U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

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EXECUTIVE SUMMARY

During the period from November 18, 1996, through January 10, 1997, the U.S. Nuclear Regulatory Commission's (NRCs) Office of Nuclear Reactor Regulation (NRR) performed a design inspection of the St. Lucie Unit 1 Auxiliary Feedwater System (AFW) and the Unit 2 Component Cooling Water (CCW) System. The inspection team was led by an inspection team leader from the Special Inspections Branch within NRR and was comprised of five contractors from Sargent & Lundy Corporation. The purpose of the inspection was to evaluate the capability of the systems to perform safety functions required by their design basis, adherence to the design and licensing basis, and consistency of the as-built configuration with the Updated Final Saftey Analysis Report(UFSAR). The systems were selected for review based upon probablistic risk, previous inspection insights, and modification history.

With regard to the Unit 1 AFW system, the team identified that the operational performance capability was acceptable and the system as installed and operated met both the original design basis and subsequent licensing commitments. In the mechanical review area, the team determined the size of the condensate storage tank, the relief capacity of the atmospheric dump valves and the flow capability of the AFW pumps to be acceptable. Also acceptable was the available net positive suction head for the AFW pumps.

The results of the team's electrical review indicated that sufficient voltage and current were available to power the equipment contained within the AFW system. Adequate circuit protection for the electrical equipment was also confirmed. The AFW pump motors were sized sufficiently.

The team's review of instrumentation and controls identified that the AFW Actuation System setpoints were sufficient to ensure automatic actuation of the AFW system when required. Also, the condensate storage tank level indication in the control room was adequate.

Walkdowns of the system revealed generally good overall material condition, with some degradation of portions of the governor assembly and inlet supply steam motor operated valve for the AFW turbine driven pump.

Notwithstanding the above positive findings, the team identified several issues relative to the system's design or the licensee's implementation of the design. Also, several issues were identified by the licensee during their preparation for the inspection. The following were among the issues identified by either the team or the licensee:

> A concern was raised over the acceptability of the technical specification limit for the condensate storage tank level. Although the licensee had in place administrative controls to ensure that an adequate volume of condensate would be maintained to meet all design basis requirements, the current technical specification limit of 116,000 gallons may not be adequate.

FP&L has not established environmental qualification for the AFW Terry Turbine Woodward Governor Control. This equipment is in an area that is subject to steam impingement and elevated temperatures for a break in the main steam lines. Preliminary information indicated that the equipment may be qualifiable.

FP&L has not included the cross-tie isolation valves which connect the Unit 1 AFW pumps to the Unit 2 condensate storage tank in their ASME Section XI Inservice Test Program. The valves have however been stroked and lubricated on an annual basis. Full flow testing from the Unit 2 condensate storage tank has not been performed.

As part of their generic FSAR review which was ongoing during the inspection, FP&L identified that operational procedures had not been written, nor had testing been performed, to confirm the operability of the circuit breakers used to transfer DC control power from a faulted electrical bus to an energized bus for the turbine driven AFW pump controls. Testing performed during the inspection showed the breaker (breakers) were not operational.

The documentation related to the troubleshooting of the above circuit breakers was not adequate. A review of the completed documentation revealed that changes in the troubleshooting plan were not sufficiently detailed in the work order used to conduct the troubleshooting. Also, the team identified several deficiencies in the specific maintenance test procedures used to perform overcurrent testing of molded case circuit breakers.

FP&L has not established a program to detect and address unidirectional drift for certain AFW instruments.

FP&L has not performed an analysis to demonstrate the acceptability of the overall loop accuracies for certain instrumentation used solely for indication.

The turbine driven AFW pump failed a surviellance due to the inability of the discharge motor operated valve to close. Upon review, the team learned of three other similar failures within the last 16 months. Two of those failures had been attributed to dirty torque switch contacts. Licensee efforts to determine a definitive root cause of the failures have not been successful.

With regard to the Unit 2 CCW system, the team identified that the operational performance capability was acceptable, and the system as installed and operated met both the original design basis and subsequent licensing commitments. In the mechanical review area, the team determined that the CCW system is capable of providing sufficient cooling capacity to cool reactor coolant auxiliary systems components during normal operation, normal plant shutdown, emergency shutdown, and during postulated design basis accidents. The available net positive suction head to the CCW pumps was determined to be acceptable, as was the overall system flow balancing.

In the electrical area, the team determined that the batteries were adequately sized, and that acceptable voltage and current are available to power the system loads under all design basis conditions. The CCW pump motor, fuse, and cable sizing were also reviewed and determined to be acceptable.

In the area of instrumentation and controls, the CCW surge tank level and heat exchanger setpoints were determined to be acceptable.

Walkdowns conducted of the CCW system revealed generally good overall material condition.

Notwithstanding the above positive findings, the team did identify a few issues that questioned aspects of the system's design or the licensee's implementation of the design. Also, some issues were identified by the licensee during their preparation for the inspection. The following were among the issues identified by either the team or the licensee:

In preparation for the inspection, the licensee determined that the operating curves used to evaluate the maximum allowable sea water temperature for various degrees of fouling of the CCW and intake cooling water heat exchangers were not adequate. The curves were based on a non-conservative assumption of fouled shutdown cooling and containment fan cooler heat exchangers. Clean heat exchangers would dissipate more heat and tend to raise the temperature of the CCW system above the 108 °F design limit.

FP&L has not performed formal calculations to support the setpoints for the CCW radiation monitors.

Overall based on the above findings, the team found the design of the two selected systems to be good, with adequate design margins. FP&L's understanding of the design basis was good, as was their inspection preparation and their ability to resolve team identified concerns. The implementation of the design was found to be adequate with some issues noted.

E1 Conduct of Engineering

E1.1 Inspection Objectives, Methodology, and Objectives

The objectives of this design inspection of the Unit 1 Auxiliary Feedwater (AFW) System and the Unit 2 Component Cooling Water (CCW) System were to evaluate the capability of the systems to perform safety functions required by their design basis, adherence to the design and licensing basis, and consistency of the as-built configuration with the Updated Final Saftey Analysis Report(UFSAR). These systems were selected for review based upon a review of the plant's Individual Plant Evaluation (IPE) (probablistic risk assessment), previous NRC inspection insights, and modification history.

The inspection team was led by a team leader from the NRC's Office of Nuclear Reactor Regulation and was comprised of five contractors from the Sargent & Lundy Corporation. The team included two engineers who evaluated the mechanical aspects of the selected systems, one electrical engineer, one instrumentation and control engineer, and one field engineer. The team was on-site for four weeks during the period of November 18, 1996, through January 10, 1997. In conducting the review, the team first assembled the design basis and licensing basis for the selected systems. A review was then conducted of the supporting calculations, analyses, and implementing procedures. Finally, in-plant observations and walkdowns of the plant equipment were performed.

E1.2 Auxiliary Feedwater System - Unit 1

E1.2.1 System Overview

The AFW system at St. Lucie is designed to provide a source of cooling water to the secondary side steam generator to cool the reactor and its coolant system whenever the normal flow of cooling feedwater to the steam generator is lost. The original system design licensed in 1974 was significantly revised/augmented in 1981 to meet post-TMI licensing requirements specified in NUREG-0737 and NUREG-0578.

The AFW system has two motor-driven pumps, A and B, and one steam turbine-driven pump, C. The two motor-driven pumps and turbine-driven pump are sized to remove decay heat from the reactor coolant system. The two motor-driven pumps are powered from separate emergency AC sources and the turbine-driven pump is fed with main steam from either of the two steam generators. The turbine driven pump controls can be powered from either the A or B station batteries.

The turbine-driven pump supplies cooling water to both the A and B steam generators by means of two separate lines, each with its own motor operated DC control valve. Each motor-driven pump normally supplies water to one steam generator. A cross connection with two remote manual normally closed 1E powered isolation valves is provided to enable the routing of feed flow of the two motor driven pumps to either steam generator. The AFW system is initiated

automatically from the control room on low water level in the steam generator, but provisions are also provided for manual operation from the control room or local shutdown panel.

For long term cooling of the primary system, the heat from the steam generator is removed via the atmospheric dump valves (ADV) to the atmosphere which is the ultimate heat sink. Each steam generator has one ADV.

The primary water supply for the AFW system is maintained in a 250,000 gallon condensate storage tank (CST) connected to the AFW pump suction. The A and B motor-driven pumps have a common suction line to the CST, and the C turbine-driven pump has a separate redundant line to the CST. There is a cross-tie from the Unit 2 CST to the suction of Unit 1 AFW feed pumps. Low water level in the CST will alarm and annunciate in the main control room. The low level set point provides adequate time for an operator to initiate make-up without compromising plant safety.

The AFW system is designed for the following safety, non-safety and quality related functions, design criteria and licensing requirements.

E1.2.1.1 System Functions

E1.2.1.1.1 Safety Functions

a. Provide feedwater to remove decay and sensible heat from the Reactor Coolant System (RCS) for the following plant conditions:

   - Loss of offsite power (LOOP) assuming the most limiting single active failure
   - Station blackout
   - Loss of normal feedwater flow, assuming the most limiting single active failure concurrent with or without a LOOP with a high energy line break in the AFW system
   - Moderate or high energy steam/feedwater line break inside or outside containment, assuming a single active failure concurrent with or without a LOOP

b. Be capable to isolate the AFW steam and feedwater supply lines from the affected steam generator following a steamline or feedline break.

c. Be capable to automatically initiate AFW flow upon receipt of an auxiliary feedwater actuation signal (AFAS) within the time frame specified by the most limiting design basis accident analyzed.

E1.2.1.1.2 Non-Safety Functions

The AFW system will provide to the steam generator water inventory during normal plant startup/shutdown operation.

E1.2.1.1.3 Quality Functions (those functions that are not safety related but are important to safety)

a. Withstand design bases earthquake loads without loss of function.

b. Support RCS heat removal to achieve and maintain hot standby, and bring the plant to shutdown cooling system entry conditions during fires that require control room evacuation or during fires not requiring control room evacuation, with or without a concurrent LOOP.

c. Capable of being periodically tested to verify functional readiness and performance.

E1.2.1.2 Conformance With Selected General Design Criteria

a. General Design Criterion 2- Design bases for protection against natural phenomena

The design of the Unit I condensate storage tank is an exception, as no protection is provided against a vertical tornado missile. Plant protection is provided by a cross-tie to the Unit 2 condensate storage tank.

All the AFW components are located above the probable maximum flood level.

b. General Design Criterion 4 - Environmental and dynamic effects design

Except for the condensate storage tank and underground suction piping to the AFW pumps, the AFW system is located in an outdoor area below the main feedwater and main steam lines, and is surrounded by tornado missile resistant shielding. The turbine-driven pump is missile shielded from the motor-driven pumps, and a pipe restraint precludes the turbine-driven pump header from whipping into the motor-driven pump header.

c. General Design Criterion 5 - Sharing of structures, systems and components

The only shared component between the AFW systems for the 2 units is the Unit 2 condensate storage tank (CST). The connection for flow to Unit 1 is done at an elevation that assures adequate condensate to Unit 1 while assuring that sufficient quantity is available for Unit 2 safe shutdown in the case of a loss of the Unit 1 CST due to a vertical tornado missile.

d. General Design Criterion 19 - Control Room

Adequate instrumentation and controls for AFW flow and steam generator level are provided to ensure that the plant can be brought to a hot

standby or hot shutdown during plant transients or accidents from either the control room or from local stations. The Auxiliary Feedwater Actuation System (AFAS) automatically initiates AFW flow to the steam generators. In the event of a steamline or feedwater line rupture, the AFAS automatically isolates the affected steam generator and feeds the intact steam generator.

e. General Design Criterion 44 - Cooling water

The AFW system provides water inventory to the steam generators for removal of decay and sensible heat. The heat is removed via the steam dump bypass system (SDBS) during normal operation, and via the SDBS, the main steam safety valves or the Atmospheric Dump valves (ADVs) during postulated accident conditions. Sufficient redundancy is provided in the AFW system to ensure

- AFW flow to the steam generators with a single active failure during transients or accidents
- isolation of the failed components.

f. General Design Criterion 45 - Inspection of cooling water system

The system was designed to assure periodic In-Service Inspection of the AFW system.

g. General Design Criterion 46 - Testing of cooling water system

The system was designed to assure that the AFW system can be tested.

- by flow transmitters to test the pumps
- pressure indicators to test pressure integrity
- remote-manual means to activate pumps, control valves from control room.

h. General Design Criterion 55 - Reactor coolant pressure boundary penetrating containment

Exception was taken to this criteria for the design of the AFW for containment isolation. Inboard and outboard containment isolation is provided by check valves.

E1.2.1.3 Conformance With Selected Regulatory/Licensing Requirements

a. Regulatory Guide 1.26

The AFW system was classified as ASME Class III or Quality Group C except for portions of the piping and components interfacing with the main feedwater and main steamline which are classified as ASME Class II or Quality Group B.

b. Regulatory Guide 1.62

The AFAS was designed to allow manual initiation from the control room. It was designed so that failure will not result in loss of manual capability to initiate AFW flow from the control room.

c. Regulatory Guide 1.97

The Reg. Guide variables for the AFW system are the AFW flow and the CST water level. The AFW flow is a Category 2 Type D variable, and is designed to meet the requirements of the above Guide. The CST water level is a Category 1 Type D variable and is designed and qualified to meet the requirements of the Guide.

d. NUREG-0578 and NUREG-0737

The following changes were implemented to the AFW system to address the requirements in the above guides:

- The AFAS was installed for automatic initiation of the AFW flow.
- Flow indication was installed in each of the AFW pump discharge headers to determine the flow to each steam generator.
- AFW system capability to achieve hot standby conditions following a loss of normal feedwater was analyzed assuming a high energy line break in the AFW system concurrent with the most limiting single active failure.

E1.2.2 Mechanical Design Review

The mechanical design review consisted of an assessment of plant design transients to establish design requirements and an assessment of thermal/hydraulic and fluid mechanics calculations to determine if the AFW system is designed to remove the required heat load. In addition, the inspection team reviewed the plant design drawings, modification packages, UFSAR, Technical Specifications, operating procedures, IE Bulletins, Notices, Generic Letters and engineering evaluations associated with the system.

E1.2.2.1 Condensate Storage Tank/Atmospheric Dump Valves

E1.2.2.1.1 Scope of Review

Evaluate the sizing of the condensate storage tank (CST) and atmospheric dump valves (ADVs) to determine the AFW system capability to remove decay heat and bring the plant to a safe shutdown.

E1.2.2.1.2 Inspection Findings

Original Design Basis

The AFW system for St. Lucie was designed to provide secondary side cooling to the steam generator to cool the primary reactor coolant system whenever there is a loss of normal feedwater flow to the steam generator. Loss of normal feedwater flow to the steam generator may occur for the following plant conditions:

1. Loss-of-offsite power (LOOP)
2. Loss of normal feedwater
3. Station Blackout
4. Feedwater line break
5. Main steam line break

For the above plant conditions the AFW system will be required to provide water inventory and heat removal capability for secondary side cooling. The water inventory to the steam generator is provided by the AFW pumps from the CST. The heat is removed from the steam generator to the condenser via the steam dump bypass system (SDBS), or to the atmosphere via the main steam safety relief valves and the ADVs.

The AFW system was originally designed for manual operation with the capability to remove sufficient decay heat from the primary system after any design basis accident to be in hot standby at 532°F for 8 hours. The system was also designed to be capable of achieving a hot shutdown condition of 300°F in 3 1/2 hours after a loss-of-offsite-power event. The system's capability to achieve hot standby after a design basis accident was evaluated assuming a concurrent loss of off-site power (LOOP) and the most limiting single active failure. The maximum designed flow required from the system to achieve hot standby was 500 gpm at a steam generator pressure of 1000 psia.

The volume of cooling water or condensate required to achieve hot standby and hot shutdown conditions was determined from Combustion Engineering (CE) calculation, F-PEC-76, "Auxiliary Feedwater Volume Required" dated 10/20/72. This calculation made the following assumptions which were considered conservative for the initial design:

(a) Cooling water temperature of 120°F
(b) Turbine-driven auxiliary feedwater pump operates providing a flow of 500 gpm to both steam generators
(c) Loss of normal feedwater coincident with a loss-of-offsite-power, a reactor trip, turbine trip and the stopping of the reactor coolant pumps
(d) Steam generator at the low water level at the time of loss of normal feedwater
(e) Reactor is at 105% of reactor power of 2570MWt or 2698MWt
(f) The time delay to initiate AFW flow to steam generator is 8 minutes or 480 seconds.

The volume of condensate required was determined to be 110,000 gallons to achieve and remain at hot standby at 532°F for 8 hours. Recovery of steam generator water level was not a design requirement. The volume of condensate required to achieve hot shutdown at 300°F in 3 1/2 hours, for a LOOP, was about 100,000 gallons.

The current Technical Specifications (TS) limit for maintaining 116,000 gallons of condensate in the condensate storage tank was based on the amount required to achieve and remain at hot standby at 532°F for 8 hours and an additional 6,000 gallons added as margin to account for the volume of condensate that is unviewable by the operator (i.e., below the condensate storage tank nozzle for the level instruments).

ADV sizing

The ADVs were originally sized for a 6"x 3" internal diameter. As part of original design, no calculation was done to determine if this sizing was adequate to achieve a hot shutdown temperature of 300°F in 3 1/2 hours. Based on the analysis done for the sizing of the Unit 2 ADVs, it was determined that the existing Unit 1 ADVs (Ref. FPL calculation performed for design modification PM/C 244/77, "Modification to ADVs and CST" dated 7/27/77) were undersized, as they did not possess the capacity at lower steam generator pressures to provide sufficient cooling to the primary system. Based on Unit 2 sizing of the ADVs, the size of the Unit 1 ADV internal diameter was modified to 6"x 4". The original design basis of the plant was also changed to remain at hot standby of 532°F for 1 hour prior to commencing to hot shutdown at the upper temperature limit of 325°F instead of 300°F. The design basis was changed by Amendment No.28 to Facility Operating License No. DPR-67 in 1978. The heat relieving capability of the ADVs was determined for a reactor power of 2560 MWt with the ADVs on both steam generators being available due to their capability to be manually operated on loss-of-offsite-power. <u>The effect of remaining at hot standby for one hour on the amount of condensate required was not however re-evaluated.</u>

Power Upgrade/NUREG 0737 Modifications

In 1981, via Amendment No.48 to Facility Operating License No. DPR-67, licensed reactor power was stretched from 2560 MWt to 2700 MWt. The same year, to address NUREG-0737 and NUREG-0578, the AFW system was redesigned for automatic actuation instead of manual operation. This redesign led to changes in the time delays for the system to either actuate to initiate cooling flow or to isolate flow to the faulted steam generator. Also, in response to NUREG-0737 and branch technical position (BTP) ICSB 13, the AFW system design capability to remove primary system decay and sensible heat for a loss of normal feedwater flow had to be analyzed assuming a concurrent line break in the AFW system together with or without a LOOP and the most limiting single active failure. The AFW line break was not part of the original design basis. In the UFSAR, the AFW line break was analyzed as part of the AFW system rather than as a design basis accident. The redesigned system was evaluated for those plant conditions where there is a loss of secondary cooling in order to establish time delays for initiation/isolation of the AFW system, for system

flow requirements, and for the volume of condensate required for safe shutdown. The plant conditions and the corresponding AFW system requirements are shown in Table 1.

The transient analysis for a loss-of-offsite-power event was performed assuming a maximum delay of 900 seconds before initiation of the AFW system and the most limiting single active failure of the turbine driven pump. This delay is considered conservative because it allows the steam generator to dry out and minimizes heat transfer from the primary to the secondary system after AFW is initiated. For this event, the transient analyses assumes sufficient condensate would be available as a supply for the AFW pumps. A reanalysis of the condensate requirements due to the power upgrade to 2700 MW was not performed.

The loss of normal feedwater flow event, described in UFSAR Section 15.2.8, is bounded by the analysis in UFSAR Section 10.5.3. In the Section 10.5.3 analysis, a break in the AFW line is considered concurrent with the most limiting single active failure, whereas in the Section 15.2.8 analysis, a line break in the AFW system is not considered. For the loss of normal feedwater flow event, the most limiting case with offsite power being available is a single active failure in the station A or B battery and an AFW line break at the discharge of the turbine driven AFW pumps. This event limits availability of cooling flow from one motor driven pump to a single steam generator being available to remove heat from the secondary side. Heat from operation of the reactor coolant pumps is added to the decay heat generated by the primary system. Initially, heat from all 4 pumps is added and later from only 2 pumps, as 2 of the pumps are required to be manually tripped after 1800 seconds into the event. With only one steam generator being available for cooldown plus the additional heat from the reactor coolant pumps, a higher volume of condensate may be required to achieve and remain at hot standby of 532°F for 8 hours than the volume of condensate determined in the original design bases calculation, F-PEC-76. In that calculation, both steam generators were assumed to be available and the heat from the reactor coolant pumps was not included. The calculation also did not address issues such as the impact of the line break on the net positive suction head of the available or operating pump and the resulting loss of condensate.

As a bound for the amount of condensate that would likely be required, an analysis performed by CE for the design of the Unit 2 auxiliary feedwater system, CE letter L-CE-2082 "Atmospheric Steam Dump, Condensate Storage Requirements" dated 3/31/77, indicates that 160,000 gallons of condensate would be required with one steam generator available to achieve hot shutdown entry conditions of 325°F in about 9.5 hours. The volume of condensate required to achieve and remain at 532°F for 8 hours is expected to be less than 160,000 gallons.

Station Blackout

For a station blackout condition (SBO), the AFW system in conjunction with the steam discharge to the atmosphere from the main steam safety valves provides the cooling on the secondary side for the primary coolant to maintain the reactor at hot standby conditions. For this event, the AFW system is

Table 1. Accident Analysis

Event Name	Analysis Reference	Concurrent Single Active Failure/ Line Break	Design Basis Requirement	AFAS Initiation/ Isolation Time	Volume of Condensate Required	Remarks
Loss-of-offsite power (LOOP)	UFSAR Section 15.2.9. CE calc. F-PEC-76	Single active failure of turbine driven pump	Hot standby at 532°F for 1 hr. followed by hot shutdown to 325°F in next 3 1/2 hrs.	900s delay to initiate (Table 15.2.9-3)	Indeterminate but less than 160,000 gallons.	The 100,000 gallons indicated in UFSAR Sec. 10.5.3 has been determined from CE calc. F-PEC-76, and corresponds to the volume required to be in hot shutdown at 300°F in 3 1/2 hrs. following reactor trip. No specific calculation exists to show the amount of condensate required to be at hot standby at 532°F for 1 hr. followed by hot shutdown to 325°F in next 3 1/2 hrs. at 2700 MW.

Event Name	Analysis Reference	Concurrent Single Active Failure/ Line Break	Design Basis Requirement	AFAS Initiation/ Isolation Time	Volume of Condensate Required	Remarks
Station Blackout	UFSAR Section 15.2.2 CE calc. F-PEC-76	None	Hold at hot standby of 532 °F for 4 hrs.	305s (max.) AFAS time delay	85,000 gallons	Based on CE calc., for station blackout condition, with no time delay, reactor power level of 2570MWt and a flow of 500 gpm, 85,000 gal. are required to maintain the plant at hot standby for 4 hours
Loss of normal feedwater flow	UFSAR Section 10.5.3 and 15.2.8	Single active failure in "A" or "B" station battery and AFW high energy line break at turbine driven pump discharge	Hot standby at 532°F for 8 hrs.	305s (max.) AFAS time delay	Indeterminate but less than 160,000 gallons	FPL to do reanalysis to account for availability of only one steam generator for cooldown, power uprate, time delays, addition of sensible heat from RC pumps

Event Name	Analysis Reference	Concurrent Single Active Failure/ Line Break	Design Basis Requirement	AFAS Initiation/ Isolation Time	Volume of Condensate Required	Remarks
Feedwater line break	UFSAR Section 15.2.8. FPL letter L-81-4 to NRC dated 1-2-81	Loss-of-offsite power with single active failure of the turbine driven pump in AFW system (original design bases)	Hot Standby at 532°F for 8 hrs.	205s delay to isolate flow to faulted SG 364s delay to initiate	Bounded by loss of feedwater flow analysis which is currently indeterm-inate	AFW required for long term cooling. Enough condensate and make-up capacity exists.
Main steam line break	UFSAR Section 15.4.6	Loss-of-offsite power	Hot Standby at 532°F for 8 hrs.	180s delay to isolate AFW flow flow to faulted SG	Bounded by loss of feedwater flow analysis which is currently indeterm-inate	AFW required for long term cooling. Enough condensate and make-up capacity exists.

considered to actuate within the designed time delay. The initial flow to the steam generator will be the designed flow of about 250 gpm to each generator. The long term flow is regulated to about 150 gpm to each generator, sufficient to remove decay heat and maintain the reactor at hot standby condition. As licensed for St. Lucie, the SBO condition is for only 4 hours, and the amount of condensate required as per CE calculation F-PEC-76 is about 85,000 gallons.

E1.2.2.1.3 Conclusion

The Unit 1 condensate storage tank has a capacity of 250,000 gallons and the tank anchorage and piping connections have been seismically analyzed for a storage capacity of 160,000 gallons. There is an administrative control to maintain the level in the tank greater than 178,000 gallons. There is also a cross-tie to the Unit 2 condensate storage tank that ensures a dedicated supply of 125,000 gallons to Unit 1. In addition, there are emergency and off normal procedures which identify the cooldown rates with 1 or 2 steam generators available, alert the operators of the amount of condensate required, and provide methods to obtain alternate makeup sources within the time frames needed. The plant preferences for the make-up sources are shown below:

(1) cross-tie to Unit 2 condensate storage tank
(2) non-safety-related demineralized water (this is also the normal make-up source)
(3) attempt to restore operation of (2) above
(4) non-safety-related city water
(5) non-safety-related fire water

Based on the administratively controlled limit of 178,000 gallons and alternate sources of condensate, sufficient condensate capacity and makeup appears to be available to cooldown the primary system after any design basis accident. However, the current TS limit of 116,000 gallons may not be sufficient. This is because the cooldown rates used in the original CE calculation were based on cooldown to hot shutdown of 300°F in 3 1/2 hours at 2698 MW versus the current design bases conditions of remaining at 532°F for 1 hour and then cooldown to 325°F at 2700 MW.

In addition, no calculation was performed to evaluate the condensate requirements to mitigate a break in the auxiliary feedwater line with only one steam generator available to provide secondary cooling. The licensee has indicated they plan to address the team's concerns by performing the following activities:

a. Review the St. Lucie Unit 1 & 2 UFSAR/TS/Design Basis Documents to identify all analyzed accidents/scenarios requiring auxiliary feedwater and determine the applicability to sizing the condensate storage tank.

b. Define the required flow rates, cooldown rates and durations of each scenario.

c. Define auxiliary feedwater requirement calculation assumptions for each scenario.

d. Prepare a calculation to supersede the existing condensate storage tank volume/ADV sizing calculations.

e. Review and revise, if necessary, current Unit 1 & 2 condensate storage tank volume requirement in TS.

f. Update the UFSAR and Design Basis Documents to reflect the new calculation assumptions and conclusions.

g. Review plant procedures/condensate storage tank level alarm set points against the volume requirements determined above to verify the auxiliary feedwater system operation meets the requirement of the calculation. Determine if the Emergency Operating Procedures (EOPs), other procedures, or set points need to be revised.

These issues concerning the required volume of condensate to be maintained in the condensate storage tank and the acceptability of the current technical specification are identified as INSPECTOR FOLLOWUP ITEM 50-335/96-201-01.

E1.2.2.2 Pump Redundancy and Flow Capability

E1.2.2.2.1 Scope of Review

Determine if AFW system is designed with pump redundancy and flow capability to respond timely and effectively to the various plant operating conditions, and remove decay and residual heat to bring the plant to safe shutdown.

E1.2.2.2.2 Inspection Findings

The AFW system was designed with two motor driven pumps and one turbine driven pump. This arrangement meets the position stated in Branch Technical Position 10-1 for pump redundancy. Each of the two motor driven pumps A and B is powered from separate emergency AC sources and the turbine driven C pump uses steam from either of the two steam generators to drive it. The DC control power can be supplied from either the A or B station battery.

The limiting design case for the turbine AFW pump is the station blackout plant condition. The motor driven pumps are designed to meet the requirements for a loss of normal feedwater or steam line break.

Each of the motor driven pumps have a design flow capacity of 325 gpm at a discharge pressure head of 2725 feet (Ref. Dwg. 8770-6078, "AUX. STM. GEN. FD. PUMP 711-N-0675 PERFORM. TEST CURVE" and Dwg. 8770-6079, " AUX. STM. GEN. FD. PUMP 711-N-0676 PERFORM. TEST CURVE"). The turbine driven pump has a design flow capacity of 600 gpm at a discharge pressure head of 2660 feet (Ref. Dwg. 8770-6083, "AUX. STM. GEN. FD. PUMP 711-N-0677 PERFORM. TEST CURVE").

The original design basis for the motor driven pumps was to deliver 250 gpm, excluding 75 gpm flow in the recirculation line, at a steam generator pressure of 1000 psia. The design basis for the turbine driven pumps was to deliver 500 gpm, excluding 100 gpm flow in the recirculation line, at a steam generator pressure of 1000 psia.

In response to NUREG-0737 and BTP ICSB 13 the accident analyses were reperformed to include an AFW line break concurrent with a loss of normal feedwater. This was not part of the original design basis. For the AFW motor driven pumps, the limiting design case is a loss of normal feedwater flow concurrent with a single active failure in the station A or B battery and an AFW line break at the discharge of the turbine driven AFW feed pump. This limits the flow to that provided by one motor-driven pump only. The transient event analyzed for this case assumed an AFW flow of 400 gpm to the steam generator at a steam generator back pressure of 910 psia to achieve safe shutdown. This 400 gpm is higher than the original 250 gpm pump design flow required to be delivered by the motor driven pump.

The original calculation PSL-1EJM-70-035, "Aux. Steam Generator Feed Pumps" dated 7/2/70 and the current calculation PSL-1FSM-96-016, "Motor Driven Auxiliary Feedwater Pump Flow Determination" dated 11/2/96 were reviewed against the pump performance curves to determine the delivered flow to the steam generator. The review determined that the expected flow delivered by a motor-driven pump at 910 psia steam generator pressure would be less than 400 gpm. The pump performance curve indicated an actual flow of 392 gpm. As a result of this concern the licensee performed an evaluation, JPN -PSL-SEFJ-96-050, "Operability assessment of the Motor Driven AFW Pump Delivered Flow" dated 11/6/96, and determined that the revised lower flow would not impact the minimum steam generator inventory used in the transient analysis. The licensee has agreed to revise the current calculation, PSL-1FSM-96-016, to make the necessary corrections as part of CR 96-9737.

The limiting design case for the AFW turbine driven pump is a station blackout. The transient analysis performed for this event assumed an AFW flow of 300 gpm to each of the two steam generators or a total flow of 600 gpm against a steam generator back pressure of 1000 psia or 985 psig. The original calculation PSL-1EJM-70-036,"Total Developed Head, TDH, Turbine Driven Pump" dated 7/7/70 was reviewed against the pump performance curve (Ref. Dwg. 8770-6083, "AUX. STM. GEN. FD. PUMP 711-N-0677 PERFORM. TEST CURVE") to determine the delivered flow to the steam generator. <u>The team's review indicated that the pump was not capable of providing a flow of 600 gpm at a steam generator pressure of 1000 psia.</u> The maximum flow that can be achieved by the pump at a steam generator pressure of 1000 psia is estimated to be above 500 gpm but less than 600 gpm; however, based on an analysis performed previously per JPN-PSL-SEFJ-96-050, the lower flow rates do not appear to be significant with respect to reactor core response. The licensee has agreed to revise calculation PSL-1FSM -96-016, to determine the turbine-driven AFW pump flow characteristics as part of CR 96-9737.

Reg. Guide 1.97 requires the flow indicators in the pump discharge lines to have a scale range that is 110% of the maximum anticipated design flow in the pump discharge lines. The existing scale range of 0-400 gpm for the motor-

driven pump was adequate based on the original maximum design flow of 325 gpm. With the new revised maximum design flow of 400 gpm in the system for the motor-driven pumps, the existing scale will not read 110% of the maximum anticipated design flow. Similarly, the existing scale range of 0-600 gpm for the turbine -driven pump was adequate based on the maximum design flow of 500 gpm. The existing scale range for this pump may also not meet the maximum flow range specified in REG 1.97. The licensee has issued CR 97-0026 to recalibrate the flow indicators and increase the scale range for the indicators in the motor-driven pump discharge lines. Also, as part of this CR, the licensee will take similar corrective actions for the turbine-driven pump discharge line if 110% of the maximum anticipated design flow in the system exceeds the existing scale range.

E1.2.2.2.3 Conclusion

The AFW system is designed with sufficient drive diversity. Inputs to the current design basis calculations assume higher flows than originally designed. The ability of the installed AFW pumps to achieve these higher flow rates has not been demonstrated, however the license's initial evaluation of this issue has determined the effect on the accident analysis to be negligible. The licensee plans to revise the calculations as per CR 96-9737.

The design flow in the motor-driven pump discharge line exceeds the current scale range of the flow indicators as specified in RG 1.97. The licensee has issued CR 97-0026 to recalibrate the flow indicators and increase the scale range for the indicators in the motor-driven pump discharge lines. The licensee also plans to do the same for the turbine-driven pump discharge line if 110% of the maximum anticipated design flow in the system exceeds the existing scale range.

The calculations supporting AFW pump flow requirements and the corresponding AFW pump flow indicators are identified as INSPECTOR FOLLOW-UP ITEM 50-335/96-201-02.

E1.2.2.3 Net Positive Suction Head

E1.2.2.3.1 Scope of Review

Determine if sufficient Net Positive Suction Head (NPSH) is available for the motor-driven and turbine-driven AFW pumps with suction from either the Unit 1 condensate storage tank or the Unit 2 condensate storage tank.

E1.2.2.3.2 Inspection Findings

Calculation No. PSL-1EJM-72-007, "NPSH Calculation for Auxiliary Feed Pumps" dated 9/7/72 was reviewed for both motor-driven and turbine-driven AFW pumps. The NPSH required for the motor-driven pumps at the design capacity of 325 gpm (250 gpm discharge flow to the steam generator at 1000 psia steam generator pressure and 75 gpm recirculation flow) is 12.5 feet. The NPSH required at 130% capacity, i.e., a flow of about 420 gpm, is 16 feet (Ref. Dwg. 8770-6078, "AUX. STM. GEN. FD. PUMP 711-N-0675 PERFORM. TEST CURVE" and Dwg. 8770-6079, "AUX. STM. GEN. FD. PUMP 711-N-0676 PERFORM. TEST CURVE"). The available NPSH

for suction from the Unit 1 condensate storage tank for the design flow is 27 feet. The available NPSH at 130% flow is 25 feet. This calculation is conservative for it assumes a static head of one foot only whereas a larger static head is available.

The NPSH required for the turbine-driven pumps at the design capacity of 600 gpm (500 gpm discharge flow to the steam generator at 1000 psia steam generator pressure and 100 gpm recirculation flow) is 20 feet at 3600 RPM. The NPSH required at 2000 RPM is 17 feet (Ref. Dwg. 8770-6083, "AUX. STM. GEN. FD. PUMP 711-N-0677 PERFORM. TEST CURVE"). The available NPSH for suction from the Unit 1 condensate storage tank for design flow is 28 feet. This calculation is conservative for it assumes a static head of only one foot whereas a larger static head is available.

During preparation for the inspection, the licensee identified that although a modification was performed to cross-tie Unit 2 condensate storage tank to the suction of Unit 1 AFW pumps, a calculation for the NPSH available for the AFW pumps from Unit 2 condensate storage tank could not be found. The suction lines from Unit 2 condensate storage tank are however similar to the suction lines for the Unit 1 AFW pumps, and the difference in piping pressure drop appears to be negligible. In addition, the static suction head from Unit 2 condensate storage tank is much larger than from Unit 1. Based on this, the team concluded that adequate NPSH would be available for the pumps.

Also, when the modification for cross-tie to Unit 2 condensate storage tank was made, modifications were made to the suction line from the Unit 1 condensate storage tank to prevent any flow from the Unit 2 condensate storage tank to the Unit 1 condensate storage tank. This modification increased the pressure drops in the piping and affects the available NPSH. However, this change in NPSH is not expected to effect the operation of the AFW pumps as an adequate NPSH margin is available for the AFW pumps.

The licensee has taken initiated corrective actions (CR 96-2758) to perform a calculation to document the available NPSH for the Unit 1 AFW pumps when taking suction from the Unit 2 condensate storage tank and revise the existing calculation for NPSH for suction from the Unit 1 condensate storage tank to reflect the existing piping configuration.

E1.2.2.3.3 Conclusion

Adequate NPSH is available for AFW pump operation for suction from the Unit 1 and the Unit 2 condensate storage tanks. In CR 96-2758, the licensee indicated they will perform a calculation to document the exact NPSH available for the Unit 1 AFW pumps when taking suction from the Unit 2 condensate storage tank. The licensee also plans to revise the existing calculation for NPSH when taking suction from the Unit 1 condensate storage tank to reflect existing piping configuration.

The need to revise the calculations for net positive suction head is identified as INSPECTOR FOLLOW-UP ITEM 50-335/96-201-03.

E1.2.2.4 Piping Design

E1.2.2.4.1 Scope of Review

Review system piping design and configuration for adherence to ASME Class boundaries, system and safety class breaks.

E1.2.2.4.2 Inspection Findings

Containment isolation for the portion of the AFW system piping penetrating containment is provided by an inboard and outboard check valve. This is an exception to GDC 55, but was an approved part of the original design as per the UFSAR. The check valves have been maintained as part of the licensee check valve program. Until recently, no leak rate testing has been performed for the check valves. The check valves have been leaking, as evidenced by higher temperatures in the piping upstream of the check valves. The temperature of the upstream piping is monitored every shift to ensure that leakage is not significant and that thermal binding of the AFW pumps will not occur. As a result of this finding, the licensee is revising their existing procedures for testing these valves to include leak rate testing at every outage and, if the leak is found to be greater than 2 gpm, to refurbish the valve.

In preparation for the inspection, the licensee identified that the temperature of the discharge piping upstream of the outboard containment isolation check valve was greater than the design temperature of 120°F. This portion of the pipe is a schedule 80 carbon steel pipe and is rated for a much higher temperature than 120°F, therefore, there does not appear to be an immediate operability concern. The stresses in this portion of the piping are being reevaluated by the licensee to account for the higher temperature under corrective action CR 96-2063.

The maximum pressure in the motor-driven pump discharge line, Line No. I-BF-28 has been identified as 1420 psig in Isometric Drawing Number 8770-G-125, Sheet BF-M-8, Revision 4. However, as per the Technical Specifications, the maximum developed pump shut off head is 1465 psig. The discharge line was not designed for a pressure equal to or higher than the pump shut-off head of 1465 psig. The pressure identified in the Isometric drawing is lower than the pressure the piping will actually be exposed to. The existing discharge piping for Line No. I-BF-28 is a schedule 80, 4-inch carbon steel pipe. For the maximum AFW temperature of 120°F, this piping will be able to withstand a maximum pressure of 1900 psig without any undue stress on the piping. Pipe supports for Line No. I-BF-28 may have to be evaluated for the increased pressure in the piping. The licensee has agreed to reevaluate this portion of the piping for increased pressure and revise design documents as necessary as part of CR 96-2972.

The AFW system meets the requirements for ASME Section III Class C. The portion of the piping that interfaces with the feedwater and main steam system are designed to ASME Section III Class B requirements. The AFW system interface with Unit 2 condensate storage tank meets the requirements of GDC 5 regarding isolation from non-safety-related systems.

The higher than originally designed flows (see section E1.2.2.2) will cause a small velocity increase in the both the AFW pump suction and discharge piping. This system is used for a very small fraction of plant life, and the small increase in velocity is not expected to cause any additional erosion or corrosion problem in the pipes.

E1.2.2.4.3 Conclusion

The safety-related portion of the AFW system piping meets the requirements of ASME Section III Class B and C with the exception of the discharge piping upstream of the outboard containment isolation check valve and the motor-driven pump discharge line. The licensee initiated CR 96-2972 to evaluate the acceptability of this piping and piping supports. The piping itself appears to be acceptable due to adequate design margins.

The check valves used for containment isolation have been leaking. The licensee is revising their existing procedures for testing these valves to include leak rate testing at every outage, and if the leak is found to be greater than 2 gpm, to refurbish the valves.

The licensee's actions to evaluate the pump discharge piping and to update the containment check valve testing procedure are identified as Inspector Followup Item #50-335/96-201-04.

E1.2.2.5 Environmental Qualification

E1.2.2.5.1 Scope of Review

Review environmental qualification of the Terry Turbine Woodward Governor Control to determine that it will perform its safety function in the environment in which it is installed.

E1.2.2.5.2 Inspection Findings

The Terry Turbine Woodward Governor Control panel is located in the turbine pump area underneath the main steam and feedwater trestle. EQ Documentation Package 1000, page 1000-3-7 discusses a feedwater or main steam high energy line break in this area. For this break, a steam environment is postulated with a steam temperature of 320°F for a total duration of 60 to 95 seconds (depending on initial power level) during which time the affected steam generator blows dry. This break would make that area a harsh environment as defined by 10CFR50.49 and would require that the equipment be qualified for its operating environment by either testing or analysis.

The team identified that the licensee has not considered the Woodward Governor Control as part of their EQ program. The licensee classified the equipment as being in a mild environment not within the scope of 10 CFR 50.49 based on the short duration of the exposure and the protection provided by equipment enclosures. The licensee stated the temperature increase inside the enclosure will lag the outside temperature due to insulation provided by the enclosure and the air space internal to the enclosure. The licensee was also trying to

retrieve some earlier documentation to demonstrate that though qualification was not required, the Woodward Governor Control could be qualified for the plant accident condition.

E1.2.2.5.3 Conclusion

The team's interpretation of 10 CFR 50.49 would require environmental qualification of the Terry Turbine Woodward Governor Control, regardless of any postulated temperature lag. An analysis for temperature lag could be used as part of the qualification analysis, but is not sufficient for excluding the equipment from environmental qualification. The licensee has initiated CR 97-0046 to address the team's concerns regarding this issue. The environmental qualification of the Woodward Governor Controls is identified as Unresolved Item # 50-335/96-201-05.

E1.2.2.6 Cross-tie Connections

E1.2.2.6.1 Scope of Review

Review the cross-tie connections for the Unit 2 condensate storage tank to the suction of the Unit 1 AFW pumps.

E1.2.2.6.2 Inspection Findings

Normally closed manual isolation valves are provided to isolate the Unit 2 condensate storage tank from the Unit 1 AFW pumps. These valves are classified as ASME Section III valves and are required to be manually opened to cross-tie the Unit 2 condensate storage tank to the suction of Unit 1 AFW pumps.

Procedure ONOP.1-0700031, Appendix D directs the operator to supply the Unit 1 AFW pumps from the Unit 2 condensate storage tank whenever off-normal requirements exists. The Job Performance Qualification requirements (JPM 108-21-06) for operator training require that the isolation valves be opened within 15 minutes after the operator is given the instruction. The isolation valves are ASME Section III valves that are required to operate to perform a safety function. The valves, however, have not been included in the licensee's ASME Section XI Inservice Testing Program. The valves have, however, been stroked and the valve mechanisms re-lubricated on an annual basis. Hence, the valves have been demonstrated to be operable.

The licensee was asked to provide test run data, test procedures, or log book verification to demonstrate that full flow testing had been performed for the cross-tie line with the suction of the Unit 1 AFW pumps tied to the Unit 2 condensate storage tank. The licensee provided flow totalizer indication data in the cross-tie line, but this did not substantiate that full flow testing had been done for the cross-tie line. Consequently, a procedure change to implement full flow testing of the cross-tie line was initiated by CR 96-2864.

E1.2.2.6.3 Conclusion

The normally closed cross-tie isolation valves had not been included in the licensee's ASME Section XI Inservice Testing Program. CR 96-2864 was initiated by the licensee to include these valves in the Section XI Inservice Testing Program and to initiate procedure change documents. Also the licensee could not substantiate that full flow testing had been done for the cross-tie line.

The need for performing full flow testing of the CST cross-flow line to Unit 2 and for performing testing of the associated valves per ASME Section XI is identified as INSPECTOR FOLLOW-UP ITEM #50-335/96-201-06.

E1.2.3 Electrical Design

The electrical design assessment consisted of reviewing design and licensing bases documents that included calculations, specifications, vendor manuals, the Updated Final Safety Analysis Report (UFSAR), the Safety Evaluation Report (SER), and Technical Specifications. Attention was given to the specific electrical attributes applicable to the AFW system.

E1.2.3.1 DC System and Batteries

E1.2.3.1.1 Scope of Review

Determine that the DC system and batteries are designed to have sufficient capacity and capability to provide AFW flow for 2 hours independent of any AC source.

E1.2.3.1.2 Inspection Findings

Load profile calculation, PSL-1-FJE-90-015 "Safety Related Batteries 1A and 1B" Rev.0 dated 1/10/91, was reviewed for battery sizing. The battery sizing calculations were performed in accordance with standard industry practice. The methodology follows IEEE Standard 485-1983. The calculation considered the lowest cell temperature as 50°F and utilized a factor of 1.25 to account for battery aging effects. A design margin of 54% was also provided for the batteries which exceeds the 10-15% recommended by IEEE 485-1983. The design calculation used vendor provided capacity curves that are based upon a fully charged cell with a nominal fully charged electrolyte specific gravity of 1.215 +/- 0.010. The calculations support the system functions and design bases for the DC power requirements.

Unlike the design calculation, the surveillance requirements use a specific gravity acceptance criteria of 1.195 or 1.190, as stated in the Technical Specification. The team identified that meeting this technical specification would not necessarily ensure that the batteries could perform to their calculated design capacity. The Technical Specification acceptance criteria of 1.195 or 1.190 does not envelope the specific gravity of 1.215 +/- 0.010 used in the design calculation. Upon further investigation, the team learned that the technical specification requirement is based on standard industry numbers and is not necessarily intended to demonstrate design capability of

the batteries. During battery surveillances, not one, but a number of battery parameters are measured. The assessment of battery operability is based on all the collective data.

The team also identified that UFSAR Table 8.3-5 is currently not representative of the "Load Profile" as shown in the calculation PSL-1-FJE-90-015 and UFSAR Figure 8.3-14. UFSAR Table 8.3-5 lists emergency loads considered for sizing of the 1A/1B battery bank whereas calculation PSL-1-FJE-90-0015 and the procurement specifications to which the batteries were purchased are more conservative and include the actual calculated worst-case emergency loads plus other additional loads necessary for operational convenience. Plant Manager Action Item, PMAI PM96-12-194 has been initiated based on the team's finding to revise UFSAR table 8.3-5 to refer to calculation PSL-1-F-J-E-90-0015 and UFSAR figure 8.3-14.

E1.2.3.1.3 Conclusion

The batteries have sufficient design margin to account for varying conditions that would occur during normal operation. A review of the related calculations and design documentation indicates that the batteries meet their design bases. The licensee's failure to keep UFSAR Figure 8.3-14 up to date as required by 10CFR50.71(e) is identified as UNRESOLVED ITEM 50-335 and 389/201-01.

E1.2.3.2 System Voltage

E1.2.3.2.1 Scope of Review

Review the design adequacy of station electric distribution system voltages.

E1.2.3.2.2 Inspection Findings

The auxiliary electric system bus voltage is maintained at the Technical Specification tolerance of +/- 10% (except for transient loading) of the nominal voltage which envelopes the design criteria. The +/- 10% tolerance in voltage is to account for offsite system variations and variations in plant auxiliary system loading. This is also the standard industry criteria for design of the auxiliary electric system when fed from off-site power. For the on-site emergency diesel bus, the actual tolerance is much smaller because the voltage can be regulated.

The Technical Specifications acceptance criteria of 4160 +/- 420 volts for the emergency diesels appeared to be too wide of a range to ensure operability of diesel voltage regulators. The team later learned that the basis for this specification was only to verify that the diesel had started and was up to speed and is based on the standard industry practice of +/- 10% of nominal voltage for the auxiliary power system and not on the capabilities of the diesels. The technical specification was not designed to verify acceptable voltage regulation. The actual regulation of the system voltage while on the diesels during steady state conditions is better than +/- 10% since the diesel voltage regulator compensates for changes in loading conditions. The voltage regulators on the diesels have a setting tolerance of +/- 0.25%, and the

accuracy and repeatability of a typical voltage regulator is about 1%. The settings for the voltage regulator are described in procedure #1-0950187, Appendix B; however, this procedure does not provide a tolerance range for the setting. The licensee initiated PMAI 97-01-092 to revise the test procedure. The voltage regulators have not demonstrated a tendency to drift and operation of the voltage regulator significantly outside of the tolerance limits would be analogous to failure of the regulators. The voltage regulator settings are checked every 18 months during diesel preventative maintenance.

When the buses are fed from offsite power, the auxiliary system transformer taps are used to regulate the voltage. The transformer tap settings are clearly marked on each transformer next to the transformer tap adjusting lever. Plant drawings 8770-G-272, 274, and 2998-G-272, 274 Sheets 1 and 2 define the tap settings for the transformers. The taps cannot be changed with the transformers energized and all transformer work is performed by the offsite Protection and Control Substation Group but is controlled by on-site personnel.

A review was performed of Calculations 8770-A-452, sheets 3, 7, and 7a, PSL-1-FEPSTR-1991-0102 and 0103, 2998-A-452, sheets 3, 7, and 7a, and PSL-2-FEPSTR-1991-0102 and 0103, and the Unit 1 Minimum Excitation Limiter setting calculation PSL-1-FEPSTO-1991-0205, rev. 0 for the main generator exciters and the main generator volts/hertz relays to ensure that the settings are within the design envelope. The team concluded that the relay settings are within the design envelop and provide adequate protection.

E1.2.3.2.3 Conclusion

The buses are adequately sized for the connected loads and short-circuit duties. The voltage ratings are also adequate for the application. As a result of the team's concerns raised regarding testing of the diesel voltage regulators, FP&L initiated two revisions to existing procedures (see PMAI 97-01-092): (a) Revise the maintenance procedure for the EDGs to include confirmation of the specific voltage regulator settings for terminal voltage; and (b) Revise the monthly EDG surveillance procedure to include a check of the terminal voltage at rated speed when the EDG is in the stand-by mode.

E1.2.3.3 Cable Sizing

E1.2.3.3.1 Scope of Review

Determine if cables are adequately sized for the equipment ratings and the short circuit duties.

E1.2.3.3.2 Inspection Findings

Short circuit calculation EC-039 for the 125 VDC batteries 1A and 1B was reviewed to determine if the short circuit currents on the DC system are within the switchgear, breaker, and cable ratings. The design methodology was found to be conventional and the assumptions reasonable. The calculation indicated the design requirements were met for the switchgear, breakers, and cables. Overall, this calculation was thorough and comprehensive.

The AC cables do not have a specific calculation that analyzes temperature rise versus short circuit duty like the DC system; however, all related documentation such as the original cable criteria in WHL-8 indicates the cables are adequately sized for the equipment ratings and the present short circuit duties.

Assumption 4.3 in calculation PSL-1FJE-94-002(GL 89-10 125vDC Motor Operated Valve Cable Voltage) identified an incorrect cable length of 4600 feet versus the actual 460 ft. This was determined to be a typographical error and DCR 960304 was issued 12/3/96 to correct the cable length on the Unit 1 Cable and Conduit List database.

A specified design basis temperature of 120°F inside containment, 104°F outside containment in adjacent buildings, and 93°F dry bulb (for 99.7% of the time) and 101 degrees F (for 0.3% or 30 hours)(EQ Doc package 8770-A-451-1000(U1) and 2998-A-451-1000(U2)) for outside ambient was considered in sizing cables. This was considered to be conservative.

E1.2.3.3.3 Conclusion

The AC and DC cable sizing was determined to be adequate.

E1.2.3.4 Fuse Sizing

E1.2.3.4.1 Scope of Review

Determine that the fuses in the MCCs are adequately sized for their application.

E1.2.3.4.2 Inspection Findings

The basis for sizing the fuses is EBASCO Unit 1 Motor Control Centers' Specification 8770-286. This specification required that control transformers rated at 150 VA be supplied for starter sizes 1 and 2, and that 500 VA transformers be supplied for starter sizes 3, 4, and 5. Further, the specification requested that the hot leg of the secondary circuit be protected by a fuse. The motor control center (MCC) vendor complied with the specification and provided 1.6 amp fuses for the 150 VA transformers and 6 amp fuses for the 500 VA transformers. These fuse sizes are consistent with the fuse sizes shown on drawings 8770-B-335, sheet C.

The size 2 starters have the highest inrush current (4.833 amp) for the 150 VA transformers, and the size 4 starters have the highest inrush current (11.375 amp) for the 500 VA transformers. The values were obtained from calculation EC-007, rev. 4. For these current values, and for the 1.6 amp and 6.25 amp fuses that were actually supplied by the vendor, the fuse opening times would be 26 and 100 seconds, respectively. Therefore, the fuses would not blow prematurely. For steady state conditions, the loading is about 114.6 VA for the size 2 starter and 192.6 VA for the size 4 starter. In each case, the continuous loading is less than the transformer rating and fuse capacity. The team concluded the MCC fuses are properly sized.

E1.2.3.4.3 Conclusion

Control circuit fuses were reviewed for the MCC's. The sizing was found to be acceptable and in accordance with the design criteria.

E1.2.3.5 Review of Modifications

E1.2.3.5.1 Scope of Review

A review was performed of selected design modifications to verify that the modifications did not invalidate the design basis.

E1.2.3.5.2 Inspection Findings

The following 3 modifications were reviewed:

- PC/M No. 95-82, "C" Auxiliary Feedwater Pump Trip and Throttle Valve Limit Switch Replacement"
- PC/M 067-185, "AFAS Interposing Relay Replacement"
- PC/M 233-184, "Auxiliary Feedwater Actuation System Modification."

PC/M No. 95-82 replaced the existing Model LS-2 level switch with a fully qualified NAMCO EA 180-35302 limit switch. For this modification, no cable routing or changes in internal wiring were involved. The new switch materials were determined to be different and the design package did not contain any information regarding the voltage and current ratings of the limit switch. The licensee provided vendor letters and specification sheets that showed that the voltage and current ratings were identical and the switches were functionally equivalent. Similar questions regarding material substitutions were raised regarding PC/M 067-185 and PC/M 233-184. In all cases the substitute materials were acceptable.

E1.2.3.5.3 Overall Conclusion

A review of three PC/M's identified a lack of detail in the packages regarding material substitutions. The licensee was able to demonstrate for the PC/Ms reviewed that the replacement parts were equal to or better than the replaced item. The teams sampling of electrical PC/Ms did not identify any design concerns.

E1.2.3.6 AFW Motor Sizing

E1.2.3.6.1 Scope of Review

The team performed a review to verify that the AFW pump motors can provide adequate torque to drive the AFW feed pumps and start on reduced system voltage.

E1.2.3.6.2 Inspection Findings

The licensee was asked for calculations that look at motor versus pump torque requirements and voltage versus torque for the AFW pump motors. Both pump/motor sets were procured as capable of starting and running with 75% of rated voltage. The licensee indicated that calculations were performed as part of the original procurement contract(s). The following drawings were presented as evidence that this requirement was met:

- Motor data sheet (8770-2336)
- Pump motor speed-torque curve (8770-2334)

The curves were reviewed to ensure that the motors are capable of starting the pumps under all the required operating modes including reduced voltages. The team concluded from these data sheets that the motors are matched to the pumps for torque requirements and are capable of starting and running the pumps under all operating conditions.

For the 4000 volt motors, motor overload protection is basically inactive until 150% of full load current is reached, due to the characteristics of the COM-5 protective relays. Since the setpoints for the relays which implement (Power Systems Branch Position #1) PSB-1 are above 90% of the bus voltage, the increased motor currents may be in a range of 110% of full load current, which is below the protective area of the COM-5 relays. Further, since all the motors were procured with a 1.15 continuous service factor, there would be no detrimental effect due to the decreased voltage. The overload protective tripping is set at 250% of full load current. The alarm is set at 150% of full load current.

For operation just below the PSB-1 degraded voltage relay setting of 3831 volts, the current is still reasonably inversely proportional to voltage (constant kVA). The degraded grid voltage relaying will trip the unit off the line long before any thermal damage occurs. Below 70% voltage, the motor will stall with the current reaching 70% of the locked rotor value. The safe stall time for this current value is in the 2.5 seconds range, whereas the maximum tripping delay of the loss of voltage relaying is 1.5 seconds.

E1.2.3.6.3 Conclusion

Motor sizing is in accordance with the design requirements. Motor/Pump torque curves are matched and capable of starting for reduced voltage scenarios.

E1.2.3.7 125 VDC MOVs

E1.2.3.7.1 Scope of Review

Review the control circuitry/design for selected 125 VDC motor operated valves to verify that they will operate at worst case minimum operating voltage.

E1.2.3.7.2 Inspection Findings

Calculation PSL-1FJE-94-002 was reviewed to determine the starting terminal voltage for the 125 VDC AFW isolation valve motors when their associated DC busses are at worst case minimum operating voltage. With an available voltage at bus 1AB-1 of 109.49 volts, the motor terminal voltages and percent rated torque were calculated and provided as input to the valve operator thrust calculations. The methodology was conventional and the assumptions reasonable. The results support the operating requirements for these motors.

The team also reviewed the "DC MOV Design Inadequacies Study," Report No. FLO-124-37.5000, Rev. 0 which addresses INPO SER 25-88 and NRC Information Notice, IN 88-72. These documents are concerned with design inadequacies affecting DC Motor Operated Valves (MOVs). Specific factors addressed in these documents included maintenance, valve specifications, motor torque under degraded voltage conditions, use of starting resistors to limit inrush current, accident temperature effects, continuous shunt field energization, high voltage motor transients, and pressure locking and/or thermal binding. Implementation of the items applicable to St. Lucie were verified by reviewing the appropriate system diagrams.

E1.2.3.7.3 Conclusion

Calculation PSL-1FJE-94-002 was reviewed to determine the starting terminal voltage for the 125 VDC AFW isolation valve motors when their associated DC busses are at worst case minimum operating voltage. The motor terminal voltages and percent rated torque were calculated and provided as input to the valve operator thrust calculations. The methodology was conventional and the assumptions reasonable. The results support the operating requirements for these motors.

E1.2.3.8 Other Electrical Issues

E1.2.3.8.1 Scope of Review

During the inspection, the team reviewed other miscellaneous electrical issues that were not part of the team's original inspection plan.

E1.2.3.8.2 Inspection Findings

During the inspection, the licensee briefed the team on an issue concerning a lack of procedures and testing for switching DC control power to the turbine driven AFW pump. The licensee indicated that they had recently identified this issue as part of a program they had implemented to review the accuracy of the UFSAR. The licensee had determined that operating procedures had not been written to perform a transfer of DC control power, as necessary to isolate a failed DC bus or battery.

As a result of further review, the licensee also identified that the undervoltage trip feature of the four circuit breakers used to complete the transfer of DC power had never been tested. The licensee issued CR

96-2825 to investigate this concern. A test was performed while the team was on-site to establish functionality of the undervoltage trip device. The undervoltage trip device failed the test for the first breaker tested. Therefore, if a DC bus transfer would have been attempted to restore power to the AB bus, the operable bus would have been closed onto a faulted bus. One or more of the breakers would be expected to open on overcurrent, thus separating the two busses. However, the AB bus would remain without a power source and the attempt would have failed. FP&L's failure to establish operating and testing procedures as necessary to ensure the operability of the DC bus tie breakers, as required by Criterion XI to 10 CFR 50 Appendix B is identified as UNRESOLVED ITEM #50-335/96-201-07. Short term and long term corrective actions for the lack of testing procedures are detailed in LER 96-016 issued on 12/20/96.

E1.2.3.8.3 Conclusion

After failing the first test of the undervoltage device, the team witnessed portions of subsequent licensee testing and troubleshooting activities and reviewed the completed testing/work packages. Based upon this review the team identified the following concerns:

* The original test procedure written for operations to perform the test was well written and received an appropriate level of review. Upon failure of the breaker to function during the test, additional testing and troubleshooting actions were not performed by procedure, but rather, by a plant work order. The initial plant work order, to remove the breaker to the shop and troubleshoot was straightforward; however, prior to removing the breaker, a decision was made to perform additional testing in place, with the breaker installed in the switchgear. This testing was performed without isolating the switchgear and was controlled by a scope change to the original work order. This scope change did not contain an appropriate level of control for the type of work being performed and was not properly integrated with the original work order. For example, it was not clear what steps of the original work order were to be completed prior to performing the steps in the scope change work order. Also, it was not clear, which of the breakers were being worked. Although, the licensee ultimately completed the job successfully, the team was concerned over the lack of adequate procedural controls for this special test. The licensee issued CR 97-0028 to investigate the team's concern and provide appropriate corrective actions. Criterion V to 10 CRF50 Appendix B requires that appropriate procedures be used for activities affecting quality. Criterion XI to 10 CFR50 Appendix B requires testing be performed in accordance with written test procedure. The lack of appropriate written controls for performing the above testing/troubleshooting activities is identified as UNRESOLVED ITEM # 50-335/96-201-08.

* After removing the circuit breaker to the shop and after repairing the undervoltage trip device, the licensee used Maintenance Procedure No. 0940074, Revision 7 to perform overcurrent testing of

molded cased circuit breakers. The team identified that this procedure was confusing with regard to the appropriate acceptance criteria to be used during performance of the overcurrent test. The licensee issued CR 96-2881 which documented the team's concerns with this procedure. The licensee's evaluation of the CR recommended several procedural enhancements that adequately responded to the team's concerns.

E1.2.4 I&C Design

E1.2.4.1 AFW Initiation and Isolation Circuitry

E1.2.4.1.1 Scope of Review

Determine the capability of the Auxiliary Feed Water Actuation System (AFAS) to automatically initiate AFW flow on low steam generator level and isolate AFW flow from the affected steam generator following a main steam line or feed line break

E1.2.4.1.2 Inspection Findings

The AFW system was initially designed for manual actuation on low steam generator level. As a result of NUREG 0578, "TMI-1 Lesson Learned Tasks Force Status Report and Short Term Recommendations," AFAS actuation logic was incorporated into the plant design for automatic initiation/isolation of the AFW flow to the steam generators.

AFW flow to the steam generators is initiated by the AFAS on low steam generator level. As stated on the Total Equipment Data Base Sheet for the AFAS cabinets the low level generator setpoint values for AFAS 1 and AFAS 2 are greater than or equal to 19.0%. The values stated in the Technical Specification Table 3.3-4 are, greater than or equal to 19% with an allowable value of greater than or equal to 18%. Per calculation no. 19367-ICE-36308, Revision 0, dated 9/9/93, " St. Lucie Unit 1 RPS, ESF, and AFAS Setpoints and Allowable Values," the calculated AFAS setpoint values for steam generator level is 18.26 % span and calculated allowable value is 17.94 % span. The setpoint value for steam generator low level envelops both the calculated and the Technical Specifications values.

The AFAS is designed to automatically terminate AFW flow to a faulted steam generator due to a main steam or main feedwater line break or a rupture in the AFW line downstream of the motor-operated isolation valve and to provide flow through the intact AFW line to the un-faulted steam generator. The isolation function for the faulted steam generator or for the ruptured AFW line is initiated by a pressure difference in the main steam or feedwater header line. The results of Setpoint Calculation no. 19367-ICE-36308 were reviewed against the instrument lists and Technical Specification Table 3.3-4 for conformance. The values were in agreement for the documents listed above.

During the review of setpoint calculations the team raised a question regarding the licensee's response to NRC Information Notice IN 89-68, "Evaluation of Instrument Setpoints During Modification." The IN discusses

tracking of setpoint drifts to determine unidirectional drift for an instrument. In case of a unidirectional drift over multiple calibration periods, the IN recommends considering the drift error as a biased error instead of the general practice of including the error as part of the square root of sum of squares (SRSS) methodology. A review of internal licensee documents in response to this IN indicated that a recommendation was made for a follow-up engineering evaluation. The licensee could not provide any documentation to show that a follow-up engineering evaluation had been performed, except for a specific evaluation that was performed for Rosemount transmitters. The licensee initiated CR 97-0037 to address the team's concern.

E1.2.4.1.3 Conclusion

The AFAS system is designed for automatic actuation in accordance with NUREG 0578. The UFSAR commitments for the system agree with the as-built conditions and various design documents. Setpoint calculation 19367-ICE-36308 and the requirements in the Technical Specifications Table 3.3-4 are in conformance.

The calculated setpoint values for the Steam Generator level and pressure differentials in the main steam or feedwater line in calculation 19367-ICE-36308 are in agreement with the Technical Specifications Table 3.3-4 requirements and various design documents.

St. Lucie has no program to track unidirectional drift for multiple calibration periods. The licensee has initiated CR 97-0037 to address the team's concern with this issue. The lack of a program to identify and track unidirectional drift is identified as INSPECTOR FOLLOW-UP ITEM #50-335/96-201-09.

E1.2.4.2 AFW Flow Indication

E1.2.4.2.1 Scope of Review

The AFW pump discharge flow indication in the control room was reviewed against the requirements contained in NUREG 0578 and RG 1.97.

E1.2.4.2.2 Inspection Findings

NUREG-0578 specified that the AFW pump discharge flow indication be safety-grade, provided in the control room, and powered from the emergency buses. The need for redundant channels was relaxed per NUREG 0737 to provide only a single channel flow indication. Also, NUREG 0737 did not specify this single channel to be seismically qualified or powered from a class 1E power source. As stated in RG 1.97, the indication for AFW flow is considered a category 2, type D variable with a required scale range of 0 to 110% of the design flow. Design flow is defined as the maximum flow anticipated in normal operation. Based on review of the instrument list for the Main and Auxiliary Feedwater system and Dwg. 8770-B-327, Sh. 602, the indications provided in the control room exceed the requirements in NUREG 0578 and NUREG 0737 in terms of channel redundancy, seismic qualifications, and class 1E power source requirements.

Per RG 1.97, identification of measurement accuracy is required for type D & E variable display channels. Since the sensors are located in an environment that could become harsh due to a line break, the team raised a question regarding the effect of the postulated environment on the instrument loop accuracy. The licensee indicated that the overall loop accuracy for this indication post accident was calculated at 13.3%. Although this seemed to be a reasonable accuracy for this instrument application, the licensee had not performed a specific analysis to quantify what an acceptable accuracy for this instrument would be. The team learned that generically, the licensee had not performed a specific analysis of acceptable loop accuracies for instruments that are used for indication only. The licensee initiated CR 97-0040 to address this issue.

Design changes in the AFW system have increased design flow in the motor-driven AFW pumps from 250 gpm to about 400 gpm (see Section E1.2.2). The existing scale range for flow indicator reads only up to 400 gpm and will not satisfy RG 1.97 which specifies an indicator scale range of 0-110 % of design flow. CR 97-0026 has been generated to replace the scale on the flow indicator and recalibrate the instrument loop. The required flow requirements for turbine-driven AFW pumps will be evaluated under CR 96-2737 and based on the result of this evaluation the scale range of the flow indicator will be changed and the associated instrument loop will be recalibrated as required.

E1.2.4.2.3 Conclusion

AFW flow indications meet the requirements of NUREG 0578, section 2.1.7.b, and exceed NUREG 0737, section II.E.1.2 requirements. The indications also meet the intent of the indication requirements as identified in RG 1.97. The licensee has not performed a specific analysis of acceptable loop accuracies for instruments that are used for indication only. The licensee initiated CR 97-0040 to address this issue. The lack of analysis to validate the acceptability of loop accuracy calculations for indication only instruments is identified as INSPECTOR FOLLOW-UP ITEM #50-335/96-201-10.

The flow indicator range for the motor driven AFW pumps currently read 0-100 % of design flow and do not meet the RG 1.97 requirement of 0-110 % of design flow because of increased flow in the AFW system due to design changes. CR 97-0026 has been generated to rectify this discrepancy. Based on the results of CR 96-2737, the indication for the Turbine Driven AF pump may also require revision. This item was identified as INSPECTOR FOLLOW-UP ITEM #50-335/96-201-02 in paragraph E1.2.2.2.3 of the report.

E1.2.4.3 CST Tank Level Indication

E1.2.4.3.1 Scope of Review

The condensate storage tank (CST) level indication in the control room was reviewed against the requirements of RG 1.97.

E1.2.4.3.2 Inspection Findings

CST level indication is classified as a RG 1.97 Category 1, Type D indication. Calculation no. PSL-1FJI-92-007, Revision 0 has been performed to document the transmitter's (LT-12-11 & 12) calibration span (ΔP), associated alarm setpoints, and to demonstrate adequate setpoint margin. The results of this calculation were in agreement with the instrument list for the Condensate system.

The team identified a discrepancy between the instrument list for the Condensate system and the RG 1.97 Parameter Summary List. The instrument list identifies the level indicating switches as non RG 1.97 related instruments, whereas the RG 1.97 parameter summary list lists them as RG 1.97 related instruments. The licensee issued CRN 036-196-6578 to rectify the discrepancy.

E1.2.4.3.3 Conclusion

The CST level indication meets the intent of RG 1.97 requirements.

E1.2.4.4 REG Guide 1.97 Instrument History

E1.2.4.4.1 Scope of Review

Review five year component history for the RG 1.97 instruments.

E1.2.4.4.2 Inspection Findings

The five year component history for the RG 1.97 related instruments for AFW and CCW system was reviewed. Several electrical signal spiking problems occurred in the instrument loops during this period. In every instance, the problem was resolved by venting the appropriate transmitter. The licensee identified the probable causes as insufficient sloping of sensing lines, insufficient drop-out in the square root extractor, or poor maintenance practices when placing transmitters back in service after maintenance. Insufficient sloping of the sensing lines and insufficient drop-out in the square root extractor would not appear to be likely root causes since the same instruments do not exhibit the same spiking phenomenon on a routine basis. A walk down of the AFW pump discharge flow transmitters was also performed to determine conformance to the sloping criteria identified on the plant drawings. Requirements for the sensing line slope are 1" per foot for unit 1 and 1/2' per foot for unit 2. Results of the walkdown were non-conclusive, because the sensing lines are located in totally-enclosed tube trays. However, the tubing trays themselves were not sloped.

As a result of the team's questions the licensee issued CR 96-3043 to address the practice of venting transmitters in a instrument loop when the loop experiences a spiking problem.

E1.2.4.4.3 Conclusion

The licensee has experienced some problem with instrument spiking. The root cause of the problem has not been definitively determined. The licensee has initiated CR 96-3043 to investigate the spiking problem.

E1.2.5 AFW Walkdown Observations and Results

E1.2.5.1 Mechanical Walkdown and In-Plant Observations

E1.2.5.1.1 Inspection Scope

The team performed a walkdown of the Unit 1 Auxiliary Feedwater System, including the Unit 1 Condensate Storage Tank and the associated missile protection structure. The initial walkdown was conducted on 11/20/96. Subsequent walkdowns and operation observations were conducted throughout the teams site visits.

E1.2.5.1.2 Observations and Findings

Condensate Storage Tank

The Unit 1 CST nitrogen blanket was not in operation at the time of the inspection. The CST is provided with a nitrogen blanket system to minimize the dissolved oxygen content in the storage tank as described in the Design Basis Document (DBD) design requirements for the CST. Without the N2 blanket on the tank, potential oxygen absorption in the CST contents may cause chemistry control problems for normal secondary system feedwater makeup. Also, direct injection of Auxiliary Feedwater with potentially high O2 levels could be detrimental in the long run since this water goes to the steam generators with no chemistry treatment prior to injection.

The licensee explained that the CST has a "top hat and doughnut" designed vent which has a water seal to isolate the tank air space from the outside atmosphere. In the past the N2 over pressure tended to blow out the water seal due to controller pressure fluctuations. When the seal blowout occurred, the tank upper air space became open to the atmosphere. Since isolating the system, the water seal has done well in precluding significant air absorption. Also, this tank is not the normal main condensate makeup. The Steam Generator Blowdown System tank supplies normal makeup to the hotwell with minimal makeup from the CST. Licensee Chemistry personnel provided the current feedwater chemistry sampling results, stating that no oxygen concentration problems are present. A new design relief assembly has been obtained for the CST Nitrogen system, and an engineering package has been prepared for installation in a future outage.

AFW Pump Areas

The team observed that the motor operated AFW pumps and missile protection separation structures are in an area open to the environment. Measures to protect electrical components from the elements appear to have been effective.

The steam turbine operated AFW pump is also in an area open to the environment. Several signs of corrosion were noted on this equipment, specifically on the steam turbine/pump assembly. The inlet supply steam MOV housing and the speed governor assembly showed the more serious corrosion of components. The licensee identified that the AFW 1C Pump Woodward Speed Governor shaft is scheduled to be replaced with one of a material less susceptible to corrosion than the current shaft. The licensee reported that the shaft for the governor will be replaced during the next refueling outage. Both the Unit 1 and Unit 2 AFW "C" Pumps governor shafts are to be replaced with the improved materials. As a preventive measure, a shaft of the original material was changed out on the 1 C pump assembly during the last major outage. Examination of the removed shaft revealed minor pitting which was not inhibiting operation of the governor device. There have been no failures of the governor identified because of this shaft.

The AFW discharge line connections into the main feed headers are operating at elevated temperatures with apparent back leakage through the AFW header check valves. The elevated temperatures extended through leaking motor operated (AFAS initiation) control valves and continue back upstream towards the pump discharge check valves. There are significant elevation changes in the piping prior to reaching the pump discharge check valves. This physical feature will lessen the probability that thermal conduction flow will increase the temperature of the lines near the AFW pumps. This high temperature issue is being evaluated under CR # 96-2063. Monitoring of the AFW pump casings temperature for back leakage through the pump discharge check valves is currently provided in the plant's surveillance program as a required shift check. The surveillances provide adequate confidence that temperatures are not elevated at the AFW pumps. Also, the licensee has initiated a Temporary Change to OP 1-0700050 to quantify the back-leakage through the AFW header check valves.

AFW MOV Covers

During performance of a walkdown of the Unit 1 auxiliary feedwater system, the team observed that canvas covers had been tied over the top of the pump discharge motor operated valves (MV-09-9, MV-09-10, MV-09-11 and MV-09-12). The team was concerned that the covers or rope could potentially become entangled in the stem of the valves and compromise the valves operation. As such, the licensee was asked to provide a copy of the installation procedure used for the covers and the engineering evaluation which analyzed the acceptability of the installation. The team was told that a documented evaluation had not been performed and a specific installation procedure had not been used. Consequently, the licensee issued CR 96-2870 to evaluate the acceptability of the installation and the operability of the motor operated valves. The evaluation to the CR stated that the installation was acceptable; however, the covers were removed. The CR evaluation also stated that installation of the covers was performed under the guidance of Administrative Procedure ADM-0010432, Rev. 7 as a minor maintenance task. Although this procedure does refer to the installation and removal of covers as a minor maintenance task, it was the team's interpretation that this procedure would only address pre-existing covers. The fabrication/installation of new covers

would require a documented engineering evaluation to assess possible effects on environmental qualification, seismic, and other component operability concerns.

Repetitive Failures of MV 09-11 and Observation of AFW Pump 1C Surveillance Test Operation

During the inspection, the licensee performed a surveillance test on MV-09-11, the 1C AFW pump discharge valve to the 1A steam generator. During the test, the valve failed to close as required, and was declared inoperable by the licensee. Later, the team learned that this valve had previously failed similar surveillances, on 8/1/95, on 7/21/96, and on 7/26/96. On two occasions the previous failures were attributed to dirty torque switch contacts. The contacts were cleaned but the torque switch was not replaced and a conclusive root cause analysis was not performed. On the other occasion, the failure was attributed to a bent stem. The team expressed a concern that the licensee had not identified the root cause of the these failures and had not corrected the problem.

After being notified of the latest test failure, the MOV system engineer took continuity readings of the switch in place and determined that there was no continuity. After removing the switch, the problem went away and could not be duplicated. Upon inspection, the contacts looked clean with no signs of foreign material evident. The licensee replaced the subject torque switch and sent the old switch to FPL's material lab for further analysis. A member of the inspection team observed the licensee's activities in the laboratory. As of the end of the inspection, FPL was unable to determine the cause of the valve's failure.

Although the team agreed that a potential cause of failure was the torque switch, several facts would challenge that assumption. First, the switch has two sets of contacts, one set for open and one set for closed. Previous failures of the valve have been in both the open and closed directions, meaning that there would have had to have been a similar problem with both sets of contacts. Second, no evidence of foreign material was found by the lab, although the lab did note the evidence of abrasion resulting from previous cleaning attempts. Third, when measured after the switch was removed, continuity was good and the failure could not be repeated.

The team witnessed the surveillance performed for the 1C AFW pump after replacing the torque switch. The unit operated approximately 30 minutes with no apparent problems. The team observed that the suction and discharge pressures were well within the acceptable ranges and were steady during the pump run. It was noted that the discharge feed line near the "T" branch line to the steam generators had some vibration when the test was being conducted. The IST Engineer measured the vibrations as being approximately 10 to 10.5 Mils in both axes adjacent to the flow element flanges. Also noted was that when the turbine 1C pump/turbine was stopped, the vibration remained, but was reduced to approximately 9 to 9.5 Mils. The licensee stated that similar pipe vibrations had been evaluated for Unit 2 and that a vibration value of 27 Mils in the piping was acceptable. The 27 Mil criterion was developed based on NUREG-75/087, draft "ANSI/ASME Requirements for Pre-operational and Initial

Startup Vibration Testing of Nuclear Piping Systems," and Ebasco Services Inc. document "Mechanics of Piping Vibration." The team had no further questions regarding the piping vibration.

E1.2.5.1.3 Conclusions

The operation of the Unit 1 CST without the designed N2 system in operation has been adequately addressed by the licensee. The licensee has adequate controls to address the potential adverse environmental conditions in the Motor Driven AFW Pumps area. For the turbine driven pump, the licensee plans to replace the Woodward governor shaft material at the next outage.

Leaking AFW header checkvalves have resulted in elevated temperatures in sections of the AFW lines and continues to be a maintenance concern for the licensee. The licensee has added a more complete testing requirement for these check valves and continues to perform surveillances on a shift basis checking the AFW pump casings for any indication of increased temperatures which will affect the pumps operational performance.

The installation of protective covers on the auxiliary feedwater MOV's without performing a 50.59 screening evaluation and without specific installation procedures is identified as Unresolved Item #50-335/96-201-11.

Although FPL has recently taken extensive actions to identify the root cause of the repetitive failures of MV 09-11, a conclusive root cause has not been determined. In addition, the team was concerned with the relatively poor overall reliability for the C train of the auxiliary feedwater pump, primarily driven by these numerous valve failures.

The licensee's surveillance test of the AFW 1C pump in preparation for returning the system to operational control following the MV 9-09 surveillance and repair was conducted in a successful manner.

E1.2.5.2 Electrical/Instrumentation and Control Walkdown and In-Plant Observations

E1.2.5.2.1 Inspection and Scope

The team conducted a walkdown on 12/4/96 on the 1A battery, battery charger, and associated buses to verify the as built system. The team also conducted a walkdown of the control room to evaluate the identification of RG 1.97 instruments. Subsequent walkdowns and operational observations were conducted throughout the teams site visits.

E1.2.5.2.2 Observations and Findings

During the team's walkdown of the 1A battery room, a vibration was noted coming from the rack for cells 31-60. During the course of the battery inspection the vibration gradually stopped. The source for the vibration was not evident. The team was informed by the licensee that a similar vibration issue had been the subject of a previous PWO (#2758/65), and that the cross

members between cells 53 and 54 had been tightened. At that time, subsequent checks indicated that this corrected the vibration problem. No further actions were taken by the licensee at that time.

As a result of the team's questions, the licensee re-evaluated the vibration and obtained a letter from C & D (the battery manufacturer) which concluded that the vibration would not impact the seismic integrity of the rack which includes the seismic qualification of the battery as a unit. The team had no further questions as overall, the battery condition was noted as being excellent. The 1A and 1AA chargers were observed and were found to be within the design parameters of the calculations. The meters indicated 136 volts and 40 amps.

The 125 VDC Load Test Panel 1A was observed to have only one nut holding the door shut. Condition report # CR-96-2958 was generated for engineering to perform an evaluation of the as-found condition to determine any adverse effect on the seismic qualification, structural integrity, and function of the panel. As a result of this observation, the licensee indicated that all missing nuts were replaced returning the cabinet to its original configuration. The evaluation of the CR also indicated that the as found condition was operable. The team also noted that coming out of the same panel were two conduits that connected to a third conduit with an unusual looking support bracket. The support bracket appeared to be an unauthorized or temporary installation. The licensee was asked to identify this bracket on a print, or provide the T-Mod or design change that installed the bracket. The licensee indicated that the bracket was used during installation for conduit alignment and was not needed as per the design. The strut and clamps were left in place for convenience. Review of the existing conditions revealed that the overall weight of the strut and clamps (approximately 4.0 pounds) is insignificant when compared to the weight of each conduit (approximately 138 pounds, based on a span of 10.75 feet).

The team conducted a walkdown on 12/5/96 of the Unit 1 control room, AFAS panel, cable spreading room, and the hot shutdown panel. The AFW and AFAS panel were as per the current plant diagrams and no adverse findings were identified. With regard to the hot shutdown panel, the team identified that the filters in the back of the panel were dirty. The licensee stated that are no requirements to clean or replace the filters in this panel as there are no components located inside the cabinet that are sensitive to dust accumulation. The team was concerned that if the filters got clogged, they could limit the flow of air through the panel which would lead to increased temperatures inside the panel.

On 12/10/96, a walkdown of the 1B and 2B startup transformers, 2B main transformer, and 2B unit auxiliary transformer was conducted to verify as built conditions and the position of the transformer taps. The transformer taps were set as per the design calculations.

The team reviewed the licensee controls for performing battery cell jumpering and found them to be adequate. The licensee has a calculation and authorizing engineering letter which allows battery cell jumpering, up to two cells per battery.

The team questioned the licensee regarding the availability of instruments to the control room operators to determine the effectiveness of the battery room forced ventilation. The licensee responded that there were several safety related instruments available to the operators to determine the status of the battery room ventilation adequacy. The team identified and the licensee agreed that some of the descriptions in the UFSAR and DBD's relating to battery room ventilation and operator actions following accident or off-normal conditions needed to be rewritten for clarity. The licensee initiated CR 97-0010 for this purpose.

The team identified that the licensee had installed white bezels on the control room instruments classified under Reg Guide 1.97 as type A, B, & C categories 1 & 2. The team found the licensee's identification scheme to be appropriate.

E1.2.5.2.3 Conclusions

The batteries were found to be adequate. Missing bolts found on the 125 VDC Load Test Panel 1A were replaced. The startup transformers and cable spreading rooms walkdowns did not identify any issues or open concerns. The licensee has not adequately responded to the team's concern regarding dirty filters on the hot-shutdown panels. This item is identified as INSPECTOR FOLLOW-UP ITEM 50-335/96-201-12.

The licensee has issued procedural changes to correct the different values stated in the battery surveillance and operating procedures. Control room indicator identification for RG 1.97 related instrument meets the intent of the requirement identified in the RG.

E1.3 Component Cooling Water System - Unit 2

E1.3.1 System Overview

The CCW system (as described in Unit 2 UFSAR Section 9.2.2 and Calculation # PSL-2EJM-82-081,CCW System Design Criteria dated 4/15/82) consists of two heat exchangers, three pumps, one surge tank, a chemical feed tank, and associated piping, valves and instrumentation. The system is arranged as two redundant essential supply headers (designated A and B) each with a pump and heat exchanger and the capability to supply the minimum safety feature requirements during plant shutdown or Design Basis Accident (DBA) conditions. A nonessential supply header (designated N) which is connected to both essential headers during normal operation is automatically isolated from both by valve closure on a safety injection actuation signal (SIAS). During normal operation, the nonessential header supplies cooling water to the following components: sample heat exchangers, boric acid concentrators, waste concentrator, waste gas compressors, letdown heat exchanger, control element drive mechanism air coolers, the reactor coolant pumps and motors, and the steam generator blowdown radiation monitoring.

The A and B headers serve the following components:

Header A	Header B
Shutdown Heat Exchanger 2A	Shutdown Heat Exchanger 2B
Containment Fan Coolers 2A/2B	Containment Fan Coolers 2C/2D
Control Room A/C 3A, 3B, 3C	Control Room A/C 3A, 3B, 3C
HPSI Pump 2A	HPSI Pump 2B
Fuel Pool Heat Exchangers 2A/2B	Fuel Pool Heat Exchangers 2A/2B

The A and B header systems are isolated from each other during accident conditions. Pump 2A serves header A and pump 2B serves header B. Pump 2C may be manually aligned with either header A or B by means of the cross-connection valving on the suction and discharge sides of the pumps.

Both the A and B supply header systems pump demineralized cooling water through the shell side of their respective component cooling water heat exchangers, through the components being cooled and back to their respective pumps. The tube side of each CCW heat exchanger is supplied with sea water from the Intake Cooling Water (ICW) system. The surge tank is connected to the suction side of the pumps and is designed to accommodate volumetric thermal expansion and contraction in the system and to maintain a static pressure head at each pump suction to insure that adequate NPSH is available for all pump operating conditions. Demineralized makeup water is added to the surge tank through an automatic water level control system by the demineralized water pump. Provisions are also made to supply makeup from the Fire Protection System. Although both essential headers share the surge tank, a baffle divides the lower portion of the tank into two separate compartments, each associated with one of the two essential headers. The cylindrical tank is 11 ft. long and is mounted horizontally at the 62 foot Floor Elevation in the Reactor Auxiliary Building. It has a 5.5 ft. diameter with a baffle height of 30 inches. Makeup water is added when the water level falls below 36 in. and a low water level alarm is initiated in the control room at 29 in. Makeup water is stopped at a surge tank water level of 48 in. and a high water level alarm is initiated in the control room at 54 in. Water level indication on the tank is provided on each side of the baffle.

Leakage of reactor coolant into the CCW system can be detected by an increasing water level in the surge tank and by radiation. A one gpm leak into the tank causes a high water level alarm in eight hours (based on an initial tank water level of 40 in. in the 66 in. diameter horizontal tank). A level switch and local gage glasses mounted on the surge tank provide control room high water level alarm and local indication of tank water level, respectively. A radiation monitor is provided in each of the redundant headers on the outlet side of the CCW heat exchangers. Should the radioactivity in the system rise above the setpoint, a high radiation alarm is actuated in the control room and the three way valve of the surge tank which

is normally vented to the atmosphere, is automatically repositioned. The system operates un-vented with relief to the Liquid Waste Management System for overpressure protection.

The CCW system's piping and valves are carbon steel. A chemical feed tank in the system permits addition of a corrosion inhibitor.

The design basis for the Component Cooling Water (CCW) system, as stated in the governing Design Basis Document DBD-CCW-2, Rev. 0, is summarized below.

E1.3.1.1 System Functions

E1.3.1.1.1 Safety Functions

a. The CCW system shall provide adequate cooling for those safety related components associated with containment and reactor decay heat removal and provide for control room cooling during accident conditions assuming a single failure coincident with loss of offsite power (LOOP).

b. The CCW system shall provide adequate cooling for those safety related components associated with achieving safe shutdown and provide for control room cooling assuming a single failure coincident with a loss of offsite power.

E1.3.1.1.2 Quality Functions (those functions that are not safety related but are important to safety)

a. During normal operation, the CCW system shall provide adequate cooling to those components important to safety such as the reactor coolant pumps, control room air conditioners, containment fan coolers, and the spent fuel pool heat exchanger.

b. The CCW system shall withstand safe shutdown earthquake loads, tornado loads or maximum flood levels without loss of safety function.

c. The CCW system shall provide adequate cooling for reactor auxiliary systems to achieve and maintain hot standby with the capability of bring the plant to cold shutdown during plant transients (including fire) with or without offsite power.

E1.3.1.1.3 Non-Safety Functions

a. The CCW system shall provide adequate cooling to auxiliary components to support normal plant operations.

b. The CCW system shall provide an intermediate barrier between the radioactive reactor coolant system (RCS) and the non-radioactive Intake Cooling Water (ICW) system.

E1.3.1.2 Conformance With Selected Design Criteria

E1.3.1.2.1 General Design Criteria

a. General Design Criterion 5 - Sharing of Structures Systems and Components

There is no sharing of CCW structures, systems or components between Units 1 and 2.

b. General Design Criterion 44 - Cooling Water

The single failure analysis of the CCW system is presented in UFSAR Section 9.2.2.3.2. There is no single failure that could prevent the CCW system from performing its safety function.

c. General Design Criterion 45 - Inspection of Cooling Water System

Inspection of the CCW system is presented in UFSAR Section 9.2.2.4.

d. General Design Criterion 46 - Testing of Cooling Water System

Testing of the CCW heat exchangers and pumps is presented in UFSAR Section 9.2.2.4.

E1.3.1.2.2 Regulatory/Licensing Requirements

a. Regulatory Guide 1.26

The CCW pumps, the suction and discharge header A and B piping and valves, and the CCW heat exchangers are designed to Quality Group C requirements.

b. Regulatory Guide 1.29

The CCW pumps, the suction and discharge header A and B piping and valves, and the CCW heat exchangers are designed to Seismic Category 1 requirements.

c. Regulatory Guide 1.102

CCW system equipment susceptible to flood damage is protected by locating all safety related components above the maximum expected water level and wave run-up during probable maximum hurricane.

d. Regulatory Guide 1.117

The CCW system is protected from tornado winds and missiles by the Component Cooling Water Structure and the Reactor Auxiliary Building.

E1.3.2 Mechanical Design

E1.3.2.1 Cooling Capacity

E1.3.2.1.1 Scope of Review

Determine if the CCW system is capable of providing sufficient cooling capacity to cool reactor coolant auxiliary systems components during normal operation, normal plant shutdown, emergency shutdown and postulated Design Basis Accidents (DBA).

E1.3.2.1.2 Inspection Findings

The CCW heat exchangers and pumps were sized based on the most limiting circumstances of the required modes under which the system would be operated, with the requirement that one heat exchanger and one pump be sufficient to meet accident conditions. These modes were defined as normal operating (power operations, two pumps/two heat exchangers), normal shutdown (plant cooldown, two pumps/two heat exchangers), emergency shutdown (plant cooldown, two pumps/one heat exchanger or one pump/two heat exchangers) and accident (emergency loads only, one pump/one heat exchanger). The design flow rates and heat loads for the various modes of CCW operation are summarized in Table 9.2-5 of the UFSAR (attached) which is based on Calculation # NSSS-010 Rev. 0, CCW Heat Loads dated June 1981. This calculation states that the Unit 2 CCW system was initially a duplicate of the Unit 1 CCW system but the flow rates and heat loads were subsequently revised to agree with the purchase specifications for the various components served by the CCW system.

For equipment sizing purposes, the maximum post accident supply temperature to the components cooled by CCW system was originally assumed to be 120°F. This is consistent with the supply temperature used for the Unit 1 CCW system, and is reflected in the values in Table 9.2.5. However, this maximum post accident supply temperature was later changed to 108°F based on the limitations of the Control Room Air Conditioning system. In the early 1990's, during preparation for the NRC Service Water System Inspection, FPL discovered that the Control Room Air Conditioning units had been purchased by a specification that erroneously listed the maximum post accident supply temperature as 100°F. Subsequent discussions with the vendor indicated that the Control Room A/C units would not work with a supply temperature of 120°F and that the best that could be expected was near full capacity operation at 108°F (see CVI Incorporated letter dated 9/23/91 St. Lucie 2 Control Room A/C Use of 120°F Cooling Water). This requirement to limit the CCW post accident supply temperature to 108°F led FPL to completely revise the Unit 2 Containment accident analysis in 1993 (Calculation # 016-AS93-C-006 St. Lucie 2 LOCA Containment Peak Pressure/Temperature Analysis at 102% Power dated 3/17/93).

The 1993 analysis specifically coupled the CCW and Intake Cooling Water (ICW) systems into the containment accident model and allowed a time dependent determination of CCW supply temperature (limited to 108°F), as well as a time dependent determination of heat transfer rate across the CCW heat exchanger.

TABLE 9.2-5

DESIGN FLOW RATES AND HEAT LOADS FOR ALL AUXILIARY EQUIPMENT COOLED BY THE COMPONENT COOLING SYSTEM

EQUIPMENT		ACCIDENT			NORMAL OPERATION			EMERGENCY SHUTDOWN	
REFUELING (No. / Units) Duty BTU/Hr)	Total Flow GPM	No. Operat. Oper.	Duty/Train No. (10^6 BTU/Hr) (10^6 BTU/Hr)	Total Flow Total Duty GPM GPM	No. Total Flow Oper.	Total Duty (10^6 BTU/Hr)	Total Flow GPM	No. Oper.	Total (10^6
1. Fuel Pool Hx (2)		0	-	-	1	12.5	3,560	1	
12.5	3,560	1	12.5	3,560					
2. Shutdown Hx (2)		1	87	4,820	-	-	-	1	
130(2)	4,820	1	29	4,820					
3. Letdown Hx (1)		0	-	-	1	21.0	1,200	1	
2.9(1)	190(1)	0	-	-					
4. Sample Hx (4)		0	-	-	3	2.08	140	2	
1.04	70	2	1.04	70					
5. HPSI Pumps (2)		1	-	35	0	-	70	0	
70	0	-	-						
6. RCP Motors (4)		0	-	-	4	6.42	848	2	
3.3(1)	848(1)	0	-	-					
7. CEDM (3)		0	-	-	3	4.4	520	0	-
520		0	-	-					
8. Waste Concen. (1)		0	-	-	1	12.7	710	1	--
1		12.7	710						
9. B. A. Concen. (2)		0	-	-	1	12.7	1,420	2	--
1		12.7	710						
10. Waste Gas Comp. (2)		0	-	-	1	-	2	0	-----
11. Blowdown Rad. Monit. (2)		0	-	-	2	-	10	0	-10-
12. Cont. Coolers (4)		2	.124	2,400	4	4.32	4,800	4	
4.32		4,800	4	4.3	4,800				
13. Control Room A/C (3)		2	0.66	160	2	.75	240	2	
.75	240	2	.75	240					

NOTES:
(1) During shutdown the letdown Hx and RC Pump Motor Loads are not concurrent with other loads.
(2) The maximum heat load for shutdown cooling Hx is 1.30 x 10^6 BTU/HR which reduces gradually to 29 x 10^6 BTU/HR.

These values reflect original procurement values. Refer to Ref. 27 in Section 6.2

The analysis resulted in a peak heat transfer rate across the CCW heat exchanger of 132.4×10^6 Btu/hr at a peak CCW supply temperature of 107.84°F. This peak post accident heat transfer rate, consistent with the CCW supply temperature less than 108°F, was used to calculate ICW performance curves as described later. Since the 1993 analysis dealt with the accident scenario, only the accident mode shown in Table 9.2.5 has been affected and all other modes have not changed. The heat loads contained in the current version of Table 9.2.5 are not correct as they have not been updated to reflect the 1993 accident analysis.

While preparing for this inspection, FPL noted a discrepancy between an operating procedure for the CCW system, and the CCW DBD and the Safety Evaluation for the Updated LOCA Containment Analysis (Calc. #JPN-PSL-SENP-93-018 Rev. 0 dated 4/29/93). Operating Procedure OP 2-0310020 incorrectly identifies the maximum CCW supply temperature as 120°F, while this design documentation limits the maximum supply temperature to 108°F. CR 96-2701 was issued on 11/1/96 to disposition this discrepancy. Specific corrective actions of this CR are to: 1) issue a PMAI to Operations to update OP 2-0310020 to identify the maximum CCW supply temperature as 108°F (due 3/30/97), and 2) issue a PMAI to Engineering to evaluate the maximum heat load placed on the CCW system while the plant cools from Hot Standby to Cold shutdown (due 1/31/97).

Based on the Containment accident analysis done in 1993, performance curves were generated for the ICW system in January 1996. These performance curves, the final output of two separate calculations (#PSL-2FJM-96-001 Rev. 0 and #PSL-2FJM-96-002 Rev. 0), plot ICW inlet (supply) temperature vs. pressure drop across the CCW heat exchanger for various percent of tubes plugged and for a 0 to 25% degraded ICW pump. The curves are plotted for two specific ICW flows and assume the maximum post accident heat transfer rate across the CCW heat exchanger of 132.4×10^6 Btu/hr, a CCW supply temperature of 108°F, and a conservative value of heat transfer coefficient, U = 300 Btu/hr ft^2 °F, based on test data for end of cycle heat exchanger tube fouling. <u>The purpose of these curves is to allow Operations to determine the maximum allowable sea water temperature that can be tolerated in order to keep the CCW supply temperature to less than 108°F in an accident situation for various degrees of fouling of both the ICW pumps and the CCW heat exchangers</u>. These curves can be used to limit unit operation if the sea water temperature gets too high, depending on the degree of fouling of the ICW pumps and the CCW heat exchangers, and were generated because of the importance of being able to keep CCW supply temperature to less than 108°F in an accident situation.

Fouling of the ICW/CCW systems is an important consideration for FPL. Sea water is supplied to the tube side of the CCW heat exchangers by the ICW pumps. This water is heavily laden with shells and silt that can cause pump degradation and heat exchanger tube fouling (increased pressure drop across the tubes). In order to control this problem, the CCW heat exchanger tubes are cleaned at every outage as well as mid-cycle as witnessed by the inspection team during a plant walkdown. This fouling problem is addressed in the calculation of the performance curves by parameterization on the percent of degradation of ICW pump performance (percent loss of flow) and by using a

CCW heat exchanger heat transfer coefficient that is typical of end of cycle fouled conditions for all cases considered (see 2A & 2B CCW Heat Exchanger Tests of 10/02/90 Performance Calculations and Results).

While preparing for this inspection, FPL determined that the fouling factors used in the 1993 Containment accident analysis for the shutdown cooling heat exchanger and the containment fan coolers may not be bounding for determining the maximum post-LOCA CCW temperatures. CR 96-2716 was issued on 10/31/96 to disposition this problem. The shutdown cooling heat exchanger and containment fan coolers remove heat from the Containment post accident. This heat is input to the CCW system which then transfers it through the CCW heat exchanger to the ICW system. The 1993 Containment accident analysis assumed that the shutdown cooling heat exchanger and the containment fan coolers were fouled per vendor recommendations and this limited their ability to remove heat from the Containment. This is a conservative assumption for maximizing containment temperature but is not conservative for CCW system performance. The greater the heat removed from the Containment the more difficult it becomes to maintain the CCW supply temperature below 108°F. The re-evaluation performed as part of CR 96-2716 considered both components to be clean and computed a maximum post accident value for heat transfer rate into the CCW system of 173.8×10^6 Btu/hr (compared to 132.4×10^6 Btu/hr computed in the 1993 Containment accident analysis). This heat must be dissipated through the CCW heat exchanger in order to maintain the CCW supply temperature below the desired limit. Using this new maximum post accident heat load, two new performance curves were generated for the ICW system assuming current values for tubes plugged and pump degradation. These new curves show that the ICW inlet (supply) temperature must be below 83°F for some conditions of CCW heat exchanger tube pressure drop in order for the CCW system to maintain its post accident temperature limit. This temperature is below the 86 to 87°F limit from the previous performance curves and is a value that ICW temperature could reach in May or late next summer.

E1.3.2.1.3 Conclusion

Based on the accident analysis done as part of CR 96-2716, the new system performance curves will be used by Operations until a more detailed analysis is performed and new curves provided. There are six Specific Corrective Actions listed in this Condition Report. These involve the determination of appropriate fouling factors for the shutdown cooling heat exchanger and containment fan coolers to be used in a revised Containment accident analysis and ultimately the development and issuance of new performance curves to Operations by 3/31/97. Design Engineering will also perform an operability/reportability evaluation of the use of the original ICW performance curves by 4/10/97. This is a unique situation where the shutdown cooling heat exchanger and the containment fan coolers can operate too efficiently; that is, if these components are too clean they can remove too much heat from the Containment post accident. The transfer of this maximum post accident heat load into the CCW system and across a fouled CCW heat exchanger results in a large temperature differential between the ICW and CCW system. As a result, a lower sea water supply temperature (ICW temperature) is necessary to keep the CCW system within its design temperature limit. This need for a low sea water supply temperature could ultimately limit the

operation of the unit during certain portions of the year when the sea water temperature can be high. The alternative would be to limit the fouling of the CCW heat exchanger by going to more frequent cleaning.

There are no short term operability concerns for the Unit 2 ICW system since the ICW inlet temperature is expected to remain below 83°F until May. A revised containment accident analysis and new performance curves are scheduled to be completed and issued by 3/31/97 as part of the corrective actions committed to in CR 96-2716. These calculations, to be performed by Design Engineering, are identified as INSPECTOR FOLLOW-UP ITEM #50-389/96-201-01. The licensee's failure to keep UFSAR Figure 9.2.5 up to date as required by 10CFR50.71(e) is identified as UNRESOLVED ITEM 50-335 and 389/201-01.

E1.3.2.2 Operational Performance Curves

E1.3.2.2.1 Scope of Review

Review the use of the performance curves by Operations to determine ICW operability.

E1.3.2.2.2 Inspection Findings

The current performance curves generated as a result of CR 96-2716 are in the Plant Curve Book in the Control Room. Direction on the use of these performance curves is provided by a "night order" titled "Instructions for the use of U#2 CCW HX Performance Curves." The night order governing the use of these latest performance curves was first issued the week of 11/7/96 and has been updated on a monthly basis. The night order provides specific instructions to be followed upon exceeding or approaching an ICW temperature of 82°F and also provides the current status of the CCW heat exchangers and the ICW pumps so that Operations will know which performance curves to use. Operators are directed to contact System Engineering to run actual performance data as necessary since the night order provides information on the heat exchangers and pumps that can be a month out of date. As a night order is only a temporary method of providing this information to operators, a PMAI has been issued to incorporate the performance curves into the appropriate operating procedure. The new performance curves to be generated by 3/31/97 will replace the existing performance curves in this operating procedure.

Operating Procedure No. OP-2-0010125 , Revision 6 (Schedule of Periodic Tests, Checks and Calibrations) is used to calculate the pressure drop across the CCW heat exchangers every shift and, using the appropriate performance curve from the Plant Curve Book, ultimately determine operability of the ICW system (see Operating Procedure No. OP-2-0010125A, Revision 9). This procedure gives operators the latest information on the status of the CCW heat exchanger in order to determine operability of the system as the temperature of the sea water increases toward the limits of the performance curves. The current version of this procedure requires this information to be taken every shift "... if the ICW Inlet Temp. to CCW Hx is greater than 85°F." This value of 85°F is non-conservative as the latest performance curves generated as a result of CR 96-2716 indicate that operability could be challenged at temperatures as low as 83°F.

E1.3.2.2.3 Conclusion

FPL has issued a PMAI to incorporate the performance curves into the appropriate operating procedure and therefore eliminate the night order that provides direction to Operations on the use of these curves. Operating Procedure No. OP-2-0010125A incorrectly references a value of 85 °F as the temperature limit above which CCW heat exchanger pressure drop information is determined on a shiftly basis. The team identified this issue as Inspector Follow-up Item #50-389/96-201-02.

E1.3.2.3 System Balancing

E1.3.2.3.1 Scope of Review

Review system balancing to insure that each load served by the CCW system receives the required flow rate for all applicable modes of operation.

E1.3.2.3.2 Inspection Findings

Balancing of the CCW system is accomplished through Operating Procedure No. OP 2-0310020 (Component Cooling Water-Normal Operation) to insure that each load served by the CCW system receives the required flow rate for all applicable modes of operation. The major loads served by the CCW system in an accident situation are the containment fan coolers and the shutdown cooling heat exchangers. The flow requirements for the containment fan coolers are in the Technical Specifications (Containment Systems, Section 3/4.6.2) as "...a cooling water flow rate of greater than or equal to 1200 gpm to each cooling unit." The flow requirements for the shutdown cooling heat exchangers are not in the Technical Specifications.

Appendix G (Essential CCW Load Flow Balance) to procedure OP 2-0310020 is used to balance CCW flow rates to the containment fan coolers and shutdown cooling heat exchanger to ensure proper post accident flow rates. This procedure is performed when shutdown cooling is not in service and is used to verify "licensing basis" flows to the essential loads. This appendix specifies that containment fan cooler flow rates "... are greater than or equal to 1200 gpm." The team noted that procedure did not include any margin from the technical specification limit and that it did not envelope the low flow setpoint which is set at 1250 gpm.

E1.3.2.3.3 Conclusion

The procedure used by FPL to balance the CCW flow rates appeared to be adequate. FPL has issued PMAI 96-12-203 to address the team's concern regarding a lack of margin in the acceptance criteria for the CCW flow to the containment fan coolers.

E1.3.2.4 Net Positive Suction Head

E1.3.2.4.1 Scope of Review

Verify the acceptability of the CCW Pump Net Positive Suction Head (NPSH) and system leakage

E1.3.2.4.2 Inspection Findings

As part of the system design review, the inspection team requested the NPSH calculations for the CCW pumps. FPL chose to perform a new NPSH calculation for these pumps, Calculation #PSL-2FSM-96-017, instead of providing any original design calculations for this parameter. The available NPSH was calculated for two points on the pump curve: the design flow of 8500 gpm, and the flow limit allowed by procedure of 10,800 gpm (slightly less than pump run out of 11,100 gpm). The results were as follows:

Design Flow Conditions (8500 gpm): $NPSH_A$=68.5 ft $NPSH_R$=44.0 ft
Procedural Limit Conditions (10,800 gpm): $NPSH_A$=63.7 ft $NPSH_R$=64.0 ft

These results show that there is sufficient NPSH for the CCW pumps during normal operation at the design flow. The small discrepancy between the available and required NPSH at the procedural limit flow is due to the conservatism built into the calculation which assumes an inlet flow temperature of 145°F instead of a more realistic 108°F). At the system design limit of 108°F there is adequate $NPSH_A$ at this flow. Calculation #PSL-2FSM-96-017 also indicated that adequate NPSH would be available at the accident flow condition of 7415 gpm @ 143.8°F.

The inspection team investigated the performance of the CCW pumps in the event of a postulated leak in the "N" header (the portion of the CCW system that serves non-safety related loads). In preparation for this inspection, FPL had generated CR 96-2810 on the same topic. The calculations for $NPSH_A$ shown above assume a water level in the surge tank of three feet. If a leak occurs in the "N" header, the water level in the surge tank will drop and the CCW pumps will tend to "run out" on their curves (because of decreased system resistance). This will lead to a situation where the $NPSH_A$ is decreasing whereas the $NPSH_R$ is increasing. If this postulated leak is not isolated in time, the water level in the surge tank could drop to the bottom of the tank and the pump could be operating near the run out point on its curve. In this situation, referencing the results in the above paragraph for procedural limit conditions, the available NPSH could be as much as three feet less than the required NPSH.

As a part of CR 96-2810, FPL has established that there is no credible leak in the "N" header that could create this operating situation in the CCW system. The valves that are required to close to isolate a leak in the "N" header will close in less than 9 seconds. This closure time is administratively required (see Administrative Procedure 2-0010125A) and is periodically verified by testing in accordance with ASME Section XI. A three foot level in the surge tank corresponds to a volume of 460 gallons in each baffled section of the tank. For this quantity of water to leak from the "N" header in 9 seconds

would require a leakage rate of 3067 gpm. The CCW system is a moderate energy system and applying this criteria for determining pipe cracks to the largest pipe in the "N" header (see UFSAR Section 9.2.2.3.2) yields a crack that would allow a leakage rate of approximately 410 gpm. This is 13% of the 3067 gpm leakage rate that is required to create a hypothetical NPSH problem for the CCW pumps. Therefore in the 9 seconds required to isolate a leak in the "N" header, the water level in the surge tank would drop just a few inches and not reach the bottom of the tank. This would result in only a small possible difference between the required and available NPSH that would exist for just a few seconds before break isolation. Finally, FPL has presented the original pump test curves performed by the vendor that show that the CCW pumps ran for approximately three minutes with insufficient NPSH and experienced no problems.

E1.3.2.4.3 Conclusion

Adequate NPSH is available to the CCW pumps for all accident and operating conditions. FPL generated CR 96-2810 to document the design of the "N" header isolation feature. In addition, CCW pump NPSH Calculation #PSL-2FSM-96-017 will be revised as necessary taking into consideration the maximum licensing basis leakage number.

E1.3.2.5 Maintenance History

E1.3.2.5.1 Scope of Review

Review maintenance history of essential components in system

E1.3.2.5.2 Inspection Findings

The inspection team requested the maintenance history of the CCW pumps and motors. This history showed that the 2B pump had an impeller replacement (in February 1993) and that the motors of all three pumps had been overhauled a total of five times. The 1B CCW pump also had its impeller replaced in 1993 and the inspection team was provided with a detailed memo documenting the investigation and examination of this pump (see memo dated 10/4/93, subject: CCW PP 1B Overhaul Inspection and Root Cause Report). Much of the information contained in this memo regarding the Unit 1 pump also applies to the Unit 2 pump. The impellers of both pumps showed a significant amount of cavitation erosion. The root cause was determined to be cavitation due to flow instabilities caused by the piping configuration design. Ideally, the inlet piping to a centrifugal pump should be a straight run of pipe between 3 and 8 suction pipe diameters long prior to flow entering the pump. Both the Unit 1 and Unit 2 B CCW pumps have an elbow connected directly to the pump casing resulting in flow instabilities and ultimately the observed impeller erosion. The pump vendor (Bingham) was contacted as to a material change but since the impellers had run times of over 15 years, no material change was recommended.

E1.3.2.5.3 Conclusion

The inspection team had no concerns with the maintenance performed on the CCW system.

E1.3.3 Electrical Design

The team reviewed electrical design and licensing bases documents including calculations, specifications, vendor manuals, the Updated Final Safety Analysis Report (UFSAR), the Safety Evaluation Report (SER), and Technical Specifications.

E1.3.3.1 Batteries

E1.3.3.1.1 Scope of Review

Determine that the existing design of the safety related batteries meets the design basis requirements.

E1.3.3.1.2 Inspection Findings

Load profile calculation, PSL-2-FJE-90-016 "Safety Related Batteries 2A and 2B" Rev.0 dated 1/10/91, was reviewed for battery sizing. The team determined that the battery sizing calculation was performed in accordance with standard industry practice and followed IEEE Standard 485-1983. The calculation considered the lowest cell temperature as 50°F and utilized a factor of 1.25 to account for battery aging effects. A design margin of 16% was also provided for the batteries which exceeds the 10-15% recommended by IEEE 485-1983. The design calculation used vendor provided capacity curves that were based upon a fully charged cell with a nominal fully charged electrolyte specific gravity of 1.215 +/- 0.010. Overall, the calculations support the system functions and design bases for the DC power requirements.

Unlike the design calculation, the surveillance requirements use a specific gravity acceptance criteria of 1.195 or 1.190, as stated in the Technical Specification. The team identified that meeting this technical specification would not necessarily ensure that the batteries could perform to their calculated design capacity. The Technical Specification acceptance criteria of 1.195 or 1.190 does not envelope the specific gravity of 1.215 +/- 0.010 used in the design calculation. Upon further investigation, the team learned that the technical specification requirement is based on standard industry numbers and is not necessarily intended to demonstrate design capability of the batteries. During battery surveillances, not one, but a number of battery parameters are measured. The assessment of battery operability is based on all the collective data.

E1.3.3.1.3 Conclusion

A review of the related calculations and design documentation indicates that the batteries meet their design bases.

E1.3.3.2 System Voltage

E1.3.3.2.1 Scope of Review

Review the design adequacy of station electric distribution system voltages.

E1.3.3.2.2 Inspection Findings

In addition to the review conducted for Unit 1 and detailed in paragraph E1.2.3.2.2 where team concerns regarding the acceptance criteria for ensuring the operability of the diesel voltage regulator were raised, a review was performed of Calculations 2998-A-452; Calculation PSL-2-FEPSTR-1991-0102 and 0103; and the Minimum Excitation Limiter setting calculation for the main generator exciters and the main generator volts/hertz relays. Based on these additional reviews the team concluded that the relay settings are within the design envelop and provide adequate protection.

E1.3.3.2.3 Conclusion

As stated in paragraph E1.2.3.2.3 the AC buses are adequately sized for the connected loads and short-circuit duties. The voltage ratings are also adequate for the applications. As stated in paragraph E1.2.3.2.2, the Technical Specifications acceptance criteria of 4160 +/- 420 volts appears to be too wide of a range to support the design. As stated in paragraph E1.2.3.2.3, St. Lucie initiated appropriate corrective actions to address this concern.

E1.3.3.3 Cable Sizing

E1.3.3.3.1 Scope of Review

Determine if cables are adequately sized for the equipment ratings and the short circuit duties.

E1.3.3.3.2 Inspection Findings

Short circuit calculation EC-039 for the 125VDC 1A and 1B batteries was reviewed to determine if the short circuit currents on the DC system are within the switchgear, breaker and cable ratings. The team determined the design methodology and calculation assumptions were acceptable. The calculation results indicated the design requirements were met for the switchgear, breakers and cables. Overall, this calculation was thorough and comprehensive.

A design basis temperature of 115°F inside containment, 104°F for temperature outside containment in adjacent buildings and site ambient temperature outside containment of 93°F dry bulb (for 99.7% of the time) and 101 degrees F (for 0.3% or 30 hours)(EQ Doc package 2998-A-451-1000) was considered in sizing cables. The team determined this to be conservative as the actual inside containment temperature recorded indicates a maximum normal operating temperature occurring during August of 106°F and during September of 107°F. The actual outside containment temperatures recorded indicate a seasonal high of about 91 degrees F.

E1.3.3.3.3 Conclusion

The DC cable sizing was determined to be adequate. The AC cables do not have a specific calculation that analyzes temperature rise versus short circuit duty like the DC system. However, all related documentation such as the original cable criteria in WHL-8 indicates the cables are adequately sized for the equipment ratings and the present short circuit duties.

E1.3.3.4 MCC Fuse Sizing

E1.3.3.4.1 Scope of Review

Determine that the fuses in the MCCs are adequately sized for their application.

E1.3.3.4.2 Inspection Findings

The basis for sizing the fuses is contained in EBASCO Unit 2 Motor Control Centers' Specification 2-2998-286. This specification was similar to the Unit 1 specification 8770-286 which was reviewed in section E1.2.3.4.2 of the report. The team determined that the designated fuses were consistent with the specification.

E1.3.3.4.3 Conclusion

Control circuit fuses were reviewed for the MCC's. The sizing was found to be acceptable and in accordance with the design criteria.

E1.3.3.5 CCW Pump Motor Sizing

E1.3.3.5.1 Scope of Review

Verify that the CCW pump motors will provide adequate torque to drive the CCW feed pumps and start on reduced system voltage.

E1.3.3.5.2 Inspection Findings

During the inspection, the licensee was asked to specifically identify any of the CCW motors that had been rebuilt or overhauled. The response identified that motor 1A was overhauled in 1990 and 1993, motor 1B overhauled in 1988 and 1991, motor 1C completely rebuilt in 1991, 2A overhauled in 1988 and 1990, 2B overhauled in 1989 and 1992, and 2C overhauled in 1990. The team reviewed the data pertaining to the rebuild of the 1C pump and determined that the motor characteristics were not modified and still match the pump torque requirements. The rebuilt motor was manufactured to the original specifications and tested for reduced voltage starting, etc.

Both CCW pump/motor sets were procured as capable of starting and running with 75% of rated voltage. Calculations were required to be performed as part of the contract(s). Although the original calculations were not provided, the following drawings were presented as evidence that calculations had been performed:

- Motor data sheet (2998-2387)
- Pump motor speed-torque curve(2998-5987).

The curves were reviewed to ensure that the motors are capable of starting the pumps under all the required operating modes including reduced voltage. The team concluded from these data sheets that the motors are matched to the pumps for torque requirements and are capable of starting and running the pumps under all operating conditions.

Calculation PSL-2-FJE-90-0020 was reviewed to verify the 23 second start requirement for the CCW motors. The calculation has a sequencer time of 6 seconds on these motors. The diesel generator start time is less than or equal to 10 seconds. The motor spin up time at 75% voltage is about 6.5 seconds. Therefore, the design appears to be satisfactory to meet the start time requirements.

The team also evaluated the protective relay setpoints for the motors, including the system voltage at just above the PSB-I degraded grid voltage relay setpoints. At this voltage, the team verified that running motors will not trip out on overcurrent or incur any thermal damage. For the 4000 volt motors, motor overload protection is basically inactive until 150% of the full load current is reached, due to the characteristics of the COM-5 protective relays. Since the PSB-1 setpoints are above 90% of the bus voltage, the increased motor' currents may be in a range of 110% of full load current, which is below the protective area of the COM-5 relays. Since all the motors were procured with a 1.15 continuous service factor, there would be no detrimental effect due to the decreased voltage. The overload protective trip setting is at 250% of full load current. The alarm is set at 150% of full load current.

For an operation below the PSB-1 degraded voltage relay Technical Specifications minimum of 3831 volts, the current is still reasonably inversely proportional to voltage (constant kVA). The Technical Specifications degraded voltage setpoint is greater than 92% at the 4160 volt level. Therefore, the current could be in a range approaching 140% of rated for a maximum duration (relay time delay) of 20 seconds. Per the typical motor safe stall time curve, the safe time at 140% current is in the 800 second range. Therefore, the degraded grid voltage relaying will trip the unit off the line long before any thermal damage occurs. Below the 70% setpoint, the motor will stall with the current reaching 70% of locked rotor value, for which the safe time is in the 2.5 seconds range, whereas the maximum tripping delay of the loss of voltage relaying is 1.5 seconds.

E1.3.3.5.3 Conclusion

The motor relay setpoints are acceptable and envelope the motors' design requirements. Motor sizing is in accordance with the design requirements. Motor/Pump torque curves are matched and capable of starting for reduced voltage scenarios.

E1.3.4 I&C Attributes of the CCW System

E1.3.4.1 CCW Surge Tank Level Control and Alarm Switches

E1.3.4.1.1 Scope of Review

The team performed a review to verify the adequacy of the set points for the CCW surge tank water level control and alarm switches.

E1.3.4.1.2 Inspection Findings

A common surge tank has been used for both essential headers to accommodate expansion & contraction of the process fluid and maintain required NPSH for the CCW pumps. The lower portion of the tank is divided into two separate compartments by a 30" baffle, creating dedicated inventory for each essential header. Each section of the tank is provided with:

- A low level switch (LS-14-1A/ 1B) to alarm in the control room on low water level condition to alert the operator to a loss of make-up and decrease in available CCW pump suction head.
- A low-low level switch (LS-14-6A/6B) to isolate the non-essential CCW header (HCV-14-8A/8B/9/10) from the appropriate essential CCW header.

Level make-up control switches (LS-14-3/4) control the tank level by opening and closing valve LCV-14-1. A high level switch common to both sections (LS-14-5) has been provided to confirm RCS in-leakage or identify a loss of make-up control. The purpose of the alarm from the switch is to warn the control room with respect to potential overflow. This alarm is one of the two variables selected (in response to RG 1.45 requirements) to monitor RC leak detection in the heat exchangers (UFSAR table 5.2-14).

The team determined the setpoints in Calculation No PSL-2FJI-92-012, "Surge Tank Level Setpoints," are in agreement with the instrument list for CCW system. The setpoints are:

Switch	Set Point
Low-Level Alarm (LS-14-1A/1B)	29"
Low-low level Auto Isolate (LS-14-6A/6B)	29"

Normal Make-Up	
Start (LS-14-3)	36"
Stop (LS-14-4)	48"
Hi level alarm (LS-14-5)	54"

E1.3.4.1.3 Conclusion

The setpoint values contained in Calculation PSL-2FJI-92-012 are acceptable and were in agreement with the design documents.

E1.3.4.2 CCW Radiation Monitor Setpoints

E1.3.4.2.1 Scope of Review

The team performed a review to verify the adequacy of the setpoints for the CCW system radiation monitors.

E1.3.4.2.2 Inspection Findings

The purpose of the radiation monitors is to detect leakage of radioactive water into the normally non-radioactive CCW system. Per the CCW flow diagram and UFSAR Section 11.5.2.2.1, the monitors sample the water supply downstream of each of the CCW system heat exchangers and return the sample to the CCW pump suction. This variable is provided in response to RG 1.45 (UFSAR Table 5.2-14). Upon detection of the radiation levels above a preset setpoint, an alarm is initiated in the control room, the atmospheric vent valve for the CCW surge tank is closed, and the CCW system then operates un-vented with relief to the Liquid Waste Management System. UFSAR Table 11.5-1 provides a typical alarm and control setpoint value. Per UFSAR Section 11.5.2.2.b, this value should be interpreted as a theoretical preliminary value.

Prior to the inspection, the licensee had identified in a CR (CR 96-2228) that setpoint calculations had not been performed for the containment radiation monitors. Likewise, the team identified that there are no formal calculations validating the UFSAR specified setpoints for CCW radiation monitors. In response to CR 96-2228, PMAI No. 96-10-191 was written for engineering to review the completeness of the design basis and associated documentation for the radiation monitors that are part of licensing basis of the plant to determine if further action is required. All radiation monitors in the UFSAR Table 11.5-1, including the CCW radiation monitors were stated for review as part of the PMAI, since UFSAR Table 11.5-1 is a part of the licensing basis of the plant.

Also during the above review, the team identified a discrepancy in the CCW radiation instrument range between UFSAR Table 5.2-14 & Table 11.5-1. Table 5.2-14 shows a range of 10^{-6} to 10^{-1} μCi/cc whereas Table 11.5-1 shows it as 10^{-7} to 10^{-2}. The licensee has written an UFSAR update package to revise Table 5.2-14 to match Table 11.5-1.

E1.3.4.2.3 Conclusions

The team identified that there are no formal calculations supporting the setpoints for the CCW radiation monitors. As a result, the licensee issued PMAI No. 96-10-191 to perform setpoint calculations for all monitors listed in Table 11.5-1. This item is identified as INSPECTOR FOLLOW-UP ITEM #50-389/96-201-03.

The team identified a range discrepancy for CCW radiation monitor between Tables 5.2-14 and 11.5-1 of the UFSAR. The licensee has written an UFSAR update package to rectify this discrepancy:

E1.3.4.3 CCW Heat Exchanger Alarm Setpoints

E1.3.4.3.1 Scope of Review

The team performed a review to verify the adequacy of the CCW heat exchanger 2A/2B outlet high temperature (shell side) alarm.

E1.3.4.3.2 Inspection Findings

The CCW heat exchanger outlet temperature is defined as a RG 1.97, Category 2, Type D variable. The required range per this RG is 40° to 200° F. Per the instrument list for the CCW system, UFSAR Table 9.2-7, and the RG 1.97 parameter summary list, temperature sensors TE-14-3A/3B and temperature recorders TR-25-2A/2B are provided to perform this function. The instrument ranges are 0°- 300° F. The high outlet temperature is alarmed in the Control Room, the Technical Support Center, and in the Emergency Response Facility in accordance with the NUREG 0696 requirements.

Per the Total Equipment Data Base (TEDB) for the temperature recorders TR-25-2A/2B, the alarm setpoint is 150°F on increasing temperature. The setpoint has been provided to alert the operator of a developing loss of heat sink condition as described by the unit off normal operating procedures.

E1.3.4.3.3 Conclusions

The CCW heat exchanger outlet temperature (shell side) indication meets the intent of RG 1.97, D2 variable requirements.

E1.3.4.4 Heat Exchanger Header Flow Alarms

E1.3.4.4.1 Scope of Review

The team performed a review to verify the adequacy of the high and low CCW heat exchanger header flow alarms in the Control Room.

E1.3.4.4.2 Inspection Findings

The CCW heat exchanger flow (shell side) indication is defined as a RG 1.97, Category 2, Type D variable. The required range per this RG is 0-110 % of the design flow (design flow is the maximum flow anticipated in the normal operation). Flow transmitters FT-14-1A/1B and flow indicating switches FIS-14-1A/1B are provided in response to RG 1.97. The instrument range is 0 - 15,000 GPM. The flow through the shell side of the CCW Hx ranges from 7415 to 14,400 GPM with the required accident flow being 7415 GPM. The HI/LO alarm setpoints are 9500 gpm and 4000 gpm respectively.

E1.3.4.4.3 Conclusion

The CCW heat exchanger flow indication meets the intent of RG 1.97, D2 variable requirements.

E1.3.4.5 Shutdown Heat Exchanger Flow Indication

E1.3.4.5.1 Scope of Review

The team performed a review to verify that the CCW flow indication from the shutdown heat exchanger is sufficient to monitor the operation of the heat exchanger.

E1.3.4.5.2 Inspection Findings

The instruments for measuring CCW flow indication from the shutdown heat exchangers are flow sensors FE-14-10A/10B, flow transmitters FT-14-10A/10B, square root extractors FF-14-10A/10B, and flow indicating switches FIS-14-10A/10B. Calculation No PSL-2FJI-92-010 states the maximum flow as being 5061 gpm and the normal flow as being between 3900 to 4900 gpm. As stated in this calculation and in the instrument list for the system, the HI/LO setpoints are 4915 gpm and 3850 gpm, respectively.

E1.3.4.5.3 Conclusion

Design documents showing CCW flow indication from the shutdown heat exchanger are in conformance with each other. The high and low setpoints are acceptable.

E1.3.5 CCW System Walkdown Observations

E1.3.5.1 Mechanical

E1.3.5.1.1 Inspection Scope

The team performed a walkdown of the Unit 2 Intake Water Cooling (ICW) System and the Component Cooling Water System (CCW). The initial walkdown was conducted on 11/21/96. Subsequent walkdowns and operation observations were conducted throughout the team's site visits.

E1.3.5.1.2 Observations and Findings

The overall appearance of the areas observed was good considering the intensive CCW/ICW "B" heat exchanger water box cleaning operation which was in progress. The licensee indicated that the operating heat exchanger "A" had been similarly cleaned prior to the current effort on the "B" assembly. It was explained that the over-design capacity of the heat exchangers, allowed for a greater amount of ICW side buildup of silt, shells and debris. This continuous deposition of debris requires cleaning during refueling outages and one approximately mid-cycle (the current effort). The System Engineers also identified that periodic replacements of the Temperature Control Valve ICW flow regulators, are required as a preventative measure due to the continuous erosion wear on the internals of the components and the near closed position of the control valves during normal operations.

Both of the heat exchangers had been re-tubed at a recent refueling outage. There has not been a problem with the HX's operating close to design requirements, due to the large margin in the design heat capacity of the units. Several tubes are plugged in each heat exchanger due to problems in the alignment during the last re-tubing operation or other damage in the tube sheets which has prevented the installation of new tubes. There are no concerns for capacity due to over-design of the coolers. Typically, any previous tube failures have been identified by CCW surge tank level changes.

E1.3.5.1.3 Conclusions

There were no concerns identified during this walkdown. The appearance of the CCW system and condition of the equipment appeared to be acceptable.

E1.3.5.2 Instrumentation and Controls

E1.3.5.2.1 Inspection Scope

The team performed a walkdown of the CCW component controls to verify conformance to 10 CFR Part 50, Appendix R, fire protection requirements.

E1.3.5.2.2 Observations and Findings

The DBD for the CCW system identifies that operator action is required to be taken at places other than the Alternate Shutdown Panel during times when the Control Room becomes uninhabitable. Examples of this include actions required to operate the CCW Pump 2C Header Select valves (1-MV-14-1 & -3, 1-MV-14-2 & -4) and the Containment Isolation Valves for Containment Coolers (1-MV-14-9, 10, 11, 12, 13, 14, 15, 16). All of these valves are required to be controlled from their respective MCC's in cases when an alternate shutdown condition arises. The team verified that the above valves could be operated from outside the Control Room during times when the Control Room becomes uninhabitable. The location of the Alternate Shutdown Panels and the CCW Motor Control Centers are on the same elevation and in close proximity to each other.

The specific operator actions required to safely shutdown the plants following a postulated fire in the Control Room or Cable Spreading room are contained in ONOP 1/2-003C135, "Control Room Inaccessibility." The Unit 2 procedure was originated in 1985 by the licensee's engineering group based on the specific requirements contained in the Unit 2 safe shutdown analysis. The Unit 1 procedure was developed based on the Unit 2 procedure. The current version of these procedures rely on actions from four operators for safe shutdown: Reactor Control Operator A; Reactor Control Operator B; the Assistant Nuclear Plant Operator; and the Senior Nuclear Plant Operator. Each operator has specific duties and actions that must be completed for safe shutdown. The Senior Nuclear Plant Operator is specifically excluded from also being a member of the fire brigade, since his first action outside of the Control Room is dedicated to manually isolating the pressurizer PORVs to preclude the potential for a loss of reactor coolant.

As described in NRC Inspection Report Nos. 50-335/85-06 and 50-389/85-06, the licensee's normal shift staffing was reviewed to verify that sufficient personnel are available to operate equipment and systems described in Emergency Operating Procedure, EOP 2-0030144, Alternate Shutdown. The inspection report indicated that adequate shift staffing was being provided to man the necessary stations to support plant operations. The shift operating personnel provided to support EOP 2-0030144 were separate from the operating personnel assigned to the fire brigade.

E1.3.5.2.3 Conclusions

The CCW system and components appeared to meet Appendix R requirements. Control room staffing had been previously reviewed and determined to be adequate.

E1.3.5.3 Electrical

E1.3.5.3.1 Inspection Scope

The team observed in the control room the 2A Diesel Generator six (6) month surveillance test on 1/8/97.

E1.3.5.3.2 Observations and Findings

Operating Procedure 2-2200050A was reviewed with the shift supervisor to understand the sequence of events and expectations when the test started. The test was witnessed up to step 30.B. No issues were identified and the recorded data was within the allowed acceptance criteria.

The following day the team was advised that the surveillance test had not been completed successfully. The 2A Emergency Diesel Generator had tripped on reverse power as the Reactor Control Operator (trainee) was lowering the load for shut down. The Licensed Operator supervising the trainee observed that the Control Room indication for 2A diesel generator electrical output (diesel load is read from a KW paper chart recorder) was slow in indicating actual load as the load was changed. As the load was lowered in preparation to open the 2A diesel output breaker, the trainee, in focusing on the KW paper chart

recorder, had lowered the governor control past the "zero" point and into the "reverse power" region such that the generator was acting as an electric motor to drive the diesel engine. The 2A Emergency Diesel tripped on reverse power as designed.

The cause of the reverse power trip was attributed to the slow response of the KW load chart combined with the inexperience of the trainee. Condition Report CR 97-0030 was written to provide documentation that this event did not adversely affect the operability of the 2A Emergency Diesel Generator. The team upon further analysis of this event noted that prior to the test (at the time of arrival in the Control Room to witness the test), the KW chart recorder that caused this event was opened and partially pulled out of the control board. The team also observed the licensed operator giving instruction to the trainee that the recorder pen sticks and that the trainee should be prepared to nudge the pen with his finger if it sticks when he begins to load the diesel.

E1.3.5.3.3 Conclusion.

Failure to document and take appropriate corrective actions for the sticking pen recorder led to mis-operation of the diesel by the operations trainee and the need for repeating the diesel surveillance. This issue is identified as UNRESOLVED ITEM #50-389/96-201-04.

E1.4 Exit Meeting

On January 10, 1997 the team members conducted a technical debrief with their utility counterparts. On January 28, 1997 the team leader conducted a final public exit meeting where the team's overall conclusions were presented. Upon commencement of the exit meeting, the NRC team leader answered questions from local media representatives. The following individuals were present for the final exit meeting:

NRC	POSITION
J. Jacobson	Team Leader, NRR
R. Pettis	Team Leader, NRR
R. Gallo	Chief, Special Inspection Branch, NRR
D. Collins	Acting Deputy Director, Division of Reactor Projects, Region II
L. Wiens	Project Manager, NRR
M. Miller	Senior Resident Inspector, Region II

FLORIDA POWER AND LIGHT COMPANY	POSITION
R. Noble	Licensing
E. Weinkam	Licensing Manager
J. Hoffman	Inspection Team Leader
R. Gil	Plant Engineering Manager
C. Bible	Engineering Manager
R. Kundalkar	Engineering Vice President
J. Scarola	Plant General Manager

J. Stall	Site Vice President
T. Plunkett	President Nuclear Operations
S. Khuzan	Chief Mechanical Engineer
R. Dietz	Licensing
M. Migliaro	Chief Electrical/I&C Engineer
J. Tringali	Pincipal Engineer/Electrical Maintenance
J. Vassello	NUSIS

ATTACHMENT 1

LIST OF INSPECTOR FOLLOW-UP AND UNRESOLVED ITEMS

NUMBER	FINDING TYPE	PAR #	TITLE
50-335/96-201-01	IFI	E1.2.2.1.3	CST VOLUME REQUIREMENTS
50-335/96-201-02	IFI	E1.2.2.2.3	CALCULATIONS AND INDICATION FOR AFW FLOW
50-335/96-201-03	IFI	E1.2.2.3.3	AFW CROSSTIE NPSH
50-335/96-201-04	IFI	E1.2.2.4.3	CALCULATION REVISION FOR AFW PIPING SUPPORTS
50-335/96-201-05	URI	E1.2.2.5.3	EQ OF WOODWARD GOV CONTROLS
50-335/96-201-06	IFI	E1.2.2.6.3	FULL FLOW TESTING OF AFW CROSSTIE
50-335/96-201-07	URI	E1.2.3.8.2	LACK OF TESTING AND OPERATING PROCEDURES FOR DC BREAKER CROSS TIES
50-335/96-201-08	URI	E1.2.3.8.3	INADEQUATE TROUBLESHOOTING DOCUMENTATION
50-335/96-201-09	IFI	E1.2.4.1.3	LACK OF TRACKING FOR UNIDIRECTIONAL DRIFT
50-335/96-201-10	IFI	E1.2.4.2.3	LACK OF LOOP ACCURACY CALCULATIONS FOR INDICATION ONLY INSTRUMENTS
50-335/96-201-11	URI	E1.2.5.1.3	LACK OF 50.59 EVALUATION FOR INSTALLATION OF MOV COVERS
50-335/96-201-12	IFI	E1.2.5.2.3	LACK OF MAINTENANCE PROCEDURE FOR CHANGING PANEL FILTERS
50-389/96-201-01	IFI	E1.3.2.1.3	CCW PERFORMANCE CURVES
50-389/96-201-02	IFI	E1.3.2.2.3	OPERATIONS NIGHT ORDERS FOR USING PERFORMANCE CURVES
50-389/96-201-03	IFI	E1.3.4.2.3	LACK OF CALC FOR CCW RAD MONITOR SETPOINTS
50-389/96-201-04	URI	E1.3.5.3.3	FAILURE TO TAKE APPROPRIATE CORRECTIVE ACTIONS FOR DEGRADED PEN RECORDER
50-335 and 389/96-201-01	URI	E1.2.3.1.3 E1.3.2.1.2	FAILURE TO UP-DATE THE UFSAR AS REQUIRED BY 10CFR50.71(e)