ENCLOSURE 2

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Inspectors:	T. Stetka, Team Leader, Engineering Branch P. Qualls, Reactor Inspector, Engineering Branch S. Burton, Resident Inspector, Projects Branch C R. Wharton, Project Manager, Office of Nuclear Reactor Regulation D. Prevatte, Consultant
Approved By:	Chris A. VanDenburgh, Chief, Engineering Branch Division of Reactor Safety

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EXECUTIVE SUMMARY

Fort Calhoun Station
NRC Inspection Report 50-285/97-06

This team inspection evaluated the current effectiveness of the licensee's plant and design engineering organizations to respond to routine and reactive site activities, which included the identification and resolution of technical issues and problems. This inspection assessed engineering and technical support by focusing on the functional aspects of the auxiliary feedwater system and portions of the component cooling water and raw water systems. The inspection also reviewed 10 CFR 50.59 safety evaluations and screenings, engineering evaluations for design modifications, and the fire protection program implementation. The inspection covered a 4-week period with 2 of these weeks conducted on site.

Engineering

- The conduct of engineering activities was considered to be generally good. Aspects of good engineering practices included strong system engineers, a reasonable engineering backlog, effective control of plant modifications, good interfaces between engineering and other plant disciplines, a good design basis information process, and an effective independent safety engineering group. However, inadequacies identified regarding the implementation of the fire protection program detracted from this performance. There were instances where design requirements were not properly incorporated into surveillance testing procedures and a technical specification limiting condition for operation and where design calculations were in error. In addition, there was an instance where an Updated Safety Analysis Report update was inadequate. The implementation of the 10 CFR 50.59 program and the technical specification interpretation program were effective.

- The licensee was effective in maintaining the design and operable status of the reviewed systems, and engineers were knowledgeable of their assigned systems. However, weaknesses were identified where surveillance test procedure acceptance criteria for safety-related pumps were inadequate and where a design calculation was in error. These findings were considered to be the first and second examples of a violation of 10 CFR Part 50, Appendix B, Criterion III (Section E1.1).

- Plant modifications were designed, installed, and tested in accordance with approved procedures. Modification packages were properly evaluated for safety impact and plant documentation affected by these modifications were properly revised (Section E1.2).

- Temporary modifications were properly implemented and appropriate application of 10 CFR 50.59 screenings and evaluations were evident (Section E1.3).

- The majority of condition reports had resolutions with proper engineering justification, adequate proposed corrective actions, and adequate operability evaluations. However, weak operability evaluations were identified in three condition reports (Section E1.4).

- Engineering actions and engineering action requests were completed in accordance with approved procedures and properly addressed problems that were related to routine and safe plant operation (Section E1.5).

- The substitute replacement item engineering change notice process was being properly implemented. Self assessments were effective in identifying deficiencies and providing corrective actions for the associated deficiencies (Section E1.6).

- An incorrect technical specification limiting condition for operation involving the minimum water level for the emergency feedwater storage tank was identified. This was considered to be a third example of a violation of 10 CFR Part 50, Appendix B, Criterion III (Section E2.1b.1).

- A discrepancy between the plant configuration and the Updated Safety Analysis Report was identified involving the diesel-driven auxiliary feedwater pump fuel oil day tank level. This was considered to be a violation of 10 CFR Part 50.71e (Section E2.1b.2).

- The controls for design basis documents were effective in maintaining the documents current and accurate. Self assessments of these documents were critical and had good findings (Section E2.2).

- The engineering backlog was reasonable and had properly set priorities. Engineering was effective in the management of the backlog and maintained an essentially constant trend (Section E2.3).

- Except for one safety evaluation weakness, the procedural guidance, program implementation, and training guidelines for the 10 CFR 50.59 safety evaluation process were very good (Section E2.4).

- The process for initiating, maintaining, and closing technical specification interpretations was good. A recently developed technical specification interpretation review panel initiated procedure revisions, a multi-disciplined technical specification interpretation technical review and provided increased management oversight of the technical specification interpretation process (Section E2.5).

- Plant walkdowns indicated that the material condition of the plant was good and that housekeeping was acceptable. The recently implemented housekeeping controls should improve these conditions (Section E2.6).

- Self assessments and reviews to determine root causes and identify improvements with the configuration process were thorough and self critical (Sections E2.7, E8.1 and E8.2).

- The issues identified in the response to the NRC's 10 CFR 50.54(f) letter on design control were resolved or were in the process of being resolved. All issues reviewed were found to be completed or properly scheduled for completion (Section E3).

- Engineers were well-qualified, familiar with their assigned and supporting systems, cognizant of system conditions, and versed in engineering procedures. Engineering expectations were effectively communicated and well understood by the engineering staff. Engineering management was effective in establishing a strong engineering work ethic. Engineers interfaced effectively with other plant organizations (Section E4).

- Engineers were qualified both in assigned and interfacing systems. Training for engineers was effective and contained a strong operations interface (Section E5).

- The nuclear safety review group was aggressive in their approach to safe plant operations. Recent self assessments and management changes resulted in improved performance and credibility with plant organizations (Section E6).

- Quality assurance audits and surveillances reflected the proper level of detail and focused attention in areas of safety significance (Section E7.1).

- Self assessments addressed areas of safety significance. The methodology for development of self-assessment activities was sound, drew proper conclusions, and developed effective recommendations and corrective actions (Section E7.2).

- The implementation of the fire protection program was poor, in that, the inspection identified five examples of the failure to properly implement the fire protection program. These included: (1) diesel generator control circuits that were not protected from a fire, (2) an inadequate alternate shutdown procedure, (3) an inadequate water curtain, (4) an inadequate reactor coolant pump motor lube oil collection system, and (5) inadequate control of fire pump operations. These examples were considered to be five apparent violations of the fire protection program (Section F1).

- The fire protection equipment required for program implementation was well maintained and available for immediate use. The fire detection and alarm capability were considered to be good (Section F2).

- With the exception of the procedure inadequacies identified in Section F1 of this report, the fire protection program procedures adequately implemented the approved fire protection program (Section F3).

- The fire protection staff was qualified and had a very good working relationship with other station organizations (Section F4).

- The fire brigade training was considered to be adequate to meet NRC requirements. However, there was a weakness in fire brigade member knowledge concerning the use of water to suppress an electrical cable fire. In addition, one deficiency regarding the training program was identified and was considered to be a sixth apparent violation of the fire protection program (Section F5).

- The fire protection organization and administration was being implemented in accordance with the fire protection program (Section F6).

- The fire protection program audits were found to comply with the minimum requirements of the program (Section F7).

Report Details

Engineering

E1 Conduct of Engineering (37550)

E1.1 System Reviews

a. Inspection Scope

The team reviewed the auxiliary feedwater system and portions of the component cooling water and raw water systems to verify that these systems were maintained as designed and in an operable status. This review included 11 system drawings, 18 procedures, 17 design calculations, 7 permanent modifications, 9 operability evaluations, and 7 vendor documents. In addition, the team interviewed the cognizant system engineers to determine the engineers' knowledge of the systems.

b. Observations and Findings

The team determined that the engineers were very knowledgeable of their systems and very capable of providing design information. This was evidenced by the engineers' ability to provide prompt responses and information for the team's questions. In addition, the team found that the licensee was properly maintaining the design and operational status of the reviewed systems. However, during review of the design calculations, the team identified discrepancies that affected system design and procedures.

b.1 Auxiliary Feedwater Pumps FW-6 and FW-10

Calculation FC05361 determined the performance requirements for the motor-driven auxiliary Feedwater Pump FW-6 and the turbine-driven auxiliary Feedwater Pump FW-10. To calculate the required developed head for each pump, Calculation FC05361 used the maximum steam generator accident pressure, which is the pressure at which the first main steam safety valve lifts. Calculation FC05361 used 1,015 psia as the main steam safety valve's nominal setpoint. This appeared to be a conservative value since the technical specifications required a main steam safety valve setpoint of 1,000 psia. However, upon further review of the calculations, the team determined that the calculation did not consider the setpoint tolerance of the main steam safety valves. Since the technical specifications allowed a setpoint tolerance of ± 3 percent, the main steam safety valve could be set as high as 1,030 psia.

In addition, the team noted that the calculation did not account for safety valve pressure accumulation. This added an additional 3 percent, or 31 psia to the setpoint value. The net effect of these errors was an increase of 46 psia to a maximum steam generator pressure of 1,046 psia.

The turbine-driven auxiliary feedwater pump's required performance was determined for the turbine/pump combination. This required consideration of the differences between the turbine inlet and exhaust conditions at various operating points and comparing these to the vendor's performance data. In calculating the pump's required performance, the licensee used the vendor's pump performance curves. The vendor's data was based on a 5.0 psig turbine exhaust pressure. However, the inspection team noted that the licensee's calculation used 1.0 psig as the exhaust pressure, which made the turbine's calculated performance appear to be better than actual performance. However, the inspection team determined that Pump FW-10 was not inoperable due to other conservatisms. These conservatisms included no credit in the calculation for the remaining steam generator inventory (i.e., it assumed an empty steam generator) and the assumption in the calculation that the pump only delivered flow early in the accident, whereas, the pump would actually provide flow for the entire accident scenario.

The licensee used the data from this calculation to determine the acceptance criteria for the pump's surveillance test procedure used for inservice testing. Although the inspection team did not identify any problems with the surveillance procedures for Pump FW-10, a problem was noted with the surveillance procedure for Pump FW-6. Procedure SE-ST-AFW-3005, "Auxiliary Feedwater Pump FW-6 and Check Valve Test," stated that the minimum allowable performance was 990 psid (developed head) at 200 gpm. This value was determined in accordance with the ASME Code, Section XI, which allowed a 10 percent degradation from the pump vendor's reference value of 1100 psid during the inservice testing. However, the team also noted that Calculation FC05361 required a minimum head of 1,032.7 psid at a 200 gpm flow, even before the errors due to steam generator safety valve setpoint tolerance and pressure accumulation were incorporated. When the calculation was modified to account for these errors, the team determined that the minimum required head should have been 1,078.7 psid at 200 gpm. Therefore, the acceptance criteria for Procedure SE-ST-AFW-3005 allowed this pump to be considered operable even though it would not have met the accident analysis minimum performance requirements.

The team noted that the latest completed data for Procedure SE-ST-AFW-3005 performed on February 27, 1997, indicated that the pump's performance was 1,100 psid at a flow of 200 gpm. These data indicated that the pump's actual performance was in excess of that required to mitigate the design accident conditions. Therefore, the team considered the pump to be operable.

NRC guidance regarding inservice testing of pumps was established in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants." This guidance indicated that it was the NRC staff's expectation that pump performance acceptance criteria be established that does not conflict with operability criteria for

flow rate and differential pressure in the safety analysis report. Furthermore, this guidance indicated that the operability limits of pumps must always meet or be consistent with licensing basis assumptions in a plant's safety analysis. The finding identified by the team indicated that the licensee's acceptance criteria development was inconsistent with this guidance.

Subsequent to the team's discovery of these discrepancies, the licensee initiated a review of all safety-related pumps in their inservice testing program. Eight additional pumps were reviewed, including the boric acid transfer, charging, component cooling water, diesel generator fuel oil transfer, high pressure safety injection, containment spray, low pressure safety injection, and raw water pumps. Of these, the high pressure safety injection, containment spray, low pressure safety injection, and raw water pumps were determined to have surveillance procedure acceptance criteria lower than what was required by the accident analyses. However, the team's review of the actual test data indicated that all these pumps had sufficient discharge head and flows for accident mitigation.

10 CFR Part 50, Appendix B, Criterion III, requires that the design bases are correctly translated into procedures. The failure of the licensee to implement design basis requirements into surveillance test procedure acceptance criteria for auxiliary Feedwater Pump FW-6, the high pressure safety injection pumps, the containment spray pumps, the low pressure safety injection pumps, and the raw water pumps was considered to be the first example of a violation of 10 CFR Part 50, Appendix B, Criterion III (50-285/9706-01).

b.2 Auxiliary Feedwater Pump FW-54

Calculation FC05336 determined the performance requirements for diesel-driven auxiliary Feedwater Pump FW-54. This calculation contained the same error discussed previously for Calculation FC05361 regarding the error in the maximum steam generator pressure. As a result, the team determined that this pump must also provide feedwater into a steam generator at a maximum pressure of 1,046 psia.

The data from this calculation were used to set the acceptance criteria for the pump's preventive maintenance procedure. Procedure OP-PM-AFW-0004, "Third Auxiliary Feedwater Pump Operability Verification," provided the performance testing criteria for Pump FW-54. This procedure stated that the minimum allowable performance was 915 psid at 299 gpm. However, the team noted that Calculation FC05336 required a minimum head of 1,078.1 psid at 325 gpm, even before the error in this calculation was corrected. When corrected, the minimum required head should have been 1,124.1 psid at 325 gpm.

The team noted that the latest completed data for Procedure OP-PM-AFW-0004 performed on March 31, 1997, indicated that the pump provided 1,223.7 psid at a flow of 325 gpm. These data indicated that the pump's actual performance was in excess of that required to provide feedwater to the steam generators.

The team noted that this pump was not safety related and, as a result, no credit was taken for this pump in any accident analysis. However, the errors noted in this calculation were considered to represent a weakness in the licensee's design process.

b.3 Component Cooling Water Temperature Transient

Calculation FC06378 determined the final nitrogen pressure in the component cooling water surge tank after a loss-of-coolant-accident temperature transient. This pressure was required to assure sufficient component cooling water pump net positive suction head for all accident scenarios.

A loss-of-coolant accident would increase the bulk component cooling water system temperature causing thermal expansion of the water in the system and a surge tank level rise. This would compress the nitrogen blanket in the tank and cause the pressure control valve to open and release nitrogen. As the system cooled, the water volume would shrink and lower the tank level and pressure because some nitrogen was lost during the transient. The calculation indicated that the final tank pressure from this transient would be 13.2 psig.

During the review of this calculation, the team noted that the calculation used a pressure control valve reset pressure of 29.8 psig. However, the calculation did not consider the single failure potential for this valve. If the valve failed, the tank's safety relief valve would have to open to protect the tank. The reset pressure for this valve was 26 psig. When the single failure was postulated, the tank pressure reduction would begin at 26 psig instead of 29.8 psig resulting in a lower final pressure. This final pressure was 9.4 psig or 3.8 psig lower than that used in the calculation. In addition, the team noted that, due to the effects of heatup, the initial system volumes for all components in the system needed to be considered in the calculation. However, the team noted that the total system volume did not include the volume of the containment coolers. When these volumes were included, the final system pressure was lowered by an additional 0.5 psig to 8.9 psig.

The pump data indicated that the component cooling water pump net positive suction head requirements would be met with a tank pressure as low as 5 psig. Therefore, the team concluded that these calculation errors did not render the component cooling pumps inoperable.

10 CFR Part 50, Appendix B, Criterion III, requires that the design control measures be adequate to assure the adequacy of hydraulic analyses. The failure of the licensee to verify the adequacy of the calculation for the hydraulic analysis of the component cooling water surge tank temperature transient was considered to be the second example of a violation of 10 CFR Part 50, Appendix B, Criterion III (50-285/9706-01).

c. Conclusions

The licensee was effective in maintaining the design and operable status of the reviewed systems. Engineers were determined to be very knowledgeable of their assigned systems. However, weaknesses were identified, in that, the surveillance test acceptance criteria for safety-related pumps were not conservative. In addition, the team identified a design error in the hydraulic analysis of the component cooling water surge tank temperature transient. These findings were considered to be two examples of a violation of 10 CFR Part 50, Appendix B, Criterion III.

E1.2 Permanent Plant Modification Review

a. Inspection Scope

The team reviewed 14 plant modifications to verify conformance with applicable installation and testing requirements as prescribed by procedures. Specific attributes reviewed and/or verified by the team included 10 CFR 50.59 safety evaluations, post-modification testing requirements, safety-related drawing updates, conformance with the Updated Safety Analysis Report and design basis documents, training requirements, and field installations.

b. Observations and Findings

The modification packages contained the required 10 CFR 50.59 screenings. The team verified that affected drawings, procedures, and references were updated associated with two fire protection modification packages, an emergency diesel generator air compressor modification package, and a spent fuel pool rerack modification package. A check of the field changes added to the modification packages indicated that they were implemented in accordance with procedures and contained the proper safety reviews. The team verified by a walkdown of the modifications that the installed changes were consistent with the package descriptions.

c. Conclusions

The plant modifications reviewed were designed, installed, and tested in accordance with approved procedures. Modification packages were properly evaluated for safety impact utilizing 10 CFR 50.59 screenings and evaluations as required. Design basis and plant documentation affected by the modifications were properly revised.

E1.3 Temporary Plant Modification Review

a. Inspection Scope

The team reviewed 13 temporary plant modifications to verify conformance with applicable installation and testing requirements as prescribed by procedures. Specific attributes reviewed by the team included 10 CFR 50.59 safety evaluations and plant installations.

b. Observations and Findings

There were 13 open temporary plant modifications. Of these 13 temporary modifications, 12 were nonsafety related. Temporary Modification 96-042, "CCW System Relief Valve Setpoint Change," was safety related and required a 10 CFR 50.59 safety evaluation. The modification increased the component cooling water tank operating pressure and gagged several system relief valves. The modification was developed in accordance with approved procedures, the appropriate safety evaluation was completed, and any Updated Safety Analysis Report references were identified. The team also reviewed the content and safety evaluation screenings on the remaining 12 temporary modifications and noted no discrepancies.

c. Conclusions

Temporary modifications were implemented in accordance with approved procedures. Effective control of the number of outstanding temporary modifications and appropriate application of 10 CFR 50.59 screenings and evaluations were evident.

E1.4 Condition Reports

a. Inspection Scope

The licensee issued condition reports as a means to identify problems with components and systems and to place these problems in their corrective action system for resolution. The team reviewed 21 condition reports to determine the adequacy of the resolution, whether the component/system operability was properly determined, and that the proposed corrective actions were adequate to preclude recurrence. In addition, the team interviewed the applicable licensee personnel to discuss the resolution of the condition reports.

b. Observations and Findings

The team noted that 18 of the 21 condition reports had resolutions with proper engineering justification, adequate proposed corrective actions, and adequate operability evaluations. However, the team identified three operability determinations that were considered to be weak.

b.1 Failure of Nonsafety-Related Instrument Air Regulator

Condition Report 199600161 identified that nonsafety-related instrument air regulators supplied 15 safety-related solenoid operated valves. The concern identified in the condition report was that a regulator failure to the full open position would subject the solenoid valves to differential pressures greater than their rated maximum operating differential pressure. The operability evaluation for this condition report, however, stated that the air regulators were not expected to fail because they were high quality, tested by the vendor, and replaced on a 5-year frequency.

The team considered this operability evaluation to be inadequate, in that, inappropriate credit was being taken for nonsafety-related equipment. However, further review by the team indicated that, as the result of this condition report, the licensee replaced the solenoid valves with valves that would operate with full instrument air system pressure applied.

Since these air regulators and solenoid valves provided the operating air supply to safety-related, air-operated valves, the team questioned the licensee regarding the operability of the air-operated valves, if at the time of a regulator failure, the solenoid valve was open. Under these conditions, full instrument air system pressure would be supplied to the air-operator. The licensee was unable to provide information to the team regarding the effect of full instrument air system pressure on the air operators. The licensee will review this finding to determine effect of full instrument air system pressure on the valve air-operators. The NRC will review the results of this evaluation. This is considered to be an inspection followup item (50-285/9706-02).

b.2 Gate Valve Pressure Locking and Thermal Binding

Condition Report 19970067 documented that shutdown cooling isolation Valves HCV-347 and HCV-348 could be susceptible to pressure locking and thermal binding conditions. To address the valve's operability for the pressure locking issue, the associated operability evaluation determined that these valves were not susceptible to pressure locking due to existing valve seat and packing leakage. This determination reasoned that this leakage prevented the valve bonnets from becoming pressurized and causing pressure locking.

The team considered this determination to be inadequate because pressure locking was a function of the valve design, whereas, the valve seat and packing leaks were a function of the valve's current material condition. The team was concerned that if these leakage conditions were corrected, the pressure locking phenomena could occur. The team noted that while the licensee developed a modification to drill the valve disks, such that bonnet pressure relief would be assured, the licensee had not yet decided to perform this modification on these valves. However, the licensee

informed the team that any maintenance on these valves was controlled by the system engineer and that these valves were inaccessible during plant operations. As a result, the licensee was confident that leak repairing maintenance on these valves would not be performed without appropriate engineering controls.

To address operability for thermal binding, the associated operability evaluation determined that the valves would be closed during startup, with a maximum temperature of 400°F and a pressure of 250 psig in the reactor coolant system. Furthermore, since the valves would not be required to open for an accident until the reactor cooling system temperature and pressure was less than 353°F and 140 psig, the evaluation determined that the valves would not be significantly cooler when they were needed to be opened. Therefore, the evaluation concluded that the valves were not susceptible to thermal binding.

The team noted, however, that since the reactor could be completely depressurized when these valves were required to open, the valves could be significantly cooler when required to open (212°F versus 400°F). Therefore, the team concluded that the potential for thermal binding still existed.

The licensee stated that since these valves were in the alternate hot leg injection flow path, for which they did not take accident mitigation credit, they had not determined if the functioning of these valves was a safety concern. The licensee was still reviewing this issue and expected to resolve the issue before the completion of the next refueling outage. Their decision to make the disk modification would be based on resolution of this issue.

The NRC requested all licensees to address the valve thermal binding and pressure locking issue by responding to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Power Operated Gate Valves." In their response to this generic letter, the licensee included these two valves as valves subject to this review. Since the licensee commitments with respect to these valves are being reviewed by the Office of Nuclear Reactor Regulation, further followup on this issue will be pursued by the program office. In addition, these specific concerns have been discussed with the appropriate program office reviewers.

b.3 Safety-Related, Air-Operated Equipment Not Protected with Local Filter

Condition Report 199700168 documented that the air operators and their respective solenoid valves (HCV-3008-0 and HCV3009-0) for Valves HCV-3008 and HCV-3009 were not protected with local air filters even though Design Basis Document SDBD-CA-105 stated that they were equipped with local air filters. The function of Valves HCV-3008 and HCV-3009 was to actuate the fire suppression deluge systems for the charcoal filters in the control room heating, ventilation, and air conditioning units. The function of Solenoid Valves HCV-3008-0 and HCV3009-0 was to protect the charcoal filters from inadvertent actuation of the fire water deluge systems. These valves were required to remain closed to perform this safety function.

The operability evaluation determined that local filters were not required because: (1) the air was filtered and dried by the plant instrument air system; therefore, the chances were remote that a particle could plug the solenoid valve orifice; (2) the deluge valves could be actuated manually; and (3) Valves HCV-3008-0 and HCV-3009-0 would fail closed and would not inadvertently actuate the deluge valves.

The team noted that the requirement for local filters was in addition to the filters and dryers at the air source. In addition, the team noted that the concern was loose particulates between the air source and the valve operators, and that the filters at the air source could not provide such protection. Furthermore, the plugging of the solenoid valves or air supplies was not a concern. The primary concern was that a particle could prevent the valves from closing completely, thereby inadvertently actuating the deluge valves.

The team concluded that the actual operability evaluation was that, since the valves were normally closed and only opened to actuate the deluge, there was no credible condition that would allow particulates to get into valves to prevent closure. However, this reason was never identified in the operability evaluation.

c. Conclusions

The majority of the condition reports had resolutions with proper engineering justification, adequate proposed corrective actions, and adequate operability evaluations. However, three condition reports were identified in which the operability evaluations were weak.

E1.5 Engineering Actions and Engineering Action Requests

a. Inspection Scope

The team reviewed one safety-related engineering action and four safety-related engineering action requests. This review compared the engineering action and action requests to associated procedures and determined if the assumptions were technically reasonable and properly supported.

b. Observations and Findings

The team found that the calculations, analysis, and methodology supported the assumptions and criteria identified for completion of the analyses. The team noted that information required by the engineering action requests properly identified the safety impact and provided resolution of the pertinent operational or maintenance problems.

c. Conclusion

Engineering actions and engineering action requests were completed in accordance with approved procedures and properly addressed problems that were related to routine and safe plant operation.

E1.6 Engineering Change Notices

a. Inspection Scope

The team reviewed 14 safety-related, substitute replacement item engineering change notices to verify conformance with applicable procedures. This review included a verification that the plant modification process was not circumvented by the use of this process. The team also reviewed a nuclear safety review group self assessment of the substitute replacement item process.

b. Observations and Findings

The substitute replacement item engineering change notice process was used to perform technical evaluations to determine suitability for an item that replaces an original or installed component. It was used when the replacement component did not have the same make, model, and/or part number as the original component.

The team found all substitute replacement item engineering change notices to be in compliance with the applicable procedures. The team also found that the nuclear safety review group sampled 50 substitute replacement item engineering change notices during their self assessment. This self assessment identified two instances, involving applications of the substitute replacement item process where the process was not effective. One discrepancy involved the installation of a nonsafety-related local steam generator direct-reading, blow-down temperature indicators with thermocouples that had a remote readout mounted locally on the wall. The other discrepancy involved a lack of detail in the engineering change notice. This engineering change notice did not provide the information to explain the reason for the expanded calibration ranges on the containment spray sodium injection tank level switches. The actual switch calibration was not affected by this discrepancy and the switches were properly calibrated. The team noted that these discrepancies were entered into the condition reporting process and were being tracked for resolution.

c. Conclusions

The substitute replacement item engineering change notice process was being implemented properly. Self assessments were effective in identifying deficiencies and providing corrective actions for the associated deficiencies.

E2 Engineering Support of Facilities and Equipment (37550)

E2.1 Review of Facility and Equipment Conformance to the Updated Safety Analysis Report Description

a. Inspection Scope

A recent discovery of a licensee operating its facility in a manner contrary to the Updated Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures, and/or parameters to the Updated Safety Analysis Report descriptions. As the result of this discovery, the team reviewed selected sections of the Updated Safety Analysis Report.

b. Observations and Findings

Procedure PED-QP-2, "Configuration Control," defines configuration changes and references the appropriate procedures for control of the same. The Updated Safety Analysis Report was maintained current using this procedure and the 10 CFR 50.59 safety review process. The team's review of one permanent modification, four temporary modifications, and two engineering change notices indicated that appropriate changes to the Updated Safety Analysis Report were completed. In addition, the team reviewed five condition reports and four quality assurance surveillances that identified discrepancies in the Updated Safety Analysis Report. The team noted that all of these licensee-identified deficiencies were administrative or typographical in nature and had no safety significance.

The team also verified portions of the Updated Safety Analysis Report while conducting detailed reviews of the auxiliary feedwater, component cooling, and raw water systems. Reviews of these systems identified the following deficiencies between the Updated Safety Analysis Report and the design specifications:

b.1 Emergency Feedwater Storage Tank

Updated Safety Analysis Report, Section 9.4.6, and the basis for Technical Specification 2.5 specified that the amount of water in the emergency feedwater storage tank was adequate to remove decay heat for 8 hours. Calculation FC06148 determined the water volume needed to satisfy this statement. During a review of this calculation, the team noted that Updated Safety Analysis Report, Sections 14.10 and 14.12, specified that the initial core power for accidents requiring auxiliary feedwater was 102 percent of rated thermal power (1,530 MWt); however, the calculation used 100 percent (1,500 MWt) to determine the auxiliary feedwater needs. Calculation FC06148 also did not account for the main steam safety valve setpoint tolerance and accumulation in determining the maximum steam generator pressure (as previously discussed in Section E1.1 of this inspection report).

Calculation FC06148 indicated that the required storage was 52,654 gallons. However, due to the power level and steam generator pressure errors in this calculation identified by the team, the required storage should have been 53,824 gallons (1,170 gallons more). In addition, the team found that the technical specification limit of 55,000 gallons, when decreased to account for the expected instrument error of ± 3.9 percent, was only 52,855 gallons (55,000 gallons minus 2145 gallons). Therefore, the technical specification limit, when decreased due to the instrument error, was 969 gallons (53,824 gallons minus 52,855 gallons) less than the 53,824 gallons required to meet the 8-hour Updated Safety Analysis Report requirement.

The team questioned the licensee regarding the levels being maintained in the tank. The licensee responded that they maintained the tank at a level of between 90 (57,500 gallons) to 100 percent of indicated full level. Therefore, the licensee was maintaining the tank at levels in excess of that determined by the corrected calculation.

10 CFR Part 50, Appendix B, Criterion III, requires that the design bases are correctly translated into specifications, drawings, procedures and instructions. The failure of the licensee to correctly translate the design basis requirement for the emergency feedwater storage tank into the technical specification limiting condition for operation is considered to be the third example of a violation of 10 CFR Part 50, Appendix B, Criterion III (50-285/9706-01).

b.2 Diesel-Driven Auxiliary Feedwater Water Pump Fuel Oil Day Tank Capacity

Updated Safety Analysis Report, Section 9.4.6, stated that the fuel oil day tank for the auxiliary feedwater pump (FW-54) provided fuel storage capacity for 8 hours of continuous operation. The team requested documentation of this capacity from the licensee. The licensee informed the team that no calculation or test results were available to demonstrate this capacity. However, due to the team's question, the licensee performed Calculation FC06638 to demonstrate the tank's capacity.

The team also reviewed Preventive Maintenance Procedure OP-PM-AFW-0004, "Third Auxiliary Feedwater Pump Operability Verification," which was used to test the capability of Pump FW-54. This procedure required refilling the fuel oil day tank if the tank was less than one-half of full after completion of the test. Using information from Calculation FC06638, the team calculated that at one-half full, the tank would contain sufficient fuel for only approximately 4.4 hours of continuous operation.

The licensee stated that it was not their intent in the Updated Safety Analysis Report that the tank be kept at a level to assure 8 hours of fuel. The licensee's position was that the Updated Safety Analysis Report documented only that the tank volume could provide 8 hours of continuous operation (if full) and not that it be

maintained full at all times. However, the team noted that this position was inconsistent with a statement in the pump installation modification package (MR-FC-88-17). This modification package stated that the fuel oil day tank would supply sufficient fuel for a minimum of 8 hours of full load operation.

10 CFR 50.71e requires that the Updated Safety Analysis Report be updated periodically to assure that the information included in the Updated Safety Analysis Report contains the latest material developed. The failure to provide an adequate update to the Updated Safety Analysis Report to reflect the actual intent of the Auxiliary Feedwater Pump FW-54 day tank capacity was considered contrary to the requirements of 10 CFR 50.71e and is considered to be a violation (50-285/9706-03).

c. Conclusion

With the exception of deficiencies associated with the emergency feedwater storage tank and the diesel-driven, auxiliary feedwater pump fuel oil day tank level requirements, no safety-significant discrepancies in the Updated Safety Analysis Report were noted. Additionally, the team concluded that the licensee was identifying and correcting deficiencies between the plant configuration and the Updated Safety Analysis Report.

E2.2 Validation and Control of Design Basis Documents

a. Inspection Scope

The team reviewed two design basis documents for conformance with the requirements identified within the Updated Safety Analysis Report. In addition, the team reviewed the licensee's procedures and controls to assess the ability to retrieve design basis documents and supporting information. The team also reviewed the licensees response to the request for information pursuant to 10 CFR 50.54(f). The results of this review are discussed in Section E3 of this report.

b. Observations and Findings

The team found that design basis documents were controlled in accordance with applicable procedures which required design documents be updated as relative information changes.

The team also reviewed three self assessments of the design basis documents against the Nuclear Energy Institute guidelines for maintaining design basis. These self assessments were critical toward identifying deficiencies. In addition, self-assessment findings were properly entered into the corrective action program.

c. Conclusions

The controls for design basis documents were effective in maintaining them current and accurate. Self assessments of design basis documents were critical toward identifying deficiencies. Deficiencies were appropriately entered into the corrective action program.

E2.3 Engineering Backlog

a. Inspection Scope

The team evaluated the extent of backlogged engineering work to determine the size of the backlog and to determine whether the backlog was being managed properly. The evaluation included a review of the backlog reports and interviews with eight members of the engineering staff.

b. Observations and Findings

The team reviewed the licensee's method used to set priorities for backlogged engineering tasks. The licensee groups priorities into categories called safety, plant operability, regulatory significance, plant improvement, and corporate significance. The safety category was the most significant and the corporate significance category was the least significant. Each category had priority levels numbered one through six with the first priority being the most significant and the sixth priority level being the least significant.

The licensee assigned engineering change notices, engineering action requests and modifications to each of these categories. Engineering actions were not included in these categories because they were already associated with a task that was assigned to a specific category and, therefore were completed when the associated task was completed. The team reviewed seven backlogged items that involved safety-related systems or items that were important to safety. The team found that the priorities for the selected items were appropriate, that they did not involve safety significant issues and that they were being properly addressed.

The team reviewed backlog trends, backlog reports, and conducted interviews to assess the management of the backlog. Interviews with engineering management indicated that better management of the backlog was required. This better management was needed to aid the engineers with their work-load management. This included integrating tasks of the same priority, implementing new procedural requirements, and accommodating any staff reductions.

Design engineering had a methodology to capture all open engineering items on one computer data base to assist the engineer with managing the backlog. The team found that system engineering was adapting the design engineering process of putting all of the backlog on one computer data base. Interviews also indicated that

engineers manage their assigned backlog by due date. It was management's expectation that engineers be familiar with assigned tasks and integrate tasks with different due dates as appropriate. Conflicts with due dates and task priorities were resolved by the applicable engineering supervisor. Changes in the due dates were often referred back to a committee to set priorities and were also tracked by date change occurrences.

Interviews with four engineers indicated that there has been no appreciable impact on their assigned backlog due to staff reductions, new training requirements, or additional procedural requirements.

A review of the backlog indicated that the amount of backlogged engineering work items remained approximately constant over the last 2 years. The team found that the engineering backlog consisted of 455 engineering change notices, 240 engineering action requests, and 84 modification requests, for a total of 779 open items. The team found that the oldest modification request was dated 1991 and that the oldest requests did not involve safety or plant significant issues. The team noted a slight increase in the backlog. The licensee attributed this increase to the large number of configuration control related items identified as a result of increased awareness of these type issues.

c. Conclusions

The engineering backlog was reasonable and had properly set priorities. Engineering was effective in their management of the backlog and was maintaining an essentially constant trend.

E2.4 10 CFR 50.59 Implementation (37001)

a. Inspection Scope

The team reviewed the licensee's 10 CFR 50.59 program guidance, implementation, and associated training. The team also reviewed 10 CFR 50.59 applicability screenings and subsequent nuclear safety evaluations for temporary modifications, permanent modifications, engineering change notices, engineering analyses, and procedure changes.

b. Observations and Findings

The licensee's 10 CFR 50.59 safety evaluation process was implemented by Procedure NOD-QP-3, "10 CFR 50.59 Safety Evaluations." This procedure delineated the policy, requirements, and responsibilities regarding the preparation and associated review of nuclear safety evaluations. The policy established a low threshold for screening applicability. Guidelines were provided for conducting the initial screening and for determining if an unreviewed safety question was involved. The procedure required a documented basis to support the screening conclusion or the safety significance determination. The team's review of previous Procedure NOD-QP-3 revisions and a current in-process revision indicated that the

process was constantly being reevaluated and enhanced. The team also noted that the licensee developed a 10 CFR 50.59 improvement program and established a 10 CFR 50.59 oversight committee. The team found these initiatives to be effective in identifying, reviewing, and resolving programmatic issues regarding the 10 CFR 50.59 review process.

In general, the team found the background documentation contained in the nuclear safety evaluations to be well developed and complete. The safety evaluations provided appropriately detailed bases for reaching conclusions regarding changes, tests, and experiments. All conclusions appeared to be logically supported and did not represent any unreviewed safety questions. However, during a review of one of the engineering change notices, the team identified a minor weakness.

Engineering Change Notice 95-130, "CCW Relief Valve Setpoint Changes," changed the component cooling water system's surge tank pressure control valve setpoint from 45 to 34 psig and the setpoints of Relief Valves AC-341 and AC-364 from 50 to 36 psig. The purpose of this modification was to assure that thermal relief valves on system heat exchangers would not open due to the increase in system pressure as the result of a system post loss-of-coolant accident heatup and, therefore, prevent a component cooling water system inventory loss.

As discussed in Section E1.1 of this report, the primary reason for maintaining nitrogen pressure on the component cooling water surge tank was to assure adequate component cooling water pump net positive suction head under all accident conditions. This modification effectively lowered the component cooling water pump final accident suction pressure. Therefore, the potential existed that this modification could have unacceptably reduced the available net positive suction head.

The team noted that the 10 CFR 50.59 safety evaluation written for this modification only addressed the potential of relief valve lifting with the resultant loss-of-inventory. This safety evaluation did not address the potential loss of the component cooling water pump net positive suction head. Therefore, the team concluded that this safety analysis was weak in that it did not address all the safety parameters affected by this modification. In spite of this safety evaluation weakness, the team noted that the component cooling water pumps still had adequate net positive suction head.

The team also interviewed training personnel and reviewed requirements, materials, and records for initial and requalification 10 CFR 50.59 training. The instructor's lesson plans were comprehensive and complete. They reflected numerous revisions to incorporate instructional improvements. The lesson plans addressed historical perspective, regulatory definition, enforcement issues, industry concerns, licensee weaknesses, and recent process enhancements. The most noted enhancement was the ISYS word search computer program, which allowed 10 CFR 50.59 preparers and reviewers to search the documented licensing and design basis for commitments and related safety impact.

The team reviewed the various training feedback mechanisms, student feedback forms, in-class manager evaluations, training advisory committee input, and direct requests for training. The team also reviewed the audit and interviewed the auditor responsible for the 10 CFR 50.59 portion of the recent Safety Audit and Review Committee Audit 62, "Performance Training and Qualification of Facility Staff." This audit did not identify any problems with the 10 CFR 50.59 training process.

The team noted that the licensee had a very effective training program. The licensee recently conducted training for approximately 90 percent of the personnel required to prepare and perform 10 CFR 50.59 safety evaluations.

c. Conclusions

Except for one weak safety evaluation, the team concluded that the licensee's procedural guidance, program implementation, and training guidelines for the 10 CFR 50.59 process were very good.

E2.5 Technical Specification Interpretations

a. Inspection Scope

The team conducted a review of the licensee's technical specification interpretations and the process for requesting, maintaining, and dispositioning these interpretations. This review was conducted due to increased NRC interest in licensee implementation of technical specification interpretations and potential inconsistencies between the technical specifications and licensee developed interpretations. The team reviewed 19 of 29 current technical specification interpretations to determine the adequacy of the program and ensure that there were no conflicts with the technical specifications.

b. Observations and Findings

The team found the technical specification interpretation program was controlled by Procedure NOD-QP-32, "Technical Specification Interpretations." This procedure provided guidance for the intent of technical specification interpretations, documented clarification for existing technical specifications, and established interpretation consistency. The procedure stated that technical specification interpretations would not be used as a revision of, or substitution for, the technical specifications. It further stated that technical specification interpretations would not be considered part of technical specifications and would never contradict or change wording, meaning, or intent of technical specification requirements. The team noted that Procedure NOD-QP-32 was being re-evaluated and enhanced as evidenced by the processing of a recent revision.

Based on a review of the 19 technical specification interpretations, the team determined that none of these interpretations were in conflict with the technical specification requirements and that the program was technically adequate.

The team also noted that the licensee convened a technical specification interpretation technical review panel in response to a licensee-identified violation of technical specifications (that was reported in Licensee Event Report 96-006) that occurred due to an inconsistent interpretation of technical specifications. The technical specification interpretation panel reviewed all open technical specification interpretations for technical adequacy and consistency with technical specifications. The panel also provided programmatic recommendations and initiated process improvements. One major programmatic improvement was the use of technical specification interpretations for only in-process technical specification changes. Other improvements were that technical specification interpretations were assigned a finite validation period of 2 years and that revised technical specification interpretations would no longer be given a new number designation.

c. Conclusions

The team concluded that the licensee's current process for initiating, maintaining and closing technical specification interpretations was good. The licensee's recently convened technical specification interpretation review panel initiated procedure revisions, provided a multi-disciplined technical specification interpretation technical review and increased management oversight of the technical specification interpretation process.

E2.6 System Walkdowns

a. Inspection Scope

At different times during the inspection, the team performed walkdowns of selected plant areas to determine the overall material condition of equipment and the maintenance of housekeeping.

b. Observations and Findings

A tour of the containment building revealed what appeared to be excessive corrosion (rust) on the component cooling water system piping. This observation was discussed with licensee personnel, at which time, the team was informed that a program had already been developed to monitor this corrosion. In addition, the licensee entered the corroded condition into the their condition reporting process as Condition Report 199700154 to assure tracking and closure. The team's review of this program, which included a review of ultrasonic test results used to monitor pipe wall thinning, indicated that the corrosion was not causing any pipe wall thinning problems at this time. The team also noted that the licensee was developing a protective coating plan to cure the problem.

The team also noted that while the housekeeping was acceptable, there were some instances where there was an accumulation of tools, cleaning supplies and carts in the auxiliary building. These observations were discussed with licensee personnel.

The team was informed that the licensee developed a program to control such instances and that the program should be effective in reducing these accumulations. None of the accumulations observed by the team were considered to be excessive and the licensee's control of transient combustibles was considered to be effective.

The team also identified several issues involving fire protection equipment and the collection of reactor coolant pump motor lubrication oil. These observations are discussed in Section F1 of this inspection report.

c. Conclusions

The team's walkdown of the plant indicated that the material condition of the plant was good and that housekeeping was acceptable. The licensee's recently implemented housekeeping controls should improve these conditions.

E2.7 Configuration Control

a. Inspection Scope

The team reviewed the corrective actions associated with configuration control. As a part of this review, the team evaluated the licensee's self assessment for configuration control to determine if the associated conclusions, recommendations, and corrective actions were adequate to address the issue.

b. Observations and Findings

On November 22 through December 10, 1996, the NRC conducted a special inspection (NRC Inspection Report 50-285/96-17) on the post-accident sampling system. As a response to the enforcement actions, which resulted from this inspection, as well as, internal concerns associated with other configuration control deficiencies, the licensee committed to perform a self assessment of their configuration control practices. The licensee committed to complete this self assessment by April 30, 1997, and send the report to the NRC by May 30, 1997. The licensee completed their self assessment on April 11, 1997, and provided the results to the inspection team.

The report, "Configuration Control Self Assessment Final Report," dated April 11, 1997, assessed current procedures, condition reports, industry experience, and included field observations. This self assessment encompassed the period of September 1995 through March 1997, which was the period that the new condition reporting program was implemented. The team noted that this period included previous plant status reports (e.g., incident reports) because these historical reports were captured by issuing new condition reports for the issues identified in these reports. Included in this assessment was a review of approximately 2400 condition reports, of which 222 were identified for further evaluation. Of the 222 condition reports identified for further evaluation, 197 were related to labeling issues and only 6 were classified as significant conditions. Other than a tornado venting issue, which had been identified prior to the self assessment as a significant condition,

none of the new significant conditions had an impact on plant safety. In addition, the heightened sensitivity to configuration control issues resulted in the licensee identifying an increased number of configuration discrepancies. The team concurred with the licensee's conclusion that the majority of events related to configuration control were historical in nature.

The team reviewed the corrective actions and recommendations identified in the configuration control self assessment. The report identified 7 recommendations, which included standardized trending for configuration issues, reinforcement of coaching and counseling for human performance issues, multiple procedural enhancements, amplification of the use of the corrective action program to include configuration issues, and revision of the corrective action program. Additionally, the report drew several conclusions which identified areas for improvement. These conclusions were dispositioned by recommendations to incorporate related items, such as procedure changes, into appropriate programs and procedures.

The team sampled condition reports relating to configuration control that were not identified within the self assessment. The team's review of these condition reports indicated that the screening process used by the licensee was effective and accurate. The team considered the corrective actions for the sampled condition reports to be reasonable for the correction of the configuration control deficiencies.

The team also reviewed the licensee's response to the 10 CFR 50.54(f) letter on design basis issues, dated October 9, 1996. The team found the material contained within this response to be consistent with the application of the configuration control process.

c. Conclusions

Self assessments and reviews to determine root causes and identify improvements with the configuration process were thorough and self critical. Configuration control issues identified prior to the completion of the configuration control self assessment resulted in two non-cited violations that are discussed in Sections E8.1 and E8.2 of this report.

E3 Engineering Procedures and Documentation (37550)

E3.1 10 CFR 50.54(f) Letter Response Review

a. Inspection Scope

The team reviewed specific attributes from the Fort Calhoun Station response to the NRC's 10 CFR 50.54(f) letter regarding adequacy and availability of design basis information. The team's review encompassed the following conditions that had, or were in the process of, being accomplished:

- The status and resolution of the licensee's action plan and open discrepancies identified from their latest review of technical specification surveillance requirements;

- The remaining 200 open items in the licensee's design basis document open item program;

- The status of the licensee's plans and scope of commitment for implementation of the Nuclear Energy Institute methodology for review of all safety systems;

- The results of the findings from the licensee's Nuclear Energy Institute style review of the chemical and volume control system, safety injection system, and auxiliary feedwater system; and,

- The status of configuration control issues.

b. Observations and Findings

The team reviewed the latest technical specification surveillance and testing assessment performed by the licensee. The results of the review indicated that discrepancies identified in the response were closed with the exception of four technical specification related items that were submitted to the NRC for an amendment to the technical specifications.

The team reviewed the status of the 200 open items identified during the licensee's review of design basis documents and entered into the design basis document open item program. The team found that 194 items were closed. The remaining six items were reassigned to the corrective action process for resolution and tracking. These items involved Design Basis Documents PLDBD-CS-52, "Heavy Loads," and SDBD-CONT-5501, "Containment." The reassignment of these six items to the corrective action process closed all design basis document open items. The team concluded that the licensee was effective with the tracking and resolution of open items related to design basis documents.

The team reviewed the licensee's plan and scope of commitment associated with the review of all safety systems in accordance with the Nuclear Energy Institute's process. The team noted that, while the formal plan was not yet established, a resource estimate to complete the plan was complete and plans for completion of the design basis document review were under development. The licensee was on schedule for completion of the design basis document review of all safety-related and safety-significant systems. This plan was scheduled for completion by the February 1999 commitment date that the licensee provided in their response to the 10 CFR 50.54(f) letter.

The team compared the review of the design basis documents to the Nuclear Energy Institute process. The team reviewed self assessments of the licensing basis for the chemical and volume control system, safety injection system, and auxiliary feedwater system. The team noted that the reviews exceeded the requirements contained in the Nuclear Energy Institute recommended process.

The team also noted that the licensee reduced the scope on subsequent design basis documents based upon repetition of reviews conducted in other self assessments or conducted during the review of prior design basis documents. Examples of overlapping reviews were the configuration control self assessment and the review of technical specification interpretations. The team did not identify any safety impact associated with the elimination of repetitive or overlapping reviews from the design basis document self assessment process.

A review of design basis document self assessments indicated that the licensee was identifying discrepancies and deficiencies associated with design documents and that these discrepancies were properly dispositioned utilizing the licensees corrective action program.

The team also reviewed the licensee's efforts to address configuration control issues. Information on these efforts are discussed in Section E2.7 of this report.

c. Conclusions

The licensee resolved or was resolving the issues identified in their response to the NRC 10 CFR 50.54(f) letter. All issues reviewed were found to be completed or properly scheduled for completion. The team concluded that the licensee was utilizing the Nuclear Energy Institute process and effectively dispositioning deficiencies.

E4 Engineering Staff Knowledge and Performance (37550)

a. Inspection Scope

The team interviewed three engineering department managers, one engineering supervisor, four system engineers, one fire protection engineer, six mechanical engineering personnel, and four design engineers. Interview topics included management expectations for staff engineers, training regarding systems interrelations, calculational analysis, and interface with other plant organizations. The team questioned staff engineers on knowledge of their assigned areas and conducted system walkdowns with the system engineers. In addition, the team conducted detailed walkdowns of the auxiliary feedwater system and fire protection system.

b. Observations and Findings

The team found an engineering staff with little change in personnel. Engineers were all long-term employees with the most junior person possessing 11 years experience at the station with 5 of those years in engineering.

Engineers were cognizant of management expectations which were clearly defined and disseminated via supervision to the engineering staff. Expectations for system engineers were issued as engineering Procedure PED-SEI-20, "Duties and Responsibilities of System Engineers." Although Procedure PED-SEI-20 contained a multitude of information, engineers were familiar with the contents and did not find the requirements excessive.

Management encouraged aggressive problem identification and resolution. The team sensed a strong questioning attitude and a sense of ownership among system engineers. System and design engineers were knowledgeable of their assigned systems and duties. System engineers conducted periodic walkdowns of their systems. Additionally, engineers were familiar with the associated engineering backlog, deficiencies, operator workarounds, and emergent maintenance items on their assigned systems.

The team found that engineers were aware of the risk significance of their assigned and interfacing systems. Also, engineers indicated a willingness to consult risk analysis engineering during the development of system outages and modifications. Application of risk models to the day-to-day engineering process was evident.

The team noted that system engineers periodically developed system report cards for their assigned systems. System report cards are managerial level documents which assess the overall health of the system. Management and engineering generally found this to be an effective overview of system condition. Additionally, engineers and management stated that this tool was useful in forcing the engineering staff to assess system operability and material condition in the aggregate.

The team also assessed the familiarity of system engineers with related engineering procedures. Engineers were familiar with the requirements and applications for procedures associated with modifications, temporary modifications, engineering changes, substitute replacement items, and configuration control.

The team interviewed and observed the performance of engineers with respect to their interface with other departments. System and design engineers interfaced regularly and were in communication over many issues. During plant tours engineers demonstrated familiarity with other organizations such as maintenance and operations. Interviews with other plant department personnel indicated a willingness to contact the system engineer when questions arose. Additionally, the team noted that personnel in the instrumentation and control, electrical, and mechanical maintenance departments had a very good working relationship with these engineers.

The team noted that all of the engineers seemed motivated, capable, and well qualified with a strong sense of ownership in the plant and in their individual responsibilities. During the team's review of materials and interviews, the engineering staff appeared attentive to detail and demonstrated a comprehensive understanding of overall plant operations, engineering principles, and regulatory requirements.

c. Conclusions

Engineers were motivated, well qualified, familiar with their assigned and supporting systems, cognizant of system conditions, and versed in engineering procedures. Engineering expectations were effectively communicated and well understood by the engineering staff. Engineering management was effective in establishing a strong engineering work ethic. Engineers interfaced effectively with other plant organizations. The team concluded that the use of system report cards was an effective management tool.

E5 Engineering Staff Training and Qualification (37550)

a. Inspection Scope

The team reviewed the licensee's training and certification program requirements for the engineering staff. This review included a review of training records and interviews with three engineering managers, three system engineers, and one design engineer.

b. Observations and Findings

The team reviewed the training and qualification records of the engineering staff and found all to be qualified both in their assigned and interfacing systems. The team also noted that significant modifications had been made to the engineering training program. Some training program changes included requalification on all systems and enhanced training in procedures. System qualifications required the engineer to review the system with a licensed operator. The engineers indicated that this qualification requirement provided several benefits including better communications with the operations staff, increased credibility with the operations staff, and an operations perspective on system performance.

c. Conclusions

Engineers were qualified both in assigned and interfacing systems. Training for engineers was effective and contained a strong operations interface. The benefit obtained, through the training program interface with operations department, was a strength in the engineering training program.

E6 Engineering Organization and Administration (37550)

a. Inspection Scope

The team evaluated the overall effectiveness of the nuclear safety review group by interviewing group personnel, reviewing selected reports, and determining if issues identified by the safety review group were properly dispositioned.

b. Observations and Findings

The nuclear safety review group is the licensee's independent safety engineering group, which consisted of five engineers and a manager who reported to the Division Manager, Nuclear Assessments. The team interviewed the nuclear safety review group manager and two nuclear safety review specialists. The interviews with the manager and specialists indicated that current management expectations included an aggressive approach to issues. Interviews with nuclear safety review group specialists indicated that recent organizational changes and managerial guidance resulted in an increased focus on safety.

The nuclear safety review group recently conducted a self assessment of their own organization and, as a result, modified procedures to enhance the performance and credibility of the organization. Recent nuclear safety review group efforts associated with boric acid batching tank heaters and the reactor coolant pump motor lube oil collection system indicated that the nuclear safety review group was aggressive in their pursuit of safety-related issues. An example of the nuclear safety review group's strong questioning attitude was the reopening and issuance of additional condition reports associated with the boric acid batching tank heaters.

As the result of these interviews, the team independently reviewed and assessed the nuclear safety review group actions for these two issues. The team found that the issues identified by the NRC involving the reactor coolant pump motor lube oil collection system were similar to those identified by the nuclear safety review group. In addition, as the result of the review of condition reports associated with the boric acid tank heaters, the team found that the nuclear safety review group identified that the heaters had insufficient capacity. Furthermore, the team noted that the nuclear safety review group persisted in their followup of this issue to assure that heater capacity was restored. The team noted that, with the exception of the completion of some minor documentation, all actions involving these heaters were completed.

c. Conclusions

The team concluded that the nuclear safety review group was aggressive in their approach to safe plant operations. Recent self assessments and management changes resulted in improved performance and credibility with plant organizations.

E7 Quality Assurance in Engineering Activities (37550)

E7.1 Quality Assurance Surveillances and Audits

a. Inspection Scope

The team reviewed 11 quality assurance surveillance reports and 1 quality assurance audit on plant engineering. These reports were reviewed to evaluate the licensee's effectiveness to self identify and resolve plant problems.

b. Observations and Findings

The team found that the licensee was self critical with the application of quality assurance audits and surveillances. Quality assurance surveillances included a sufficient sampling of items necessary to meet the scope of the surveillance. Corrective actions and recommendations were appropriate for the deficiencies identified during the surveillance process.

The team noted that there were 25 surveillances and audits of engineering activities during the last 2 years. The team found that the surveillances covered diverse topical areas and reflected similar findings to those identified by the team. An example of the licensee's effectiveness with identifying quality related issues was demonstrated in Surveillance Report E-96-4. In this report, the licensee identified discrepancies associated with cable tray locations and entered these items into the corrective action system for tracking and resolution.

c. Conclusions

Quality assurance audits and surveillances reflected the proper level of detail and focused attention in areas of safety significance.

E7.2 Self Assessments

a. Inspection Scope

The team reviewed various self assessments to assess the depth and critical nature of internal assessments of engineering and related activities.

b. Observations and Findings

In addition to those self assessments discussed in other sections in this report, the team reviewed an additional nine self assessments.

These nine self assessments were conducted by the nuclear safety review group. The process included interviews, document reviews, and observation of activities. Nuclear safety review group recommendations were assigned and tracked within their own tracking system. Issues warranting corrective action were entered into the condition reporting system. The team found that nuclear safety review group findings were being properly resolved.

c. Conclusion

Self assessments addressed areas of safety significance. The methodology for development of self-assessment activities was sound and drew proper conclusions and effective corrective actions.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item 50-285/9618-02: Failure to follow configuration change control Procedure PED-QP-2 for: (1) replacing springs and spiral pins on main steam line radiation monitor isolation valves; (2) installing an actuator cylinder on a component cooling water outlet valve; and (3) installing a gasket on the safety injection and refueling water tank vent.

(Closed) Unresolved Item 50-285/9703-01: Failure to follow a preventive maintenance order to replace a needle spring on the turbine-driven auxiliary feedwater pump control relay.

Background - These two items were identified as unresolved items because of the continuing NRC concerns with the adequacy of the licensee's configuration control program. In both cases, maintenance personnel failed to follow plant procedures resulting in a loss of configuration control for applicable plant components.

On November 22 through December 10, 1996, the NRC conducted a special inspection (NRC Inspection Report 50-285/96-17) on the post-accident sampling system. As a response to the enforcement actions which resulted from this inspection, the licensee committed to perform a self assessment of their configuration control practices. The licensee committed to complete this self assessment by April 30, 1997, and send the report to the NRC by May 30, 1997.

Inspection Followup - Both of these unresolved items were licensee identified and involved the failure to follow plant procedures. As a result of this procedure adherence problem, plant configuration control was lost. A review of the licensee's self assessment of the configuration control process (discussed in Section E2.7 of this inspection report) concluded that the licensee had aggressively pursued the resolution of the configuration control issues. Specifically, with respect to these procedure adherence issues, the licensee had planned coaching and counseling for personnel and procedure enhancements to provide clarification.

The failure to follow procedures and instructions was considered to be a violation of 10 CFR Part 50, Appendix B, Criterion V. However, the team noted that the licensee's corrective actions for an escalated enforcement action (50-285/EA96-489) that could have prevented this occurrence were in progress and were not yet completed when these items were identified. Therefore, these violations could not reasonably be expected to have been prevented by the licensee's corrective actions for a previous violation or a previous licensee finding. In addition, the licensee had documented the correction of these violations in the corrective action program. Based on these findings, these licensee-identified and corrected violations are being treated as noncited violations, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-285/9706-04).

E8.2 (Closed) Unresolved Item 50-285/9703-02: Failure to follow an engineering change notice and maintenance work document for the replacement of a mechanical seal on a spent fuel pool cooling pump and a demineralized water transfer pump.

Background - Although the two spent fuel cooling pumps and the two demineralized water transfer pumps were identical, they had different shaft flinger ring configurations. Maintenance personnel failed to follow the instructions of engineering change notices and maintenance work documents during the replacement of the pump seals. This item was identified as an unresolved item because of the continuing concerns by the NRC with the adequacy of the licensee's configuration control program.

Inspection Followup - On November 22 through December 10, 1996, the NRC conducted a special inspection (NRC Inspection Report 50-285/96-17) on the post-accident sampling system. As a response to the enforcement actions which resulted from this inspection as well as internal concerns associated with other configuration control deficiencies, the licensee committed to perform a self assessment of their configuration control practices. The licensee committed to complete this self assessment by April 30, 1997, and send the report to the NRC by May 30, 1997.

The spent fuel cooling pump discrepancy was identified by the NRC, while the demineralized water transfer pump discrepancy was identified by the licensee. As a result of this procedure adherence problem, plant configuration control was lost. A review of the licensee's self assessment of the configuration control process (discussed in Section E2.7 of this inspection report) concluded that the licensee was aggressive in their approach to the resolution of the configuration control issues. Specifically, with respect to these procedure adherence issues, the licensee planned coaching and counseling for human performance issues and procedure enhancements to provide clarification.

The failure to follow procedures and instructions was considered to be a violation of 10 CFR Part 50, Appendix B, Criterion V. However, the team noted that the licensee's corrective actions for an escalated enforcement action (50-285/EA96-489) that could have prevented this occurrence were in progress and were not yet completed when these items were identified.

The team further noted that the configuration control concern was a flinger ring left installed on one of the spent fuel cooling pumps and one of the demineralized water transfer pumps during replacement of the seals with a new type seal. Since the new seal design did not require such a ring, the installed ring had no effect on the operation of the pumps and, therefore, would not become a more significant safety and regulatory concern. Therefore, the violation did not have any actual or potential impact on safety. In addition, while this violation did suggest a programmatic problem with procedure adherence by maintenance personnel, this problem was being corrected and there was minor safety or regulatory impact.

The team concluded that, while this was a violation of 10 CFR Part 50, Appendix B, Criterion V, it constituted a violation of minor significance and is being treated as a noncited violation, consistent with Section IV of the NRC Enforcement Policy (50-285/9706-05).

F1 Fire Protection Program (64704)

a. Inspection Scope

The team inspected the licensee's fire protection program to verify that the licensee had properly implemented and maintained the fire protection program required by the operating license. The team reviewed fire protection procedures, administrative controls, quality assurance findings, fire brigade qualifications, and fire brigade staffing in accordance with the approved fire protection program. The team also conducted extensive walkdowns of the facility to verify licensee implementation of the approved fire protection program.

b. Observations and Findings

During the inspection, the team noted that most administrative controls were properly implemented and that most administrative control procedures were adequate. In addition, the firewatch personnel were well qualified, plant housekeeping was adequate for control of transient combustible materials, and the station fire response equipment was well maintained. However, the team also identified the following examples where the fire protection program was not adequately implemented.

b.1 Alternate Safe Shutdown Procedure AOP-06

During the plant walkdowns, the team observed that the redundant electrical cables for plant safe shutdown equipment were in close proximity to each other in the cable spreading room. 10 CFR Part 50, Appendix R, Section III.G.2, provides separation criteria for redundant safe shutdown trains in the same fire area. The licensee did not meet the Appendix R, Section III.G.2, cable separation in the cable

spreading room, which was directly below the control room. 10 CFR Part 50, Appendix R, Section III.G.3, requires that if Section III.G.2 criteria cannot be met then alternative or dedicated shutdown capability be provided. For this reason, the licensee's Fire Hazards Analysis took credit for alternative shutdown capability for a fire in this area.

The team reviewed Procedure AOP-06, "Fire Emergency." This procedure provided the operators' direction on implementing the alternative shutdown method for a fire in either the control room or cable spreading room. During this review, the team noted that the procedure did not direct the operators to implement alternate shutdown for a cable spreading room fire, in that, the procedure did not provide any criteria for the operators to use to determine when a control room evacuation should be made during a cable spreading room fire. Specifically, Procedure AOP-06, Section 4.0, Step 7, directed that if a control room evacuation is required, the operators should evacuate the control room by performing additional steps in the procedure. These steps did not provide the guidance necessary to ensure a timely control room evacuation for a cable spreading room fire. This was inconsistent with other steps of the procedure where the operators were provided guidance regarding actions to be taken to mitigate other plant fires that did not involve control room evacuation.

A subsequent interview with one shift supervisor indicated that the shift supervisor would not evacuate the control room until the control room became uninhabitable. The team considered that such an extended control room evacuation delay could threaten the ability to achieve alternative shutdown. Additional interviews with operations management indicated that management expectations were that the operators would judge fire severity in the cable spreading room before evacuating the control room.

10 CFR Part 50, Section III.L.3, requires that procedures shall be in effect to implement the alternative shutdown capability. The failure to have a procedure to provide guidance regarding alternative shutdown implementation for a cable spreading room fire is considered to be an apparent violation of 10 CFR Part 50, Appendix R, Section III.L.3. This is considered to be the first apparent violation of fire protection program requirements (50-285/9706-06).

On April 18, 1997, the licensee issued a memorandum to operations department personnel to inform the operating crews, as an interim action until the procedure can be revised, about the necessity of implementing alternative shutdown for a cable spreading room fire. At the conclusion of the inspection, the licensee's engineering staff was evaluating the issue to determine the most effective method of ensuring that alternate safe shutdown can be implemented and to prevent a premature control room evacuation.

b.2 Diesel Generator B Not Electrically Isolated for Control Room Fire

Subsequent to the onsite team inspection, while evaluating the needed guidance for the operators to respond to a fire in the cable spreading room, the licensee identified that the Diesel Generator 2 engine tachometer circuit was not protected from the effects of a control room or cable spreading room fire. As a result, these potential fires could damage the tachometer circuit and render the diesel speed sensing circuit inoperable. If the speed sensing circuit was inoperable, the diesel would fail to start and could not be started or operated locally.

The licensee issued Condition Report 199700523 to provide an operability evaluation for this condition. In addition, the team discussed the condition and corrective actions with licensee personnel. The licensee's report identified that the output of speed sensing circuit, YE-6148-DG-2, was not isolated from the control room. Therefore, a fire could cause 120 AC voltage or 125 DC voltage to be induced on the tachometer circuit and render the diesel speed sensing circuitry inoperable. In addition, since this diesel generator was protected for use as the AC power supply in the event of a control room or cable spreading room fire, the diesel generator was required to achieve safe shutdown conditions.

The speed sensing circuit's function was to disengage the air start motors, open the diesel generator outside air dampers, flash the generator field, and to provide engine speed indication at the local panel and the control room. Failure of this circuit would prevent the diesel from automatically starting and would prevent local control due to lack of speed indication for the operator. In the event of a control room or cable spreading room fire requiring alternative shutdown, the operators were required to operate Diesel Generator 2 locally.

The team noted that this condition existed since the original design of the facility and was not found during the initial safe shutdown analysis to comply with the 10 CFR Part 50, Appendix R, requirements, which was conducted during the early 1980's.

10 CFR 50.48(b) requires that all nuclear power plants licensed to operate prior to January 1, 1979 satisfy the requirements of Sections III.G, III.J, and III.O of Appendix R to 10 CFR Part 50. Fort Calhoun Station was licensed on August 9, 1973. 10 CFR Part 50, Appendix R, Section III.G.1.a, requires that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or alternate shutdown be free of fire damage. This is considered to be a second apparent violation of fire protection program requirements (50-285/9706-07).

b.3 Fire Barrier Separating Fire Areas 6 and 20

During a plant walkdown, the team noted that the stairwell opening between the grade and basement levels was also between Fire Areas 6 and 20 in the auxiliary building. This stairwell opening had a deluge system installed around the lower part of the opening rather than a "water curtain" as approved in the licensee's fire

protection program. The system consisted of six pendant heads evenly spaced with two on each side, approximately 5 feet apart and one on each end of the opening in the center. There was no draft curtain installed around the stairwell opening. There also was no area sprinkler system installed on either level of the auxiliary building in these fire areas. This deluge system would actuate when two smoke detectors, located in the area of the opening on the ceiling of the basement level, detected smoke. The lower portion of the opening would then be filled with a water spray.

The team also noted by reviewing the licensee's fire hazards analysis that Fire Area 6 contained cables for the low pressure safety injection pumps and high pressure safety injection pumps. In addition, Fire Area 20 contains charging pump cables and the charging pumps. Therefore, a single fire could damage both trains of reactor makeup water and prevent achieving and maintaining hot shutdown conditions. In addition, licensee personnel stated that most of the other equipment and cables needed to achieve safe shutdown conditions were also in one of these fire areas.

National Fire Protection Association, National Fire Code 13, Section 4-4.8.2.3, requires that large unenclosed floor openings be protected by draft stops in combination with closely spaced sprinklers. The code then describes the design requirements for a "water curtain." These design requirements involve high density water spraying against a draft stop to form a water curtain around the opening. The code provides a . exception for large openings where both floors are protected by an automatic sprinkler system.

The team reviewed licensee submittals dated December 12, 1979, January 18, 1980, and May 20, 1980; NRC Safety Evaluation Reports dated August 23, 1978, and November 17, 1980; National Fire Protection Association, National Fire Code 13, "Standard for the Installation of Sprinkler Systems," and licensee evaluations dated April 17 and May 1, 1977, to determine if the installed configuration was previously approved by the NRC and if the installation was adequate to protect redundant safe shutdown equipment from damage from a single fire.

A review of these submittals and the NRC Safety Evaluation Reports indicate that a "water curtain" was to be installed around this opening. No exception to National Fire Protection Association, National Fire Codes was mentioned in any of the referenced documents concerning the design of the water curtain. The licensee informed the team that, since the system's installation in 1980, they considered this configuration to be a "water curtain". In addition, the licensee considered this configuration to be previously approved by the NRC.

Based on the team's questions, the licensee performed an additional analysis to justify the acceptability of the installation. This analysis, dated May 1, 1997, concluded that the water curtain design was adequate. However, review of this analysis by the team and the NRC program office, concluded that the license's installation did not constitute a "water curtain" as described by the National Fire Code.

10 CFR Part 50, Appendix R, Section III.G.1 requires that Fire protection features shall be provided for structures, systems, and components important to safe shutdown. In addition, it requires that these features shall limit fire damage so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or alternate shutdown station was free of fire damage.

The failure to install a "water curtain" as approved in the licensee's fire protection program, resulted in a lack of protection against a single fire from damaging both trains of hot shutdown equipment required to achieve hot shutdown. This is considered to be an apparent violation of 10 CFR Part 50, Appendix R, Section III.G.1 and is a third apparent violation of the fire protection program requirements (50-285/9706-08).

The licensee did not agree with this apparent violation. The licensee stated that fire barriers, allowed by 10 CFR 50.48(a) and previously approved by the NRC, were acceptable. However, all licensing documentation presented to and reviewed by the team indicated that a "water curtain" would be installed around this opening and that this "water curtain" would be designed as described by National Fire Protection Association, National Fire Code 13. None of the documentation provided a description of the actual configuration of the installed "water curtain." Therefore, the NRC did not have an opportunity to review and accept the installed configuration. The inspectors concluded that the "water curtain" currently installed at the station does not meet these fire protection requirements and, as such, is not an adequate installation.

b.4 Reactor Coolant Pump Motor Lube Oil Collection System

Reactor coolant pump motor lube oil leakage from the Reactor Coolant Pump D motor was identified in 1995 by the licensee. During the 1996 outage, NRC inspectors noted that this leakage was continuing from the Reactor Coolant Pump D motor. As a result of this observation, a December 20, 1996, conference call was held between the NRC and the licensee. This call raised NRC concerns regarding the adequacy of the pump motor lube oil collection system.

The licensee responded to the NRC's concerns by letter dated February 7, 1997. In that letter, the licensee stated that they believed that their lube oil collection system was already approved by the NRC and that they had an exemption from having a collection system for the low pressure components. Furthermore, the letter provided details of the licensee's compensatory measures for continued operation with uncollected leakage from the Reactor Coolant Pump D motor. The NRC staff reviewed the licensee's response and did not agree that the NRC granted an exemption for collection of low pressure component oil leakage and did not find the licensee's compensatory measures to be adequate. The NRC staff also did not find the licensee's corrective actions to be appropriate. These conclusions were provided to the licensee in a telephone conference call prior to the inspection and the licensee implemented additional compensatory measures.

During this inspection, the team further reviewed the licensee's reactor coolant pump motor lube oil collection system. This review included a review of the licensee's compensatory measures for continued plant operation with lube oil collection system deficiencies. In addition, because of a forced shutdown due to a steam line rupture, the team had the opportunity to perform a visual inspection of the reactor coolant pump motor lube oil collection system.

During a walkdown in the containment building on April 29, 1997, the inspection team observed that the upper reservoir level transmitter and three resistance temperature detectors for the Reactor Coolant Pump A, C, and D motors did not have any lube oil collection system. The team also noted that the three drain ports for the shaft air seal and the lower reservoir resistance temperature detector and level instrument for each pump also did not have a lube oil collection system. Furthermore, the team noted that the high pressure collection system for these pump motors was degraded such that all oil leakage would not be collected by the existing oil collection system. The team noted that the Reactor Coolant Pump B motor was of a different design than the other pump motors and, therefore, this motor had different lube oil collection requirements. The team noted that this motor also did not have a lube oil collection system for the lower bearing reservoir level indication instrumentation, sight glass, and resistance temperature detector.

As a result of walkdowns in the lower level of the containment building, the team observed lube oil on reactor coolant system insulation near the "B" reactor coolant pump motor. The team noted that this finding was observed a week after the plant had been shutdown and after the licensee's staff had conducted two inspections of the area for evidence of lube oil.

As corrective action prior to restarting from the forced outage, the licensee installed a lube oil collection system for the "A", "C", and "D" reactor coolant pump motor upper drain ports and the "B" reactor coolant pump motor lower bearing instrumentation. This installation provided adequate collection of the lube oil that was actually leaking from the "D" reactor coolant pump motor. In addition, the licensee repaired the degraded seals on the high pressure lube oil collection system.

On April 30, 1997, the licensee proposed compensatory measures to the team to allow operation with a deficient lube oil collection system. The compensatory measures allowed the licensee to safely operate the unit while evaluating additional corrective actions for the collection of leakage from low pressure components. The inspection team, in consultation with the program office, concluded that these initial compensatory measures lacked specificity and were not acceptable. Therefore, the licensee provided new compensatory measures on May 2, 1997, which contained the appropriate level of detail concerning how the operators would monitor the condition of the lube oil collection system and the compensatory actions to be taken. These compensatory measures included closely monitoring oil levels and specifying actions for the operators to take if levels changed by specified amounts.

10 CFR Part 50, Appendix R, Section III.O, requires reactor coolant pumps to be equipped with a lube oil collection system capable of collecting lube oil from all pressurized and unpressurized leakage sites on the reactor coolant pump motors. The failure to have an adequate lube oil collection system on the reactor coolant pump motors is considered to be a fourth apparent violation of fire protection program requirements (50-285/9706-09).

b.5 Fire Suppression Water System Flushing

In 1977, when proposing the technical specifications for the fire suppression water system, the licensee included a statement that the system would be operable during flushing operations. The NRC approved the proposed technical specifications in June 1977, but did not agree (nor approve) that the system was operable during flushing operations.

During flushing operations, the licensee placed both fire suppression water pumps in the "pull-to-lock" position. This action disabled automatic starting of the pumps, as described in the fire protection program. The fire suppression water system was then connected to the potable water supply to flush the system of silt, which might accumulate in the piping since the normal water supply to the suppression system was from the Missouri River.

The potable water system was not designed to supply an adequate amount of water to meet system requirements during a fire. In addition, the plant review committee, when reviewing Technical Specification Interpretation 90-01, in 1990, noted that the NRC's intent was that the system be considered inoperable when performing flushing operations and that the appropriate technical specification limiting condition for operation be entered.

In 1994, the licensee removed the fire protection requirements from the technical specifications and placed them in the Updated Safety Analysis Report. On August 8, 1996, the licensee issued Standing Order SO-G-103, "Fire Protection Operability Criteria and Surveillance Requirements," which allowed the operators to place the pumps in the "pull-to-lock" position without declaring the system inoperable. Declaring the pumps inoperable would require the stationing of continuous firewatches during flushing operations.

While reviewing the licensee's control of the fire suppression water pumps, the team noted that on November 10, 1996, and on February 10, 1997, the licensee performed fire suppression water flushing operations. While these operations rendered the fire suppression water system inoperable, the team noted that the operators did not declare the fire suppression system inoperable or station firewatch personnel to compensate for the inoperable suppression systems.

License Condition 3.F of Facility Operating License DPR-40, for the Fort Calhoun Station, requires implementation of the approved fire protection program, as described in the Updated Safety Analysis Report for the facility. Table 11.2 of the Updated Safety Analysis Report states that the fire suppression water system shall

be operable, except during system testing, jockey pump maintenance, or training. Table 11.2 also requires that a continuous firewatch be established for the diesel generator rooms, the compressor room, and in the corridor between Fire Areas 6 and 20 when the suppression systems are inoperable. The failure to declare the fire water suppression system inoperable and to establish firewatches is considered to be a fifth apparent violation of fire protection program requirements (50-285/9706-10).

c. Conclusions

The team considered the implementation of the fire protection program to be poor. The team identified five examples of apparent failures to properly implement the fire protection program. These examples were considered to be apparent violations of fire protection program requirements.

F2 Status of Fire Protection Facilities and Equipment (64704)

a. Inspection Scope

The team performed a walkdown of all areas of the facility containing safe shutdown equipment. These walkdowns encompassed the fire detector system in Fire Area 32 and the air compressor room, which contained both auxiliary feedwater pumps. The team also visually inspected fire protection equipment located throughout the facility and visually inspected fire brigade response equipment located in the turbine building storage locker outside of the control room.

The team also randomly selected components required for post control room fire safe shutdown by Procedure AOP-06, which could be required for safe shutdown during a control room fire, to verify that they were accessible, well labelled, and had adequate emergency lighting to perform required tasks.

b. Observations and Findings

The team observed that all fire response equipment was well maintained, accessible, within calibration, and in good working order. All valves observed by the team in the fire suppression system were in their proper position. Fire water pumps and equipment were operable and well maintained.

The fire detectors in Fire Area 32 were installed in accordance with National Fire Protection Association, National Fire Code 72E, "Automatic Fire Detectors." The licensee tested the control room fire alarm for the team. The team considered that the alarm would alert the operators to a plant fire and would isolate the location of the alarming detector for the operators. The licensee's plant computer kept a record of all alarms and thus allowed event reconstruction.

The team noted that all fire brigade response equipment located in the turbine building storage locker was well maintained and ready for immediate use.

The team also noted that randomly selected components which were required for use in Procedure AOP-06 were well labelled, accessible, and had adequate emergency lighting available.

During the plant walkdown, in Fire Area 32, the team noted a sprinkler nozzle over the steam-driven auxiliary feedwater pump which was partially blocked by a security cage. The licensee wrote Condition Report 199700410 to document and to correct the deficiency. The licensee's evaluation concluded that the sprinkler was degraded but not inoperable. The team agreed with the licensee's evaluation.

c. Conclusions

Fire protection equipment required for program implementation was well maintained and available for immediate use. The team considered the fire detection and alarm capability to be good.

F3 Fire Protection Procedures and Documentation (64704)

a. Inspection Scope

The team reviewed the licensee's approved program as defined in the Updated Safety Analysis Report for the facility. The team reviewed the procedures listed in the attachment to this report to verify that the procedures adequately implemented the licensee's approved program.

b. Observations and Findings

The team found that, with the exceptions noted in Section F1 of this report, the procedures reviewed properly implemented the program as approved by the NRC.

c. Conclusions

With the exception of the procedure inadequacies identified in Section F1 of this report, the team determined that the fire protection program procedures adequately implemented the approved fire protection program.

F4 Fire Protection Staff Knowledge and Performance (64704)

a. Inspection Scope

The team reviewed the adequacy of the fire protection engineering staff by interviewing engineers and conducting plant walkdowns with staff members.

b. Observations and Findings

Discussions with the engineers indicated that they understood NRC requirements for the fire protection program and the National Fire Protection Association, National Fire Code requirements. They also demonstrated a detailed understanding of fire hazards associated with the station and a detailed knowledge and understanding of the systems, testing, and analysis associated with the fire protection program.

During the walkthrough inspection, the team observed discussions, regarding the program, between the fire protection staff and various other site personnel. The team observed that the fire protection personnel had a very good working relationship with other onsite organizations.

The team noted that the fire protection program discrepancies, identified during this inspection, were the result of the 10 CFR Part 50, Appendix R, analysis performed by the licensee during the early 1980's. The problems predate the current engineers and were not considered to be problems which would be identified during routine program duties.

c. Conclusions

The plant had a qualified fire protection staff which had a very good working relationship with other station organizations.

F5 Fire Protection Staff Training and Qualification (64704)

a. Inspection Scope

The team reviewed the readiness of fire brigade personnel to fight fires. The review included fire brigade composition, qualifications, and training. The team also reviewed the fire brigade training records to determine if the fire brigade was being trained in accordance with the fire protection plan.

b. Observations and Findings

The fire brigade consisted of five individuals per shift. In addition, the fire brigade leader was a licensed operator and had sufficient knowledge of safety-related systems to understand the effects of fire and fire suppressants on plant safe shutdown capability.

All personnel were trained in an initial training class and subsequent requalification classes. Each member was required to participate in at least one drill annually. In addition, the licensee performed quarterly fire drills. The team verified that the required annual physical examinations were performed on all fire brigade members. The team also verified that the licensee's program required that all classroom training be repeated over a 2-year period that was implemented to meet license condition 3.F.

As a result of this review effort, the team identified that one of the required training sessions, Lesson 1604-03, "Fire-Brigade Equipment," was not conducted since the April through June training cycle in 1994 for qualified fire brigade members. The team determined that eight out of the nine randomly selected fire brigade members had not received the required training. During the inspection, the licensee informed the team that while they did not meet their fire protection program required classroom training, they had conducted practical training under other programs such as the health physics respiratory program. Therefore they considered their fire brigade personnel qualified to perform their fire brigade duties. The team verified selected portions of the other training requirements (e.g., fire extinguisher training) was conducted. As the result of this independent verification, the team concurred with the licensee's conclusion.

License Condition 3.F requires that all provisions of the approved fire protection program described in the August 23, 1978, Safety Evaluation Report, be implemented. Section 3.5 of the August 23, 1978, Safety Evaluation Report required that fire brigade classroom training be repeated every 2 years. This is considered to be a sixth apparent violation of fire protection program requirements (50-285/9706-11).

Fire brigade personnel were also interviewed by the team to determine if they were knowledgeable of the fire brigade program requirements, specific locations of safety equipment in the plant, and understood the effects of fire on the safe shutdown capability of the plant. Each fire brigade member stated that they were not assigned duties which would interfere with the ability to respond to a fire. The team questioned three fire brigade members and one fire brigade leader concerning the use of water on an electrical cable fire. The team found that the fire brigade members were deficient in their knowledge of such fires when they stated that water would be used on cable fires only as a last resort. The fire brigade leader, however, had a good understanding of the need to use water to extinguish electrical cable fires. Subsequent to the interviews with the fire brigade members, the licensee issued further training information to all fire brigade members which provided emphasis concerning the use of water to extinguish electrical cable fires.

c. <u>Conclusions</u>

The team considered the fire brigade training to be adequate to meet NRC requirements. However, the team identified a weakness in fire brigade member knowledge concerning the use of water to suppress an electrical cable fire. In addition, one deficiency regarding the training program was identified and was considered to be a sixth apparent violation of fire protection program requirements.

F6 **Fire Protection Organization and Administration (64704)**

a. Inspection Scope

The team reviewed the fire protection organization designated to implement the fire protection program.

b. Observations and Findings

The fire protection organization was described in Standing Order, SO-G-102, "Fire Protection Program Plan." This plan detailed staffing and responsibilities for the implementation of the program. The team noted that the program was implemented according to the approved program.

c. Conclusions

The team found that the fire protection organization and administration was implemented in accordance with the fire protection program.

F7 **Quality Assurance in Fire Protection Activities (64704)**

a. Inspection Scope

The team reviewed the 1994, 1995, and 1997 quality assurance audits to verify that the audits met the requirements of the approved fire protection program.

b. Observations and Findings

Section 3.7 of the licensee's Fire Hazard's Analysis contained the fire protection quality assurance requirements. Section 3.7.11 contained the audit requirements.

The team found that the program required annual audits which audited design and procurement documents, procedures, surveillances, and test activities. The team noted that the audits accomplished the minimum program requirements. The team also noted that the audits did not identify the problems identified in this report because the audits reviewed the implementation of program procedures but did not review the early 10 CFR Part 50, Appendix R, compliance assumptions.

c. Conclusions

The team found the fire protection program audits to be in compliance with the minimum requirements of the program.

F8 Miscellaneous Fire Protection Issues (92903)

F8.1 (Closed) Unresolved Item 50-285/9616-01: Observation of lube oil leakage from Reactor Coolant Pump Motor RC-3B.

Background - During the 1996 outage, the NRC resident inspectors identified that oil leakage from the Reactor Coolant Pump D motor had accumulated inside the containment. As a result of this finding, a December 20, 1996, conference call was held between the NRC staff and the licensee. This call resulted in NRC concerns regarding the adequacy of the reactor coolant pump motor lube oil collection system. This item was considered unresolved pending resolution of the NRC's questions. The licensee responded to the NRC's concerns by letter dated February 7, 1997.

Inspection Followup - The team reviewed the adequacy of the reactor coolant pump motor lube oil collection system as documented in Section F1.b.4 of this report. For the reasons discussed in that section, the failure to have an adequate reactor coolant pump motor oil collection system was considered to be an apparent violation of 10 CFR Part 50, Appendix R, Section III.O.

V. Management Meetings

X1 Exit Meeting Summary

The team presented the inspection results to members of licensee management at the conclusion of the inspection on May 2, 1997. The licensee acknowledged the findings presented. In addition, the licensee stated that they disagreed with the NRC regarding the design adequacy of their water curtain discussed in Section F.1.b.3 of the inspection report. They considered their design to be adequate and to be previously approved by the NRC.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No information supplied to the team was considered to be proprietary.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Cavanaugh, Licensing
J. Chase, Plant Manager
S. Chomos, Fire Protection System Engineer
R. Conner, Manager, Training
K. Erdman, Fire Protection Design Engineer
C. Fritts, Auxiliary Feedwater System Engineer
S. Gambhir, Division Manager, Engineering and Operation Support
J. Gasper, Manager, Nuclear Projects
G. Gates, Vice President, Nuclear
R. Jaworski, Manager, Design Engineering
E. Jun, Mechanical Design Engineer
J. MacKinnon, Chairman, Safety Audit and Review Committee
R. Mueller, Supervisor, Instrumentation and Control
R. Phelps, Manager, Station Engineering
J. Ressler, Senior Nuclear Design Engineer
A. Richard, Supervisor Mechanical Systems (Design Engineering)
R. Ridenoure, Operations Supervisor
R. Short, Manager, Operations
J. Tills, Manager, Nuclear Licensing
D. Trausch, Manager Nuclear Safety Review
R. Wylie, Manager, Nuclear Construction Management

LIST OF INSPECTION PROCEDURES USED

IP 37001	10 CFR 50.59 Safety Evaluation Program
IP 37550	Engineering
IP 64704	Fire Protection Program
IP 92903	Followup - Engineering

LIST OF ITEMS OPENED AND CLOSED

Opened

50-285/9706-01	VIO	Failure to translate design criteria into plant operations, procedures, and specifications.

50-285/9706-02	IFI	Review the licensee's determination of the effect of high pressure air on safety-related, air-operated valves.
50-285/9706-03	VIO	Failure to update the Updated Safety Analysis Report to provide actual intent of diesel-driven auxiliary feedwater pump fuel oil day tank capacity.
50-285/9706-04	NCV	Failure to follow configuration control procedures.
50-285/9706-05	NCV	Failure to maintain configuration control on plant equipment.
50-285/9706-06	APV	Failure to have an adequate alternative shutdown procedure.
50-285/9706-07	APV	Failure to have a protected train for alternate shutdown that was free from fire damage.
50-285/9706-08	APV	Failure to have an adequate water curtain fire barrier.
50-285/9706-09	APV	Failure to have an adequate reactor coolant pump motor lube oil collection system.
50-285/9706-10	APV	Failure to declare the fire water suppression system inoperable and to establish firewatches during flushing operations.
50-285/9706-11	APV	Failure to conduct all fire brigade classroom training every two years.

<u>Closed</u>

50-285/9616-01	URI	Review adequacy of the licensee's reactor coolant pump motor lube oil collection system.
50-285/9618-02	URI	Failure to follow configuration change control procedure.
50-285/9703-01	URI	Weak configuration control during maintenance on auxiliary feedwater control relay.
50-285/9703-02	URI	Spent fuel pool pump and demineralized water transfer pump configuration differences.
50-285/9706-04	NCV	Failure to follow configuration control procedures.
50-285/9706-05	NCV	Failure to maintain configuration control on plant equipment.

LIST OF DOCUMENTS REVIEWED

Plant Procedures

Procedure	Revision	Title
AOP-06	4.2	Fire Emergency
OI-FP-1	25	Fire Protection System Water System
OI-RC-9	31	Reactor Coolant Pump Operation
OP-PM-FP-1000	2	Quarterly Fire Protection Drain Valve Flush and Alarm Test
PED-GEI-3	18	Preparation of Design Change Packages
PED-GEI-29	4	Facility Change Evaluation
PED-GEI-35	1	Preparation of Engineering Assistance Requests for Minor Configuration Changes/Replacement
PED-GEI-52	0	Preparation of Field Design Change Requests
PED-GEI-60	6	Substitute Replacement Item Evaluations
PED-SEI-20	0	Duties and Responsibilities of System Engineers
PED-EEI-1	4	Human Factors Engineering Instruction
PED-QP-1	8	Engineering Assistance Requests
PED-QP-24	24	Configuration Change Control
PED-QP-13	3	Design Basis Document Control
NSRG-1	8	Nuclear Safety Review Group Charter
MD-AD-0007	0	Bolting
SS-ST-MS-3003	1B	Main Steam Isolation Check Valve Inspection
SE-ST-AFW-3006	14	Auxiliary Feedwater Pump FW-10, Steam Isolation Valve and Check Valve Tests
OP-ST-AFW-0004	12	Auxiliary Feedwater Pump Operability Test
SE-ST-AFW-3005	10	Auxiliary Feedwater Pump FW-6 and Check Valve Test
IC-CP-01-1183	13	Calibration of Emergency Feedwater Storage Tank FW-19, Loop L-1183
IC-CP-01-1188	11	Calibration of Emergency Feedwater Storage Tank FW-19, Loop L-1188
OP-PM-AFW-0004	9	Third Auxiliary Feedwater Pump Operability Verification

ARP-CB10,11/A9	11	Annunciator Response Procedure A9 Control Room Annunciator A9
ARP-AI-66A/A66A	5	Annunciator Response Procedure A66A Control Room Annunciator A66A, AFWAS/DSS
ARP-AI-66B/A66B	6	Annunciator Response Procedure A66B Control Room Annunciator A66-B, AFWAS/DSS
ARP-AI-30A/A33-1	8	Annunciator Response Procedure A33-1 Control Room Annunciator A33-1 Engineered Safeguards
OP-ST-AFW-0004	12	Auxiliary Feedwater Pump Operability Test
OI-DW-3	26	Condensate Storage Tank Operations
OI-AFW-4	24	FO-37 Fuel Oil Transfer Pump Operation
IC-ST-AFW-3002	1	Instrument Air Accumulator/Check Valve Operability Test
IC-ST-AFW-3001	2	Accumulator Check Valve Test for Auxiliary Feedwater Pump Minimum Flow Recirculation Valves
SS-ST-SI-3018	3	Inspection of Check Valves SI-139 and SI-140
NOD-QP-3	16	10 CFR 50.59 Evaluations
NOD-QP-31	11	Operability and Reportability Determinations
NOD-QP-22	11	Preparation and Approval of a Safety Analysis for Operability
NOD-QP-32	4	Technical Specification Interpretations
NOD-QP-7	17	Facility License Changes
NOD-QP-10	10	NRC Exercise of Enforcement Discretion
PED-QP-5	9	Engineering Analysis Preparation, Review and Approval
PED-QP-2	24	Configuration Change Control

Standing Orders

SO	Revision	Title
SO-R-02	3	Condition Reporting and Corrective Action
SO-G-21	62	Modification Control
SO-O-25	49	Temporary Modification Control
SO-M-101	2	Maintenance Work Control

SO-G-6	5	Housekeeping
SO-G-28	39	Station Fire Plan
SO-G-58	24	Control of Fire Protection System Impairments
SO-G-91	9	Control and Transportation of Combustible Materials
SO-G-102	1	Fire Protection Program Plan
SO-G-103	5	Fire Protection Operability Criteria and Surveillance Requirements
SO-G-107	4	Storage of Transient Equipment and Material to Prevent Seismic Interactions
SO-M-9	21	Fire Protection During Flame Cutting, Grinding, and Welding Operations
SO-O-1	31	Conduct of Operations
SO-O-38	10	Firewatch Duties and Turnover Procedures

Temporary Plant Modifications

TM	Title
TM 96-014	Reactor Coolant Gas Pressure Hi
TM 96-018	Equipment Drain Header Soft Patch
TM 96-022	Containment Low Purge Flow
TM 96-033	In-line Isotopic Post Accident Sampling System Samples
TM 96-039	Install Door Between Railroad Siding and Corridor 26
TM 96-041	Turbine Bearing SLOP Drain Valves
TM 96-042	Component Cooling Water System Relief Valve Setpoint Change
TM 96-044	SI-7B Pressure Gage Replaced with a Plug
TM 96-046	SI-7B Charging Valve Replacement
TM 96-048	CET B9 and B15 Cable Swap
TM 97-002	TE-3123 Jumper and Wiring Change
TM 97-005	Temporary Air Supply for FCV-1904C
TM 97-006	CW-1C Discharge Pipe Leak Repair

Permanent Plant Modifications

Modification	Title
MR-FC-94-020	VA-46 A and B Improved Reliability
MR-FC-92-040	Temporary Compressor for Diesel Generator 1 Starting Air Modification
MR-FC-94-024	Diesel Generator 1 Inlet Damper Upgrade
MR-FC-92-019	New Sectional Valve in Turbine Building Standpipe Loop FP-795
MR-FC-93-022	HCV-746 A/B Replacement
MR-FC-91-009	Spent Fuel Pool Rerack
MR-FC-94-018	Back-Up Raw Water Tie-In for Emergency Feedwater Water Storage Tank in Room 81
MR-FC-88-017	Diesel-Driven Auxiliary Feedwater Water Pump

Drawings

Drawing	Revision	Title
11405-M-10, Sht 2	9	Auxiliary Coolant Component Cooling System Flow Diagram
11405-M-10, Sht 1	65	Auxiliary Coolant Component Cooling System Flow Diagram
11405-M-10, Sht 3	12	Auxiliary Coolant Component Cooling System Flow Diagram
11405-M-10, Sht 4	7	Auxiliary Coolant Component Cooling System Flow Diagram
11405-M-40, Sht 1	33	Auxiliary Coolant Component Cooling System Flow Diagram
11405-M-40, Sht 2	25	Auxiliary Coolant Component Cooling System Flow Diagram
11405-M-40, Sht 3	21	Auxiliary Coolant Component Cooling System Flow Diagram
80048	5	Emergency Feedwater Tank
11405-M-253, Sht 1	33	Steam Generator Feedwater and Blowdown
11405-M-253, Sht 4	21	Steam generator Feedwater and Blowdown
11405-M-254, Sht 2	22	Flow Diagram Condensate

Calculations

FC05692	3	Minimum Net Positive Suction Head Available Calculation for Component Cooling Water Pumps
FC06378	2	Determination of Final N_2 Pressure in Component Cooling Water Surge Tank Following a Component Cooling Water Temperature Transient
EA-FC-95-012	0	Effect of Post-Design Basis Accident Component Cooling Water Temperature Transient on Components
FC05346	1	Accumulator Sizing for Auxiliary Feedwater Control Valves HCV-1107A&B, HCV-1108A&B, HCV-1105, HCV-1106, FCV-1368, FCV-1369
FC05361	2	Auxiliary Feedwater System Calculation (Pump Design and Turbine Drive)
FC05691	2	Air Accumulators Operable Time Requirements
FC05336	0	Determine Differential Head and Suction/Discharge Pipe Sizes for Startup Feedwater Pump
EA-FC-93-032	0	Feed-Rate of Steam Generator Following a Loss of Feedwater
FC05071	0	Emergency Feedwater Storage Tank Level for Pump Total Dynamic Head Calculation
FC06148	1	Auxiliary Feedwater Storage Requirements
FC06111	0	Tank Curve FW-19
FC05336	0	Determine Differential Head and Suction/Discharge Pipe Sizes for Startup Feedwater Pump
FC06537	B	Raw Water Fill Line to FW-19 Friction Loss
FC06024	1	N_2 Cylinder Sizing for HCV-480, 481, 484, 485
FC04286	A	Evaluation Criteria for Threaded Fasteners with Incomplete Thread Engagement
FC06638	0	Capacity of Fuel Oil Tank FO-38
FC06641	1	Analysis of Auxiliary Building Water Curtains

Operability Evaluations

Number	Title
199500216	Operability Evaluation of Safety-Related, Power-Operated Gate Valves (Generic Letter 95-07)
199500383	Operability of Valves HCV-1384, HCV-1386 and Associated Piping and Isolation Valves
199600161	Generic Letter 91-15 Review
199600927	Steam Generator Orifice Plates Design
199700015	Net Positive Suction Head of Safety Injection and Containment Spray Pumps
1997000[illegible]9	Parts Replacement on Solenoid Valves
199700049	Pressure Drop to Main Steam Safety Valves
199700067	Gate Valve Pressure Locking and Thermal Binding
199700168	Missing Local Instrument Air Filters

Condition Reports

Number	Title
199700410	Evaluation of Wet Pipe Sprinkler System for Turbine Driven Auxiliary Feedwater Pump
199700416	Missed Fire Brigade Classroom Lecture
199700495	Missing Tubing Clamps on Reactor Coolant Pump Lube Oil Collection System Drain Lines
199700498	Pull-to-Lock Operability of Fire Suppression Water Pumps
199700523	Diesel Generator 2 Speed Sensing Circuit Not Isolated from Effects of Control Room Fire
199700350	Containment VA-8B Cooling Coil; Component Cooling Water Outlet Valve
199700264	Demineralized Water Surge Tank DW-39; Transfer Pump
199700206	Spent Fuel Pool; Circulating Pump
199700153	Auxiliary Feedwater Pump; Turbine Driven
199700133	Boric Acid Batching Tank Heater 4
199700111	Rockshaft for HCV-1042A Missing Set Screw Holes
199700019	Radiation Monitor RE-064 Isolation Valve

199700014	Jumper on TB-1 Not on Drawing
199700008	Safety Injection & Refueling Water Tank (Safety Injection Refueling Water Tank)
199601556	Shutdown Cooling Heat Exchanger AC-4A; Component Cooling Water Outlet valve
199601030	Boric Acid Batching Tank Heater 4
IR 950617	Boric Acid Batching Tank Heaters
IR 950166	Boric Acid Batching Tank Heaters
199700409	Component Cooling Water Pump Flange Thread Engagement

Safety Audit and Review Committee Audits

Number	Title
94-SARC-013	SARC Audit Report 25S and QA Audit 25 Fire Protection/Loss Prevention
95-SARC-020	SARC Audit Report 25A Fire Protection/Loss Prevention
96-SARC-022	SARC Audit Report 25A and 25B Audit Report 25 Fire Protection/Loss Prevention

Quality Assurance Audit and Surveillance Reports

Number	Title
96-QA/QC-002	Quality Assurance Audit Report 69 Safety System Functional Inspection Diesel Generator System
95-CQA-033	Quality Assurance Surveillance Report E2-95-1 Electrical Equipment Qualifications
95-CQA-024	Quality Assurance Surveillance Report E4-95-2 Modifications During Outages
95-CQA-018	Quality Assurance Surveillance Report E6-95-1 Design Basis Document/Drawing Verification
95-CQA-066	Quality Assurance Surveillance Report E6-95-3 design basis document/Drawing control
95-CQA-086	Quality Assurance Surveillance Report E6-95-4 design basis document/Drawing control
95-CQA-019	Quality Assurance Surveillance Report E7-95-1 Station Engineering and Technical Reviews

96-CQA-008	Emergent Quality Assurance Surveillance Report E-96-1 Use of Teflon Tape in Chemical Feed (Hydrazine) System
96-CQA-042	Quality Assurance Surveillance Report E4-96-1 Modifications During Outages
97-QA/QC-032	Quality Assurance Surveillance Report E6-97-1 Design Basis Document/Drawing Control

10 CFR 50.59 Screening and Safety Evaluations:

Number	Title
MR-FC-84-155C	Replacement Process and Effluent Radiation Monitors-Package C which included 11 50.59 Applicability Screenings and 7 Unreviewed Safety Question Determinations
MR-FC-95-026	Fuel Storage Tanks Inside of Protected Area
ECN 95-337	New Test Terminal Blocks for Waste Gas Analyzer Relays
ECN 95-394	HCV-482/3 A/B Valve Blocks
ECN 92-559	Circulating Water Pump Motor Lower Bearing Oil Drain Valve
ECN 93-415	Change Out of Blowdown Pump Trip Override Switch
ECN 96-159	Plug for CW-1C Upper Drain Line

Technical Specification Interpretations

Number	Technical Specifications/Updated Safety Analysis Report Section
97-06	Technical Specifications 1.3(B) and Technical Specifications 2.15 Bases
92-08	Technical Specifications 2.1.1(3)c and 2.1.1(4) Exception
96-10	Technical Specifications 2.1.6(4)
97-05	Technical Specifications 2.1.6(4) Bases
96-06	Technical Specifications 2.1.6(5)
96-03	Technical Specifications 2.1.7
93-03	Technical Specifications 2.4(1)a
97-08	Technical Specifications 2.5
95-04	Technical Specifications 2.7(1)j
97-07	Technical Specifications 2.7(1)j
97-03	Technical Specifications 2.7(2)j

97-04	Technical Specifications 2.10.2(4)a
94-13	Technical Specifications 2.10.2(4)
95-12	Technical Specifications 2.10.4(5)(b)
95-06	Technical Specifications 2.10.4(5)(b)
95-16	Technical Specifications 2.12(3)
95-15	Technical Specifications 2.12(4)
93-12	Technical Specifications 3.1 Table 3-2
96-14	Technical Specifications 3.2 Table 3-4
93-15	Technical Specifications 5.0

Substitute Replacement Item Engineering Change Notices

Number	Revision	Title
ECN 91-361	2	Replacement of SA-170 Valve
ECN 96-307	1	Packing Substitution for Crane Valves
ECN 96-354	0	Replacement of Defective Mounting Track for Relays
ECN 96-312	0	Replacement of LIA-906Y
ECN 96-322	0	Replacement of SBM Switch
ECN 96-398	0	HCV-484/485 Coupling
ECN 96-402	1	Safety Injection Accumulator Pressure Gage Replacement
ECN 96-415	0	Control Element Drive Mechanism Threaded Seal Substitute
ECN 96-428	1	Subcooled Margin Monitor Power Supply
ECN 96-462	0	Bushing for Anchorhead on Containment Dome Tendon
ECN 97-002	0	Replacement of HCV-348 Breaker
ECN 97-030	0	HE-2 Accumulator Substitute Replacement
ECN 97-093	0	Replacement CR120A for HCV-2877A
ECN 97-127	0	Replacement B/PIA-102Y
ECN 95-130	0	CCW Relief Valve Setpoint Changes

ECN 95-124	0	CB-520-SR-60 Spring Replacement
ECN 95-321	0	FW-10 Oil Relay Assembly
ECN 96-277	0	FW-663 Valve Replacement
ECN 96-278	0	Replacement of HC-1045 Hand Control Switch

Engineering Assistance Requests

Number	Title
EAR 97-092	Replacement Spring Pack for Limitorque Operator SMB-00
EAR 97-073	Diesel Generator Droop Setting
EAR 95-111	Containment Spray Pump Recirculation Valves
EAR 93-030	Auxiliary Feedwater Pump FW-10 Differential Pressure Indicators

Engineering Analyses

Number	Revision	Title
EA-FC-96-051	2	IE Bulletin 80-06, "Engineered Safety Features Reset"
EA-FC-96-001	0	Criticality Safety Evaluation of the Fort Calhoun Spent Fuel Storage Racks for Maximum Enrichment Capability
EA-FC-92-077	0	Licensing Report for Spent Fuel Storage Capacity Expansion
EA-FC-93-042	1	Containment Seal Penetration Evaluation
EA-FC-93-047	1	Halon System Operability Evaluation

Design Basis Documents

Number	Revision	Title
SDBD-AC-SF-102	9	Spent Fuel Pool and Fuel Pool Cooling Design Basis Document
SDBD-DG-112	15	Emergency Diesel Generators Design Basis Document
SDBD-MS-125	9	Main Steam and Turbine Steam Extraction Design Basis Document

Nuclear Safety Review Group Assessments and Special Reports

Number	Revision	Title
SRG-95-018	0	Oversight of the Test Engineering Group
SRG-95-036	0	Followup Assessment of Operator Work-arounds
SRG-95-057	0	4160 Volt Circuit Breaker Replacement
SRG-95-086	0	Diesel Generator Hot Weather Operation Issue
SRG-96-046	0	Reactor Coolant Pumps Vibration Monitoring
SRG-96-079	0	Backlog of Engineering Change Notices/Engineering Analysis Requests/Maintenance Requests
SRG-96-084	0	Absence of Steam Generator Orifice Plates in Loss-of-Coolant Analysis
SRG-96-091	0	Maintenance Timeliness and Configuration Control
SRG-97-006	0	Steam Leak Resulted Due to Blown Gasket
SRG-97-015	0	Auxiliary Feedwater Pump (FW-110) Low Discharge Differential Pressure
SRG-97-022	0	Nuclear Safety Review Group Review of Substitute Replacement Item Engineering Change Notice

Maintenance Work Request

MWR 9701551	Sealing Reactor Coolant Pump Lube Oil Collection System

Miscellaneous Documents

Technical Specifications

Updated Safety Analysis Report

NOD Technical Staff (Technical Specifications) Training Handbook Lesson Plan 2327-07, Revision 10, dated February 3, 1997 (Initial)

NOD Qualified Reviewer Process and 10 CFR 50.59 Safety Evaluation Biennial

Requalification Training Lesson Plan SEAD-36, Revision 3 (Requalification)

LIM-97-0034, "Cancellation of Technical Specification Interpretation 95-12 and 91-02," dated March 31, 1997

LIM-97-0020, "Cancellation of Technical Specification Interpretation 95-09," dated February 19, 1997

LIM-96-0165, "Review of Technical Specification Interpretations," dated November 14, 1996

LIM-96-0164, "Review of Technical Specification Interpretations," dated November 11, 1996

LIM-96-0163, "Review of Technical Specification Interpretations," dated November 11, 1996

LIM-96-0145, "Meeting to Review Technical Specification Interpretations," dated October 18, 1996

LIM-96-0047, "10 CFR 50.59 Oversight Committee Charter," dated April 2, 1996

LIM-96-0015, "50.59 Improvement Plan Action Assessment Team," dated January 18, 1996

PED-FC-95-33, "10 CFR 50.59 Improvement Program," dated September 28, 1995

LER 96-006-00, "All Charging Pumps Disabled Due to an Inadequate Administrative Control," dated October 6, 1996

Licensee letter to the NRC dated December 12, 1979, concerning a water curtain

Licensee letter to the NRC dated January 18, 1980, concerning a water curtain

Licensee letter to the NRC dated May 20, 1980, concerning a water curtain

EOS-DEN-97-0165, "Final Report for the Self Assessment of Licensing Basis Conformance for the Auxiliary Feedwater System," April 17, 1997

PED-DEN-97-0049, "Final Report for the Self Assessment of Licensing Basis Conformance for the Safety Injection System," February 7, 1997

PED-DEN-96-0442, "Final Report for the Self Assessment of Licensing Basis Conformance at Fort Calhoun Station (Chemical and Volume Control System)," October 3, 1996

PED-DEN-95-574, Modification MR-FC-91-009, "Spent Fuel Pool Rerack," August 7, 1995

PED-FC-94-1172, "Revision of Updated Safety Analysis Report," Sections 9.3.1 and 9.7.4.2, September 29,1994

LIM-95-0208, "Updated Safety Analysis Report Changes for the 1995 Updated Safety Analysis Report Update," September 3, 1995

PED-FC-93-2426, "Plant Review Committee Interim Subcommittee for Updated Safety Analysis Report Changes," July 7, 1993

FC-T-058-97, "Personnel Qualified to Perform/Review 10 CFR 50.59 Safety Evaluations," March 13, 1997

FC-T-253-96, "Personnel Qualified to Perform/Review 10 CFR 50.59 Safety Evaluations," October 23, 1996

LIC-97-0039, NRC Inspection Report 50-285/96-17, Reply to a Notice of Violation, March 31, 1997

Configuration Control Self Assessment Report, April 11,1997

Production Engineering Division Organizational Chart, Chart 3.1.01, Revision 0

Nuclear Program Planning Manual, Section 5, "Prioritization," Revision 6, March 15,1996

Nuclear Energy Institute, NEI-96-05, "Industry Initiative to Address Licensing Basis Conformance Issues," Revision 0

Vendor test curves for Auxiliary Feedwater Water Pump Serial 691-N-0436

Vendor test curves for Auxiliary Feedwater Water Pump Serial 891-C-0033

Vendor test curves for Auxiliary Feedwater Water Pump Serial 8494DBE

Vendor Manual TD C438.0020, "Instructions for Installation, Operation, and Maintenance of Coffin Turbo Pump Type DEB Auxiliary Feed Pump," Revision 3, December 5, 1994

Vendor Manual TD B580.0160, "Installation and Operation Instructions for Motor Driven Auxiliary Feed Pump Type DVMX," Revision 1, June 28, 1994

Vendor Manual TD G080.2260, "Instruction Manual for Custom 8000 Horizontal Induction Motors Dripproof, Splashproof or Weather-Protected Type 1," Revision 2, June 8, 1990

Dresser Relief Valve Bulletin 71.4:98 concerning valve reset pressures

Calculation, "Usable Capacity of Emergency Feedwater Storage Tank FW-19," dated November 23, 1988

Engineering Evaluation for Technical Specification Interpretation 90-01, Operability of Fire Water Suppression System

E-Mail report to T. Boyce (THB)
E-Mail report to NRR Event Tracking System (IPAS)
E-Mail report to Document Control Desk (DOCDESK)

bcc to DCD (IE01)

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