable ambient conditions for operation of the CCW system. In particular, the team reviewed FSAR Section 9.4.2, "Auxiliary Building," and portions of FNP-2-SOP-58.0, "Auxiliary Building HVAC System," Revision 26.

b. Observations and Findings

The room containing the CCW pumps and heat exchangers was served by the nonsafety-related non-radioactive heating and ventilation system and redundant safety-related coolers cooled by service water. REA-0873, "Auxiliary Building Room Coolers Attendant Equipment Evaluation," documented that the CCW equipment in the CCW room could operate if one cooler was out of service in a design-basis accident situation where a single failure causes loss of the other cooler. The team did not identify any concerns with the interface between the ABV and CCW systems.

c. Conclusions

The team concluded that the design of the interface between the ABV and CCW systems was satisfactory.

E1.3.6 System Walkdown

a. Inspection Scope

The system walkdown included examination of the CCW pumps, heat exchangers, and other CCW equipment in the CCW room; the CCW surge tank; and the CCW valves and piping in the containment penetration area outside containment. The team also verified the consistency of selected portions of the system with plant drawings, as well as the consistency of the instrument calibration intervals with licensing documents. In addition, the team verified the as-built designs and identified RG 1.97 instruments.

b. Observations and Findings

The team determined that the overall material condition of the plant areas was good. The equipment sampled matched the design requirements. The team verified that instrumentation installation met the guidance provided in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 2.

The team observed that the calibration frequency of the SW pressure transmitter Q1P16PT3001B had a 3-year calibration interval. This calibration interval appeared to be excessive. The team was concerned that an instrument calibrated every three years could drift out-of-tolerance. An extended calibration interval usually would require a supporting analysis of drift error. Since the licensee could not produce an uncertainty calculation for the instrument loop PT 3001B, the team questioned the current calibration interval. The team found that the" M-85" maintenance program was tracking this device. The program flagged any device if any two of the preceding four calibration intervals exhibited out-of-tolerance conditions. There had been one calibration performed on Q1P16PT3001B since the "M-85" program was implemented and one calibration before that recorded an out-of-tolerance as-found calibration. The licensee issued ABN 97-0-1410 to implement a revised calibration interval.

c. Conclusions

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

E1.3.7 FSAR and FSD Review

a. Inspection Scope

In this portion of the design review, the team evaluated the FSAR sections and FSDs related to the CCW system and the associated electrical and I&C systems.

b. Observations and Findings

The team identified the following discrepancies in the FSAR:

1. Table 9.4-6A listed the room temperature for the Component Cooling Pump Room at the beginning of the post-accident period as 119°F, whereas Table 3.11-1 indicated a design temperature of 104°F for the same room.

2. CCW relief valves Q2P17V153, V154, V155, and V158 were not listed on Table 6.2-39 as part of the containment isolation boundary.

3. Table 9.3-1 did not include valve HV 2229, which was also a safety-related, air-operated valve that received a safety injection actuation signal (SIAS).

4. Several differences existed between FSAR Tables 9.2-6 and 9.2.7 and Tables T-1 through T-5 in the CCW FSD. For example, FSAR Table 9.2-6 listed the charging pump lube oil cooler flow as 20 gpm and FSD Table T-2 lists this flow as 30 gpm.

The above discrepancies had not been corrected and the FSAR updated to ensure that the information included in the FSAR contained the latest material as required by 10 CFR 50.71(e). This issue was identified as Unresolved Item 50-348; 50-364/97-201-18. The licensee issued ABN 97-0-1044 and REA 97-1410 to revise the FSAR.

The team also identified the following discrepancies in the CCW FSD A-181000:

1. Table T-1 stated the charging pump lube oil cooler total heat load as 0.036x10E6 instead of 0.096x10E6.

2. Sections 2.1.1.1 and 2.1.1.2 did not contain the current system flow rates and heat loads as documented in calculations 37.4, 37.7, and 37.8.

3. The results of calculation BM-96-1211-002, which evaluated the system for operation with degraded CCW pumps, were not included.

The licensee issued ABN 97-0-1044 to correct the FSD.

c. Conclusions

The team concluded that in several instances the FSAR was not revised in accordance with 10 CFR 50.71(e) requirements. Discrepancies were also identified during the FSD review. The weaknesses mentioned above either have been, or are currently being evaluated by the licensee.

E1.4 Other Related Electrical Systems Review

E1.4.1 AC System

a. Inspection Scope

In this portion of the design review, the team evaluated the emergency diesel generator (EDG) loading calculation, AC system voltage calculations, swing diesel generator operation, and degraded grid relay settings. In particular, the team focused this evaluation on assessing the consistency between the selected systems and the AC system design bases.

b. Observations and Findings

The team reviewed calculation E-42, "Steady-State Diesel Generator Loading Calculation for LOSP, SI, and SBO," Revision 8, to verify that the analysis was consistent with the design basis information of the FSAR and FSD A181004, "Electrical Distribution System," Revision 10, and A-181005, "Diesel Generator System," Revision 10. Calculation E-42 had been maintained current with the plant installed configuration. Various plant modifications which affected calculation E-42 were documented within the calculation and within Table 8.3-1

of the FSAR. Additionally, calculation E-42 (sheet 4) identified diesel generator operating restrictions when lightly loaded (less than 30% load) and that certain motor control center (MCC 1F and 1K) alignments were required. The licensee demonstrated that this information was incorporated in plant procedures, FNP-1-EEP-1, FNP-1-ESP-0.1, FNP-1-SOP-36.3, and FNP-2-SOP-36.3.

AC system calculations SE-94-0470-001, "Farley Unit 1 As-built Load Study Update," Revision 1; SE-94-0470-004 ,"As-Built Load Study Summary Calculation," Revision 1; SE-94-0470-007, "Farley Unit 2 As-built Load Study Update," Revision 0; and SE-94-0470-005, "As-built Load Study Summary Calculation," Revision 1, were reviewed by the team. Methodology, assumptions, and inputs were consistent with design criteria, as-built data, and design bases information. Adequate AC voltage was available during normal, maintenance/refueling, and LOCA conditions for those electrical loads reviewed.

The diesel generator system consisted of five diesel generators (DGs) supporting both Units 1 and 2. Normal assignment of the diesels was: DG 2A (4075 kW) and DG 1B (4075 kW) to Unit 1; DG 1C (2850 kW) and DG 2B (4075 kW)to Unit 2; and SBO DG 2C (2850 kW) served both Units. DG 2A and DG 1C would be automatically aligned to either Unit 1 or 2 depending on the accident condition. The team reviewed the diesel generator logic and verified that proper diesel generator alignment would be achieved during normal and accident conditions. It was also verified that this operation was consistent with the design bases of the FSAR, the TS, and FSD A-181005.

The team reviewed the design bases of the degraded grid protection, as stated in FSAR Section 8.1.1. The team verified that the design bases were conveyed into the engineering design documents. During the plant walkdown, the team verified that the relays were set as stated on the relay setting sheets.

c. Conclusions

The team concluded that adequate AC supply was available for both normal and accident conditions. The electrical loading of the individual components had been considered in the diesel generator calculations. The AC system design documents were consistent with the design bases.

E1.4.2 <u>125 Vdc Battery Systems</u>

a. Inspection Scope

In this portion of the design review, the team evaluated the safety-related Class 1E 125 Vdc Auxiliary Building battery system. In particular the purpose of this evaluation was to verify that the engineering design and installation were consistent with the relevant design bases and industry standards.

b. Observations and Findings

The team determined that the 125 Vdc Auxiliary Building battery system described within FSD A-181004, "Electrical Distribution System," Revision 10, and electrical calculations E-95, "Battery Capacity Calculation for LOSP and

LOSP+LOCA Situations and Limiting Battery Load Profile," Revision 7; E-144, "Determination of Battery Capacity Margins for Adequacy of Voltage at Safety-Related Components for Various Load Profiles," Revision 4; and E-116, "Minimum Available DC Voltage and Permissible Control Circuit Lengths for Limiting Battery Load Profile," Revision 4; was consistent with the FSAR. The assumptions and methodology used in calculations E-95, 116, and 144 were in agreement with IEEE Standard 485-1983,"IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations."

The team noted that FSAR Section 8.3.2 had been extensively revised as a result of the previous NRC inspection findings questioning the adequacy of Auxiliary Building battery duty cycle and voltage requirements. As a result, the FSAR clarified the design-basis requirement for Auxiliary Building battery as a duty cycle of 1 minute with a load profile of 500 amperes for LOSP or LOSP+LOCA and a 2-hour duty cycle with a load profile of 350 amperes for LOSP or LOSP+LOCA assuming a battery charger failure. The analytical basis for the change was contained in calculation E- 144, "Determination of Battery Capacity Margins for Adequacy of Voltage at Safety-Related Components for Various Load Profiles." The team noted that the 1 minute duty cycle requirement was not identified in TS Section 4.8.2.3.2.c.5.

The team reviewed technical specifications (TS) Section 4.8.2.3.2.c.5 and procedure FNP-1(2)-STP-905.1, "Auxiliary Building Battery Service Test," Revisions 5-10, and noted that the TS criterion of 1.75 Vdc per cell (terminal voltage 105 Vdc) for Auxiliary Building batteries 1A, 1B, 2A, 2B was not in agreement with calculation E-144, Revision 4, which required battery terminal voltage greater than 105 Vdc for the battery load profile specified in calculation E-144 and FSAR Section 8.3.2.1.1.1.2. Specifically, the new voltage requirements in accordance with the above calculation were 108.31, 112.43, 113.34, 112.25 for first minute and 108.16,110.75,107.80 and 109.35 for the end of 2-hour for batteries 1A, 1B, 2A and 2B, respectively. The individual cell voltage will be calculated by dividing the battery terminal voltage by number of cells (60 cells).

Production Change Notice (PCN) B-92-0-8099, Revision 0-2, incorporated the battery duty cycle and load profile changes in FSAR Section 8.3.2 and calculation E-144, Revisions 0, 1 and 2, determined the required battery terminal voltages. The required battery terminal voltages at the end of one minute in accordance with the above calculations were: Battery 1A, 112 Vdc; 1B, 114 Vdc; and 2A and 2B, 112.8 Vdc. The required terminal voltage at the end of two hours was 110 Vdc for all four batteries. 10 CFR 50.59 evaluations performed for PCN B-92-0-8099 and the changes to FNP-1(2)-STP-905.1 stated that TS Section 4.8.2.3.2.c.5 was not affected.

Subsequent revisions of the calculation (Revisions 3 and 4) also did not identify the need to change the TS. The current revision of the surveillance procedure (FNP-1(2)-STP-905.1) showed the acceptance values (design requirement from calculation E144) as "Engineering Acceptance Criteria" and not as a TS requirement. The team determined that the surveillance requirement specified in TS was less conservative (allowed battery cell voltages to decrease to 1.75 Vdc which is equivalent to 105 Vdc battery terminal voltage) and would not have met the design requirement for supplying

adequate voltage for all safety-related components. FSAR Section 8.3.2 states that batteries are designed to provide adequate voltages required for safety-related components during normal and accident conditions. FSAR Section 8.3.2 also states that a service test of each battery be performed on the load profiles listed in 8.3.2.1.1.1.2 during each refueling outage or at intervals of 18 months.

10 CFR 50.59 requires prior Commission approval for any proposed change, test, or experiment that involves a change in the technical specifications incorporated in the license. 10 CFR 50.59 also requires that the licensee shall submit an application for license amendment pursuant to 10 CFR 50.90 for a change in technical specifications. The team noted that safety evaluations for Auxiliary Building battery service test procedure FNP-1(2)-STP.905.1 and PCN B-92-0-8099 including calculation 07597-E144 for FSAR revision failed to identify the required change for the technical specifications (TS) Section 4.8.2.3.2.c.5. The licensee also did not submit the application for license amendment pursuant to 10 CFR 50.90. This issue was identified as Unresolved Item 50-348; 50-364/97-201-19. The licensee issued ABN 97-0-1043 to revise the TS and the procedures.

The team reviewed the latest battery capacity testing and service testing results available during the inspection for Auxiliary Building batteries 1A, 1B, 2A, 2B and determined that the batteries met the requirements of IEEE 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Storage Batteries for Generating Stations and Substations," and calculation E-144 for battery terminal voltages.

Operator training manual OPS-40204E/52103C, "DC Distribution," dated May 1996, was reviewed by the team to verify conformance to the design bases. Two discrepancies were identified. The Auxiliary Building battery room high-temperature alarm setpoints on page T-2d for Temperature Switches 3884A/B were higher than the allowable design criteria of 110°F, and page 10 indicated that the Auxiliary Building batteries were sized with an end voltage of 105 Vdc (1.75 Vdc/cell), which was not in conformance with engineering calculation E-144. The licensee stated that pen-and-ink changes had been noted in the training manual text used by the instructors as an interim change and issued ABN 97-0-1043 to revise the training manual.

c. Conclusions

The team concluded that the design/licensing bases for the Auxiliary Building batteries had been maintained except for voltage requirements for TS battery service test. Calculations were adequate and their assumptions and methodology were consistent with industry standards.

E1.4.3 Modifications

a. Inspection Scope

In this portion of the design review, the team evaluated two electrical modifications to verify that the design bases and design configuration were being maintained and that the associated 10 CFR 50.59 safety evaluations were adequate.

b. Observations and Findings

The team reviewed PCN-B-92-2-8068, which provided the design for the replacement of the Unit 2 auxiliary batteries 2A and 2B. The design change modification replaced the existing "GNB" batteries with "C&D" type LCU-27 batteries. Engineering evaluations were performed to substantiate that no changes to the FSAR were required. The following calculations were reviewed for consistency with the modification: E-95, "Battery Capacity Calculation for LOSP and LOSP+LOCA Situations and Limiting Battery Load Profile," Revision 7; E-116, "Minimum Available DC Voltage and Permissible Control Circuit Lengths for Limiting Battery Load Profile," Revision 4; E-144, "Determination of Battery Capacity Margins for Adequacy of Voltage at Safety-Related Components for Various Load Profiles," Revision 4; and E-42, "Steady State Diesel Generator Loading Calculation for LOSP, SI and SBO," Revision 8.

All calculations except for E-42 were adequately updated for the battery replacement. The discrepancy in calculation E-42 is discussed in Section E1.5 of this report. The team confirmed that combustible loading and seismic analyses were reviewed by the licensee for any impacts. The review indicated that both the overall battery weight and combustible material was less than the original installation.

The team also reviewed DCP 95-0-8853-0-001. This modification provided the design for replacement of the Westinghouse Type SV-1 undervoltage relays in the emergency DG relay panels. The modification identified a relay replacement which had operating characteristics (operating speed, deadband, contact capacity, burden) similar to the SV-1 relay. The licensee ensured that the seismic capability of the new relay was adequate. No concerns were identified.

No concerns were identified with the 10 CFR 50.59 evaluations.

c. Conclusions

The electrical modifications reviewed by the team were designed in accordance with existing design bases, necessary documents were identified for update, and installations completed.

E1.4.4 System Walkdown

a. Inspection Scope

The team performed a walkdown of the Auxiliary Building batteries (1A, 1B, 2A, 2B); the Unit 2 CCW pump area; the Unit 1 AFW area; and the Class 1E battery charger and UPS area, emergency switchgear rooms, and DG rooms for Units 1 and 2. In particular, the intent of the walkdown was to verify that the electrical installations conformed to the relevant engineering design criteria and the applicable drawings.

b. Observations and Findings

All auxiliary building batteries were installed as shown on the drawings. Auxiliary batteries 2A and 2B reflected the design configuration of PCN-B-92-2-8068. Battery room temperatures were within the design range (60°F-110°F), the areas were clean of debris, the battery racks were clean of battery acid, and the individual cells were clean of dust and battery acid. The Unit 1A battery exhibited slight corrosion on various intercell connectors. The licensee stated that maintenance would correct this condition.

The team verified that the electrical installations were in accordance with the design criteria of A-172389, "Cable Working Specifications," Revision 91; A-177541, "Tray and Conduit Details and Notes," Revision 8; and A-177538, "Electrical General Details & Notes," Revision 24. Electrical separation requirements were reviewed to ensure the installations complied with FSAR Section 3A (which discusses conformance with RG 1.75) and various installations involving non-1E loads powered from class 1E sources were observed within the DG building. All installations complied with the engineering design criteria.

The team verified that 4.16 kV breaker cubicles (DF-03, 04, 05 and DG-04, 05, 10) and MCC cubicles FA-H4 (MCC 1A) and FA-14 (MCC 2A) were installed and configured per the appropriate engineering documents. The team noted that the calibration due date on the 50G relay on cubicle 1-DF05 was past due. A review of the Preventive Maintenance Task Planning System indicated that the 50G relay actually required calibration by November 25, 1997. The test frequency of the relay had been changed since its last calibration, in accordance with recommendations from the plant's Reliability Centered Maintenance (RCM) Program.

The licensee had also taken precautionary measures to remove any 4.16 kV breakers, which were not in the connected position, from the switchgear while evaluating the seismic adequacy of unconnected breakers in the switchgear.

The team identified housekeeping issues such as debris in the cable trays, unsecured ladders, fans and carts and cables outside the raceways. This was brought to the licensee's attention. The licensee took prompt measures to correct them.

Various fire barrier penetration seals were observed. Penetration 45-121-26 was located in the wall between the Unit 1 hot shutdown panel area (Fire Area 12) and an adjacent cable chase (Fire Area 13) and was sealed with silicone foam. The penetration contained electrical raceways/cables and copper tubing. The copper tubing was installed under Change Notice No. BM-3095 (dated August 11, 1980). The design change provided additional cooling to Room #254 so as to maintain a temperature of less than 80°F. The copper tubing and electrical raceway/cables were installed through the penetration in support of the room cooling modification.

A review of the design change did not reveal any engineering evaluation of fire barrier 45-121-26 for the modification. The team questioned the qualification of penetration 45-121-26 with copper tubing. In response, the licensee provided Factory Mutual Test Report 27390 as the test results for silicone foam qualification for fire barrier penetrations. The actual configuration of copper tubing was not tested in the Factory Mutual Report.

Change Notice No. BM-3095 did not indicate if penetration 45-121-26 was previously reviewed for the installation of copper tubing. The licensee's preliminary evaluation showed that there were no concerns with the qualification of this penetration seal. The licensee stated that they used engineering judgement during the initial installation to ensure that the above test enveloped the configuration of the penetration. The licensee also stated that a calculation would be developed to ensure that the copper piping installation would not degrade during a fire and breach the fire barrier. FSAR Section 9B.2.2.5.3 stated that for the fire barrier penetrations that were not in the as-designed condition, an evaluation was required to establish its qualification. At end of the inspection, an evaluation was not available for team's review. The licensee issued REA 97-1407 to resolve this item. Review of testing documentation or analysis to show that as-built configuration matched the tested configuration for fire penetration seal was identified as Unresolved Item 50-348/201-20.

c. Conclusions

In general, the installation of cables, raceways, and equipment was in conformance with design documents. However, in one instance the licensee did not have adequate documentation to show that as-built configuration matched the tested configuration for fire penetration seal. The material condition was generally adequate, but the team identified the need for improvement in housekeeping area.

E1.4.5 <u>FSAR and FSD Review</u>

a. Inspection Scope

In this portion of the design review, the team evaluated the FSAR sections and FSDs related to the electrical systems.

b. Observations and Findings

The team identified the following discrepancies in the FSAR:

1. Section 8.3.1.1.3.A.2 stated that the unit auxiliary transformer "B" megavolt-ampere (MVA) rating at 65°F was 47.99 instead of 46.7 as shown on drawing D-202700.

2. Section 8.3.1.1.9B referred to Section 8.3.1.1.3 for interrupting capacities for distribution panels. However, Section 8.3.1.1.3 did not include interrupting capacity data for distribution panels.

3. Section 8.3.1.2 stated that there were twenty-one 600-V/208-V motor control centers; however, the actual number of motor control centers identified is nineteen.

The licensee issued ABN 97-0-1043, REA 97-1410, and ABN 97-0-1054 to correct these discrepancies. The above discrepancies had not been corrected and the FSAR updated to ensure that the information included in the FSAR contained the latest material as required by 10 CFR 50.71(e). This issue was identified as Unresolved Item 50-348; 50-364/97-201-21.

The team identified the following discrepancies in the DG System FSD, A-181005 Revision 10:

1. Section A.4.1.3 stated that diesel generator 1C was only lightly loaded (6% of its continuous rating) in all four scenarios, but it did not agree with calculation E-42, Revision 8. The calculation stated that the load was only 39 kW (less than 1.5%).

2. Section 5.10.8.7 stated that the DG lube oil heaters were powered from 120/208V MCC distribution panels instead of the DG 600-120/208V auxiliary transformers.

3. Open items DG-FSD-006, 018, 019, and 024 resulted from the licensee's review of the FSD were not incorporated in the FSD. These open items dealt with system operating information concerning temperature and pressure for the diesel generator intercooler and lubricating oil system components.

The licensee issued ABN 97-0-1053 to revise the FSD.

c. Conclusions

The team concluded that in several instances the FSAR was not revised in accordance with 10 CFR 50.71(e) requirements. Discrepancies were also identified during the FSD review. The weaknesses mentioned above either have been, or are currently being evaluated by the licensee.

E1.5 Control of Calculations

a. Inspection Scope

In this portion of the design review, the team evaluated numerous engineering calculations related to the AFW and CCW systems, as discussed in other sections of this report. In particular, the team reviewed the design control aspects associated with these calculations.

b. Observations and Findings

The team determined that in several cases the calculations that had previously been superseded were not identified as such on the calculation index, and design-basis calculations were not appropriately revised to show the existing design condition. In addition, the licensee did not always revise affected calculations when new calculations were performed. The following examples represent these weaknesses:

- Calculation 38.06, "Determine Flow rate Through Pipe Break in CCW System," Revision 0, concerning the CCW surge tank low-low level setpoint, superseded calculation 35.5, "Evaluation of CCW Surge Tank Level Setpoints," Revision 0, yet calculation 35.5 was shown as active on the index.

- Calculation 34.5, "Component Cooling Water System NPSH (ES No. 90-1820)," Revision 0, was affected by calculation 39.3, "CCW Surge Tank Analytical Limit for Level Setpoint," Revision 0, as discussed in Section E1.3.2.1b of this report, but calculation 34.5 was not revised accordingly.

- Calculation 25.3, "Overpressurization of AFW Piping During Overspeed Testing," Revision 0, determined that an unacceptable piping pressure could develop in the event of turbine overspeed of 125% of rated speed. This calculation was not revised to reflect the revised overspeed setpoint of 115% of rated speed nor was a calculation prepared to supersede calculation 25.3. The design bases for acceptability of the overspeed setpoint were documented in modification PCN B-88-1-5003, "Change Overspeed Trip Setting for TDAFW Pump," Revision 1.

- Calculation E-144, "Determination of Battery Capacity Margins for Adequacy of Voltage at Safety-Related Components for Various Load Profiles," Revision 2, reflected a new battery installation but calculation E-42, "Steady-State DG Loading Calculation for LOSP, SI and SBO," Revision 8, which used calculation E-144 as an input, was not revised accordingly.

- AFW flow calculations 40.02, "Verification of AFW Flow Bases," Revision 3 (Unit 1), and the equivalent calculation for Unit 2 (38.04) provided design-basis information for the AFW system; yet calculation 35.04, "Auxiliary Feedwater System Head Curves (ES 90-1831)," Revision 0, which provided similar information, was not annotated to indicate it did not contain design-basis information.

- Calculation SC-96-1211-002, "CCW Heat Exchanger Maintenance Repairs," Revision 1, stated that it was judged that the increase in weight resulting from the CCW heat exchanger modifications would not affect the acceptability of the foundation anchorages. However, the calculation for the anchorages was not revised nor was the increase in heat exchanger weight reflected in the appropriate calculation.

The licensee stated that the philosophy used for calculations before the implementation of procedure NEP 4-4, "Preparing and Reviewing Calculations," in 1996 was that calculations were performed to evaluate a potential design parameter on an "as-designed" basis. The licensee also stated that subsequent design decisions that were different but substantiated by a particular calculation may not have been updated into the calculation but should be found documented in letters or other documentation.

The team did not identify any systematic method of correlating these letters or other documentation with the affected calculation. Therefore, the team was concerned that existing design-basis calculations could be superseded or not consistent with the as-built plant and these calculations used inappropriately in subsequent analyses and design decisions. The team did not identify any instances where the use of calculations that were not consistent with the as-built plant resulted in an adverse effect on safety. Adherence to current calculation procedures and practices would have prevented most of the above deficiencies. The team noted that the new calculations were controlled properly, but the licensee did not take any action to correct deficiencies with existing calculations.

The licensee issued REA 97-1407 to appropriately correct the above deficiencies in calculation and design control. 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 and Farley Support Procedure GO-M-1, "Designer Interface Document," to ensure that plant design-basis documentation is maintained current. This issue was identified as Unresolved Item 50-348; 50-364/97-201-22.

c. Conclusions

The team concluded that control of design-basis calculations was weak before the licensee implemented Procedure NEP 4-4.

V. Management Meeting

XI Exit Meeting Summary

On March 14, 1997, the team members conducted a pre-exit meeting with the licensee. On March 26, 1997, the team leader conducted a final public exit meeting, during which the team's overall conclusions and inspection findings were presented. The licensee did not identify as proprietary any information provided to, or reviewed by, the team. Upon conclusion of the exit meeting, the NRC team leader and others answered questions from local media representatives.

APPENDIX A

Open Items

This report categorizes the inspection findings as unresolved items and inspection follow-up items in accordance with the 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 II office will issue any enforcement action resulting from their review of the identified unresolved items. An inspection follow-up 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.

Item Number	Finding Type	Title
50-348;50-364/97-201-01	URI	Unprotected CST Connections (Section E1.2.2.1b2)
50-348;50-364/97-201-02	URI	Tornado Protection of CST Level Instrumentation (Section E1.2.2.1b3)
50-348;50-364/97-201-03	URI	AFW Check Valve Reverse Flow Testing (Section E1.2.2.4b)
50-348;50-364/97-201-04	URI	AFW Check Valve forward Flow Testing (Section E1.2.2.4b)
50-348;50-364/97-201-05	URI	TDAFW Battery Testing (Section E1.2.3.1b)
50-364/97-201-06	URI	TDAFW Battery Installation (Section E1.2.3.1b)
50-348/97-201-07	IFI	CST Level Alarm (Section E1.2.4.1b)
50-348;50-364/97-201-08	URI	Tornado Protection of TDAFW Pump Vent Stack (Section E1.2.6b1)
50-348;50-364/97-201-09	URI	Tornado Missile Spectra (Section E1.2.6b2)
50-348/97-201-10	IFI	CST Level Transmitter Freeze Protection (Section E1.2.6b4)
50-348;50-364/97-201-11	URI	AFW FSAR Discrepancies (Section E1.2.7b)
50-348;50-364/97-201-12	URI	Stress Analysis Temperature (Section E1.3.2.1b5)

50-348;50-364/97-201-13	URI	MOV Design-Basis Differential Pressure (Section E1.3.2.3b)
50-348;50-364/97-201-14	IFI	CCW Pump Testing (Section E1.3.2.4b)
50-348;50-364/97-201-15	URI	Post Modification Testing (Section E1.3.2.5b)
50-348;50-364/97-201-16	IFI	Calculation Discrepancies (Section E1.3.2.5b)
50-348;50-364/97-201-17	URI	Drawing and Procedure Discrepancies (Section E1.3.2.6b)
50-348;50-364/97-201-18	URI	CCW FSAR Discrepancies (Section E1.3.7b)
50-348;50-364/97-201-19	URI	TS change for Auxiliary Building Battery (Section E1.4.2b)
50-348/97-201-20	URI	Fire Barrier Penetration Seal Documentation (Section E1.4.4b)
50-348;50-364/97-201-21	URI	Electrical FSAR Discrepancies (Section E1.4.5)
50-348;50-364/97-201-22	URI	Control of Calculations (Section E1.5b)

APPENDIX B

EXIT MEETING ATTENDEES

Southern Nuclear Operating Company, Inc.

W.G. Hairston, President & Chief Executive Officer
J.D. Woodard, Executive Vice President
D.N. Morey, Vice President
J. Carlington, General Manager, Nuclear Support
R. Hill, General Manager, Farley Nuclear Plant
B.D. McKinney, Nuclear Engineering & Licensing Manager
D. H. Jones, Engineering Manager
J.E. Odom, Operations Superintendent
D.E. Grissette, Operations Manager
J.J. Thomas, Engineering Support Manager
G.S. Waymire, Administration Manager
E.F. Bates, Nuclear Engineering & Licensing
W.H. Warren, Engineering Support Supervisor
G.P. Crone, Training Supervisor
J.W. McGowan, Safety Audit and Engineering Review Group Supervisor
H. Mahan, Senior Engineer
B. Badham, Safety Audit and Engineering Review Group
C.D. Nesbitt, Administration

U.S. Nuclear Regulatory Commission

R. Mathew, Team Leader, Special Inspection Branch, NRR
D. Collins, Deputy Director, Division of Reactor Safety, RII
T. M. Ross, Senior Resident Inspector, Farley
J. Bartley, Resident Inspector, Farley
K. Clark, Public Affairs Office, RII

Public Members

D. Pearson, WOOF Radio
J. Davis, WTVY- TV CH 4
D. Corb, WTVY- TV CH 4

APPENDIX C

LIST OF ACRONYMS USED

ABN	As Built Notice
ABV	Auxiliary Building Ventilation
AC	Alternating Current
AFW	Auxiliary Feedwater
AMSAC	ATWS (Anticipated Transient Without Scram) Mitigating System Actuation Circuitry
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BOP	Balance of Plant
B&PV	Boiler and Pressure Vessel
CCW	Component Cooling Water
CDF	Core Damage Frequency
CST	Condensate Storage Tank
DC	Direct Current
DCP	Design Change Package
DG	Diesel Generator
DR	Deficiency Report
EOP	Emergency Operating Procedure
ERP	Emergency Response Procedure
FNP	Farley Nuclear Plant
FSAR	Final Safety Analysis Report
FSD	Functional System Description
gpm	Gallons Per Minute
I&C	Instrumentation and Control
IA	Instrument Air
IEEE	Institute of Electrical and Electronic Engineers
IP	Inspection Plan
ISA	Instrument Society of America
IST	Inservice Testing
kV	Kilovolt
kW	Kilowatt
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
MCC	Motor Control Center
MDAFW	Motor-Driven Auxiliary Feedwater
MOV	Motor-Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MVA	Megavolt-Ampere
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation, Office of (NRC)
NSSS	Nuclear Steam Supply System
NUREG	NRC Technical Report Designation
P&ID	Piping and Instrumentation Diagram
PCN	Production Change Notice

PCR	Production Change Request
PDE	Procurement Deviation Evaluation
PORC	Plant Operations Review Committee
PRA	Probabilistic Risk Assessment
RCM	Reliability Centered Maintenance
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REA	Request for Engineering Assistance
RG	Regulatory Guide
SBO	Station Blackout
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SRP	Standard Review Plan
SRSS	Square Root of the Sum of the Squares
SSC	Structure, System, and Component
SSSA	Self-Initiated Safety System Assessment
STP	Surveillance Test Procedure
SW	Service Water
TCN	Temporary Change Notice
TDAFW	Turbine-Driven Auxiliary Feedwater
TS	Technical Specifications
UPS	Uninterruptable Power System
URL	Upper Range Limit
USQ	Unreviewed Safety Question
V	Volts
Vac	Volts Alternating Current
Vdc	Volts Direct Current
WR	Work Request

APPENDIX D

LIST OF DOCUMENTS REVIEWED

FSDs

Component Cooling Water System Functional System Description, A-181000, Rev. 6
Auxiliary Feedwater System Functional System Description, A-181010, Rev. 1
Functional System Description Diesel Generator System, A-181005, Rev. 10
Functional System Description Electrical Distribution System, A-181004, Rev. 10

SSSAs

Component Cooling Water System, Self-Initiated Safety System Self-Assessment, November 22, 1989

Auxiliary Feedwater System, Self-Initiated Safety System Self-Assessment, March 3, 1995

FSAR/PSAR (Partial Review)

PSAR Section 5.1.2.5, Tornado Loads, Amendment 6
FSAR Section 3A, Regulatory Guide 1.75 Section, Rev. 9
FSAR Section 6.5, Auxiliary Feedwater System
FSAR Section 3.3, Wind and Tornado Loadings
FSAR Section 3.5, Missile Protection
FSAR Section 3.1, Conformance with NRC General Design Criteria
FSAR Section 3K.4.1.4.7, Flooding
FSAR Section 6.2.4, Containment Isolation System
FSAR Chapter 15, Accident Analyses
FSAR 7.0 Instrumentation and Control
FSAR 8.0 Electric Power
FSAR Section 9.2.2, Cooling System for Reactor Auxiliaries
FSAR Section 9.1.3, Spent Fuel Pool Cooling and Cleanup System
FSAR Section 9.2.5, Ultimate Heat Sink
FSAR Section 9.3.1, Compressed Air System
FSAR Section 9.2.1, Station Cooling Water System
FSAR Section 9.4.2, Auxiliary Building
FSAR Section 3.1, Conformance with NRC General Design Criteria
FSAR Section 6.2.4, Containment Isolation System
FSAR Section 9B, Appendix 9B Fire Protection Program, Rev. 12
FSAR Section 3.2, Classification of SSCs
FSAR Section 3.K.4.1.2.7, Flooding
FSAR Section 9.2.6, Condensate Storage Facilities

Relay Settings

A-177048 Sht. 254, Relay Settings, Rev. 1

A-177048 Sht. 38, Relay Settings, Rev. 1

A-177048 Sht. 209, Relay Settings, Rev. 1

A-177048 Sht. 44, Relay Settings, Rev. 0

A-177048 Sht. 346, Relay Settings, Rev. 3

A-177048 Sht. 261, Relay Settings, Rev. 2

A-177048 Sht. 275, Relay Settings, Rev. 2

A-207048 Sht. 44, Relay Settings, Rev. 0

A-207048 Sht. 270, Relay Settings, Rev. 0

A-207048 Sht. 256, Relay Settings, Rev. 0

A-207048 Sht. 269, Relay Settings, Rev. 0

A-207048 Sht. 255, Relay Settings, Rev. 0

A-207048 Sht. 254, Relay Settings, Rev. 2

A-207048 Sht. 38, Relay Settings, Rev. 0

A-177048 Sht. 505, Relay Settings, Rev. 2

A-177048 Sht. 505A, Relay Settings, Rev. 1

A-177048 Sht. 506, Relay Settings, Rev. 2

A-177048 Sht. 506A, Relay Settings, Rev. 1

A-207048 Sht. 505, Relay Settings, Rev. 2

A-207048 Sht. 505A, Relay Settings, Rev. 2

A-207048 Sht. 506, Relay Settings, Rev. 2

A-207048 Sht. 506A, Relay Settings, Rev. 2

Calculations

BM-95-0776-001, CCW System Evaluation Using Degraded CCW Pump Curve, Rev. 0

BM-95-0961-001, Verification of CST Sizing Basis, Rev. 1

BM-96-1171-001, Heat-Up of Turbine Drive AFW Pump Room with Loss of Ventilation, Rev. 0

CST Manufacturer's Design Calculation, Charge Number 72-4859

REES-F-91-006, Estimate of Core Damage Frequency from a Tornado Missile Striking Exposed CST Piping, Rev. 0

REES-F-94-014, Probability of Tornado Striking SW DG Return Lines in Excavation Trench, Rev. 0

SC-96-1211-002, CCW Heat Exchanger Maintenance Repairs, Rev. 1

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