UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

December 30, 2009

David J. Bannister, Vice President
and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4
P. O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION- NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000285/2009006

Dear Mr. Bannister:

On September 17, 2009, the US Nuclear Regulatory Commission (NRC) completed a component design bases inspection at Fort Calhoun Station. The enclosed report documents our inspection findings. The preliminary findings were discussed on September 17, 2009, with Mr. Bannister and other members of your staff. After additional in office inspection, a final telephonic exit meeting was conducted on December 28, 2009, with Mr. Reinhart, Site Vice President, and others of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The team reviewed selected procedures and records, observed activities, and interviewed cognizant plant personnel.

Based on the results of this inspection, the NRC has identified five findings that were evaluated under the risk significance determination process. Violations were associated with five of the findings. Five of the findings were found to have very low safety significance (Green) and the violations associated with these findings are being treated as noncited violations, consistent with Section VI.A.1 of the NRC Enforcement Policy.

If you contest any of the noncited violations, or the significance of the violations you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the US Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 East Lamar Blvd., Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the

Regional Administrator, Region IV, and the NRC Resident Inspector at Fort Calhoun Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with Code of Federal Regulations, Title 10, Part 2.390 of the NRC's Rules of Practice, a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Thomas Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

Dockets: 50-285
License: DPR-40

Enclosure:
Inspection Report 05000285/2009006
w/Attachments: (1) Supplemental Information
(2) Failure Probability of the Alternative Mitigation Strategy

cc w/enclosure:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report Nos.:	05000285/2009006
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	August 24 - 28, 2009 Onsite September 8 - 18, 2009 Onsite September 18 - December 28, 2009 In-Office
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Approved By:	Thomas Farnholtz, Branch Chief Engineering Branch 1

SUMMARY OF FINDINGS

IR 05000285/2009006; August 24-28, 2009, and September 8-18, 2009; Fort Calhoun Station: baseline inspection, NRC Inspection Procedure 71111.21, Component Design Basis Inspection.

The report covers an announced inspection by a team of four regional inspectors and two contractors. Five findings were identified. Three of the findings were of very low safety significance (Green). Two of the findings were determined to be Severity Level IV violations. The final significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, Significance Determination Process. Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649*,* "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC Identified Findings

Cornerstone: Mitigating Systems

- Green. The team identified a Green, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, from February 1992 to September 8, 2009, the licensee failed to adequately evaluate the seismic qualification of the raw water pumps to ensure that the pumps' anchor bolts imbedded in the floor would meet Seismic Class I standards. The team determined that the February 1992 seismic analysis was not conservative for the following reasons:

    - The weight distribution of the pump/motor assembly in the analysis did not correctly apply the center of gravity of the pump to the loading analysis.

    - The stress analysis of the anchors did not include the weight of the water in the piping.

    - The stress analysis did not include the nozzle loads applied to the pump due to the weight of the discharge piping.

  The licensee entered the issue into their corrective action program as CR 2009-3977, and performed a preliminary operability evaluation of the support components which determined that the pumps would remain operable following a safe shutdown earthquake. The team reviewed the evaluation, and concurred with the operability evaluation.

  The finding is more than minor because it adversely affected the design control attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance (Green) because it was a design deficiency that did not result in actual loss of safety function. This finding was not assigned a crosscutting aspect because the underlying cause was not indicative of current performance (Section 1R21.2.15).

- Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, from August 9, 1973, to September 8, 2009, the licensee failed to prescribe instructions into procedures that would ensure that the plant could be safely shutdown at the probable maximum flood elevation of 1009.3 feet mean sea level. The licensee's updated safety analysis report, technical specifications, and station procedures state that protection of the raw water pumps against flooding up to the probable maximum flood height of 1009.3 feet mean sea level is accomplished by sandbag berms and flood gates. During an intake structure walkdown, the team observed two unsealed, 14-inch diameter fire protection piping penetrations in the outer wall, with the bottom of the penetration at elevation 1008.5 feet mean sea level. The penetrations had an air gap of about ½-inch between the wall and the pipe. After reviewing station procedures, the team determined that the unsealed penetrations would not be sealed during flooding conditions.

  As a result of the team's concern, the licensee entered the issue into their corrective action program as CR 2009-4166 and CR 2009-6195, and verified that there are no other open penetrations in the building walls below the flood level of 1009.3 feet mean sea level. The licensee changed procedure GM-RR-AE-1002 to provide temporary sealing of the penetrations if predicted floods occurred before the permanent seals were installed. The licensee stated that the penetrations will be permanently sealed before the spring 2010 flood season.

  This performance deficiency is more than minor because it adversely affected the Mitigating Systems Cornerstone attribute of external events and affected the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The finding affected the Mitigating Systems Cornerstone because flood protection was degraded. The team determined that the finding resulted in the degradation of equipment and functions specifically designed to mitigate a flooding initiating event. In addition, during a flooding event, the loss would degrade two or more trains of a multi-train safety system. Therefore, the finding was potentially risk significant to flood initiators and a Phase 3 analysis was required. The final change in core damage frequency was calculated to be 8.2×10^{-7} indicating that the finding was of very low safety significance (Green). This finding was not assigned a crosscutting aspect because the underlying cause was not indicative of current performance (Section 1R21.2.15).

- Green. The team identified a Green, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" for failure to take adequate corrective action following the discovery of water intrusion in manholes MH-5 and MH-31 in 1998, 2005, and 2009. Specifically, from 1998 to September 11, 2009, the licensee failed to take corrective action to establish an appropriate monitoring frequency that would mitigate potential common mode failure of raw water 5kV motor cables in underground ducts and manholes. The licensee entered the condition into the corrective action program as

CR 2009-4216. The corrective action changed the manhole inspection schedule from an 18-month schedule to a quarterly schedule.

The finding is more than minor because it adversely affected the Mitigating Systems Cornerstone attribute of design control for ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance (Green) because it was a design deficiency that did not result in actual loss of safety function. This finding has a crosscutting aspect in the area of human performance, decision making, because the licensee failed to use conservative assumptions in decision-making and adopts a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, since 2005, the licensee decided to postpone installation of proposed water level control corrective actions and failed to appropriately monitor water intrusion in MH-5 and MH-31 multiple times [H.1(b)](Section 1R21.3.4).

- SL-IV. The team identified a Severity Level IV, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records," for failure to maintain original records of the seismic and tornado analysis of the intake structure. Specifically, in 2005, the licensee could not retrieve the original design documentation of the seismic and tornado analysis of the intake structure. This condition was documented in CR 200504345. After the licensee determined the documentation was not retrievable, the licensee reconstituted the seismic and tornado analysis of the intake structure. These analyses were available during the team's inspection.

  This finding is assessed through traditional enforcement because the finding has the potential for impacting the NRC's ability to perform its regulatory function. Using Inspection Manual Chapter 0612, Appendix E, the finding is more than minor because the records were not retrievable. Using Supplement I of the NRC Enforcement Policy, this finding will be treated as a Severity Level IV violation. This finding was not assigned a crosscutting aspect because the underlying cause was not indicative of current performance (Section 4OA5.1).

- SL-IV. The team identified a Severity Level IV, noncited violation for failure to update the final (updated) safety analysis report in accordance with 10 CFR 50.71(e). Specifically, the licensee failed to update Section 9.8, "Raw Water Systems," of the Fort Calhoun Station Updated Safety Analysis Report after constructing a sheet pile alignment wall alongside the intake structure in 1982. Furthermore, this modification removed the slope from the river bottom. Additionally, recent sounding records indicate the river bottom near the intake structure is approximately the same depth as the center of the channel, thus, invalidating the updated safety analysis report statement. The licensee entered this condition into the corrective action program as CR 2009-3927.

  The finding is more than minor because the finding is determined to have a material impact on safety. Specifically, with the new sheet pile alignment wall, it could lead to a barge strike that is different than described in the updated safety analysis report. Using Supplement I of the NRC Enforcement Policy, this finding will be treated as a Severity

Level IV violation. This finding was not assigned a crosscutting aspect because the underlying cause was not indicative of current performance (Section 4OA5.1).

B. Licensee-Identified Violations.

None.

REPORT DETAILS

1 REACTOR SAFETY

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and important design features may be altered or disabled during modifications. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity Cornerstones for which there are no indicators to measure performance.

1R21 Component Design Bases Inspection (71111.21)

A team of NRC inspectors selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than two or a Birnbaum value greater than 1E-6.

a. Inspection Scope

To verify that the selected components would function as required, the team reviewed design basis assumptions, calculations, and procedures. In some instances, the team performed calculations to independently verify the licensee's conclusions. The team also verified that the condition of the components was consistent with the design bases and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For the review of operator actions, the team observed operators during simulator scenarios, as well as during simulated actions in the plant.

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense-in-depth margins.

The inspection procedure requires a review of 20 to 30 total samples, including 10 to 20 risk-significant and low design margin components, 3 to 5 relatively high-risk operator actions, and 4 to 6 operating experience issues. The sample selection for this inspection was 15 components, 5 operator actions, and 4 operating experience items.

.2 **Results of Detailed Reviews for Components:**

.2.1 Voltage Regulator Emergency Diesel Generator (1 & 2)

a. Inspection Scope

The team inspected the electrical portions of the emergency diesel generator, including the voltage regulator, to verify the adequacy of the equipment to respond to design basis events. The team reviewed diesel generator load sequencing logic to verify the appropriate functionality was implemented. The team reviewed completed surveillances to verify that the technical specification requirements were met and to assess the response of the exciter and voltage regulator.

Additionally, the team reviewed protection/coordination and short-circuit calculations to verify the standby diesel generator was adequately protected by protective devices during emergency operation. The team reviewed calculations to verify that: 1) steady-state loading was within design capabilities; 2) adequate voltage would be present to start and operate connected loads; and, 3) operation at maximum allowed frequency would be within the design capabilities. The team reviewed the standby diesel generator load sequence time delays. The team reviewed the standby diesel generator voltage regulator maintenance and control voltage to verify that the components would function when required. Finally, the team reviewed the recorder printout from the latest diesel generator automatic loading test to ensure the voltage returned to nominal between automatic load steps.

b. Findings

No findings of significance were identified.

.2.2 Containment Fan Cooler 7C & 7D

a. Inspection Scope

The team reviewed the design basis loading, motor protective relaying, protective circuit, and device settings controls tied to the emergency diesel generator sequencing of the fan coolers. Additionally, the team reviewed selected calculations for electrical distribution system load flow/voltage drop, short-circuit, and electrical protection and coordination. The adequacy and appropriateness of design assumptions were reviewed to verify that bus capacity was not exceeded and bus voltages remained above minimum acceptable values under design basis conditions. The fan cooler motor protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of containment fan cooler and containment spray pump during worst-case short-circuit conditions. The team also reviewed breaker calibration and preventive maintenance programs associated with the breakers.

Additionally, the team reviewed the Fort Calhoun Updated Safety Analysis Report (USAR), technical specifications, design basis documents, calculations, maintenance records, and modification documents for the water management modification. This modification removed the actuation of the containment spray pumps following a loss of coolant accident, and credited the containment fan coolers and filters for pressure and temperature control and removal of radioactive particulates. The team reviewed the analysis to ensure that the required mission time of the fan coolers was 30 days. The team reviewed the modifications to determine if the previously installed partial strainer was adequate by elimination of the containment spray function for a loss of coolant accident. Additionally, the team reviewed re-analyses of the containment pressure/temperature responses to ensure that the analyses provided the revised design basis for the containment fan cooler requirements.

b. Findings

No findings of significance were identified.

.2.3 125 Vdc Station Batteries 1 & 2

a. Inspection Scope

The team reviewed the 125 volt direct current (Vdc) system drawings, USAR, technical specifications, 125 Vdc system design basis documents, and the system health reports for the 125 Vdc system. The team reviewed the battery sizing calculations to ensure the replacement batteries were adequately sized. Additionally, the team reviewed the assumed correction factors for temperature and aging to establish design margin. Finally, the team compared the calculated profiles to the results of last battery discharge tests to verify that the established margin was correct.

The team reviewed surveillance procedures for the batteries to ensure correct acceptance criteria, including minimum cell voltage and specific gravity and minimum battery terminal voltage to ensure correct action limits were set for those battery parameters. The team also compared those limits against the technical specifications, procurement specifications, and the manufacturer’s specifications and instruction manual. Additionally, the team reviewed the 125 Vdc voltage drop calculations to ensure that sufficient voltage would exist at the safety-related loads during a battery design discharge.

In addition, the team performed walkdowns of the battery rooms and 125 Vdc systems. The team reviewed the battery room heating, ventilation, and air conditioning systems and heating load calculations to ensure the correct parameters were used for battery room temperature and air flow.

b. Findings

No findings of significance were identified.

.2.4 Motor-Driven Auxiliary Feedwater Pump FW6 Breaker 1A3-16

a. Inspection Scope

The team reviewed the 4 kilovolt (kV) switchgear breaker maintenance history and related condition reports to assess the reliability of the power supply for the motor-driven auxiliary feedwater pump. The team reviewed the control circuit for the circuit breakers controlling power to the power circuit to verify that it would operate during design basis events. The switchgear's protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst-case, short-circuit conditions. Additionally, the team reviewed the degraded and loss of voltage relay protection schemes. The 125 Vdc voltage calculations were reviewed to determine if adequate voltage would be available for the breaker open/close coils and spring charging motors. Finally, the team reviewed the sizing of the 5 kV power cables for the auxiliary feedwater pump motors.

b. Findings

No findings of significance were identified.

.2.5 Raw Water Pump Breaker 1A3-9

a. Inspection Scope

The team reviewed the 4 kV switchgear breaker maintenance history and related condition reports to assess the reliability of the power supply for the raw water pumps. The team reviewed the control circuit for the circuit breakers controlling power to the power circuit to verify that it would operate during design basis events. The switchgear's protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst-case, short-circuit conditions. Additionally, the team reviewed the degraded and loss of voltage relay protection schemes. The 125 Vdc voltage calculations were reviewed to determine if adequate voltage would be available for the breaker open/close coils and spring charging motors. Finally, the team reviewed the sizing of the 5 kV power cables for the raw water pump motors.

b. Findings

No findings of significance were identified.

.2.6 House Service Power Transformer T1A-3 & -4 (161kV)

a. Inspection Scope

The team reviewed the system one-line diagrams, voltage tap settings, nameplate data, protective relay settings, and loading requirements to determine the adequacy of the transformer to supply required power to the associated 4160 volt alternating current (Vac) buses. In addition, the team reviewed the results of recently completed transformer protective relay preventive maintenance.

For the 161 kV switchyard, the team reviewed offsite power connections and the transmission operator notification protocols. To assess the potential vulnerabilities of the transformer to external hazards, the team performed a visual inspection of the 161 kV switchyard to assess the installation, configuration, and material condition.

The team reviewed selected calculations for electrical distribution system load flow/voltage drop, degraded voltage protection, short-circuit analysis, and electrical protection and coordination. This review was conducted to determine the adequacy and appropriateness of design assumptions. In addition, the team verified that transformer capacity was not exceeded and bus voltages remained above minimum acceptable values under design basis conditions. The team reviewed the operating procedures which would address potential minimum voltage at the 161 kV switchyard following a trip of the Fort Calhoun Station generator and resultant fast transfer of the loads from the station auxiliary transformers to the house service power transformers to ensure adequate interface with the offsite power supply organization.

Finally, the team evaluated selected portions of the licensee response to NRC Generic Letter (GL) 2006-02, “Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power,” dated February 1, 2006.

b. Findings

No findings of significance were identified.

.2.7 Containment Spray Pump Breaker 1B4B-1

a. Inspection Scope

The team reviewed actuation circuits, interlocks, and emergency diesel generator loading associated with containment spray pumps. The team reviewed control circuits, selected calculations for electrical distribution system load flow/voltage drop, short-circuit analysis, and electrical protection and coordination. The team reviewed the adequacy of design assumptions in calculations to verify that, upon starting and running of the containment spray pumps, bus stability is maintained within design basis conditions and requirements. The pump motor protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of the containment spray pump during worst case short-circuit conditions. The team also reviewed breaker calibration and preventive maintenance programs and logic functional tests associated with the pumps.

b. Findings

No findings of significance were identified.

.2.8 Emergency Diesel Generators 1 & 2 Common Cause Failures

a. Inspection Scope

The team reviewed the USAR, technical specifications, design bases documents, calculations, corrective maintenance, and testing of the emergency diesel generator systems. The team performed several detailed walkdowns of the emergency diesel generator rooms and essential support systems to determine whether design or operational conditions existed that would compromise the performance of both emergency diesel generators. In particular, the team reviewed external flood and seismic evaluations of the fuel oil storage tank and emergency diesel generator day tanks to ensure that the selected equipment could withstand external flooding and seismic loads. Additionally, the team reviewed internal flooding studies to ensure that there was no potential to flood the building and cause common cause failure of the emergency diesel generators.

The team reviewed the fuel oil, lube oil, cooling, and starting air systems of the emergency diesel generators. For the fuel oil system, the team reviewed recent oil sample results and fuel oil consumption to ensure technical specification requirements were met. For the lube oil system, the team verified that sufficient lube oil supplies are onsite to support extended diesel generator runs, if required. The air-cooled diesel generator radiator system design requirements were reviewed to ensure adequate performance of the radiator under worst-case ambient conditions. The diesel generator room ventilation system design was reviewed, including failure positions of air-operated dampers. The team reviewed the emergency diesel generator air start system configuration which included a connection between two of the emergency diesel generator air start accumulators to ensure that any failure in the connecting air lines would not result in loss of air start capability for the emergency diesel generator. The team reviewed inservice testing of the safety-related portion of the starting air system.

b. Findings

No findings of significance were identified.

.2.9 Diesel-Driven Auxiliary Feedwater Pump FW-54

a. Inspection Scope

The diesel-driven auxiliary feedwater pump is independent of all plant support systems and has its own fuel oil storage tank. The team reviewed design documents, including drawings, calculations, procedures, and the design basis document to determine the design requirements for the diesel-driven auxiliary feedwater pump. Hydraulic analyses were reviewed to verify adequacy of net positive suction head and verify adequacy of surveillance test acceptance criteria for pump minimum discharge pressure at required flow rate. Maintenance testing results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Fuel oil sample results were reviewed to ensure fuel oil quality is maintained. The team performed a walkdown of the diesel-driven auxiliary feedwater pump area and supporting equipment to determine whether the alignment was in accordance with design basis and procedural requirements, and to assess the material condition of the pump and diesel-engine driver. Preventive and corrective maintenance records were reviewed to ensure the auxiliary feedwater pump and diesel-engine driver were properly maintained. Finally, the team reviewed corrective action documents to ensure problems associated with the diesel-driven feedwater pump were appropriately identified and corrected.

b. Findings

No findings of significance were identified.

.2.10 Motor- Driven Auxiliary Feedwater Pump FW-6

a. Inspection Scope

The team reviewed design documents, including drawings, calculations, procedures, and the design basis document to determine the design requirements for the auxiliary feedwater motor driven pump, FW-6. Hydraulic analyses were reviewed to verify adequacy of net positive suction head and verify adequacy of surveillance test acceptance criteria for pump minimum discharge pressure at required flow rate. Inservice testing results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Pump actuation logic test results were reviewed to ensure the motor driven auxiliary feedwater pump would start in accidents and events as described in the USAR. In addition, the licensee responses and actions related to IE Bulletin 88-04, “Potential Safety-Related Pump Loss,” were reviewed to assess implementation of operating experience related to pump minimum flow requirements, and pump-to-pump interaction. The inspectors reviewed condensate storage tank design criteria, including seismic qualification and usable volume calculations, to ensure the motor-driven auxiliary feedwater pump had adequate safety-grade water supply.

The team performed a walkdown of the motor-driven auxiliary feedwater pump area and supporting equipment to determine whether the alignment was in accordance with design basis and procedural requirements, and to assess the material condition of the pump and motor. Preventive and corrective maintenance records were reviewed to ensure the auxiliary feedwater pump was properly maintained. Finally, the team reviewed corrective action documents to ensure problems associated with the pump were appropriately identified and corrected.

b. Findings

No findings of significance were identified.

.2.11 Power Operated Relief Valve PCV 102-1

a. Inspection Scope

The team reviewed design calculations that were performed to determine the lift settings and required stroke timing of the power operated relief valves while in the low temperature overpressure protection mode of operation. Low temperature overpressure protection mode for pressure relief consists of two power operated relief valves with variable lift settings that act as safety relief valves to limit the pressure buildup in the reactor coolant system. While operating in the low temperature overpressure protection mode, procedure restrictions ensure the integrity of the reactor coolant pressure boundary is not compromised by violating the pressure and temperature limits of 10 CFR Part 50, Appendix G, “Fracture Toughness.” The team also reviewed instrument setpoint and uncertainty calculations for pressurizer level, temperature, and pressure instruments that are relied upon during the low temperature overpressure protection operation. The team reviewed operational mode change procedures to ensure the low temperature overpressure protection controls on pump starts, and pressurizer level were in place when required. This review was done to ensure the assumptions of power operated relief valve stroke time, reactor coolant system pressurization rate due to the operation of safety injection pumps and reactor coolant pumps, and the low temperature overpressure protection procedure restrictions in the analysis were translated into plant operating procedures. The team reviewed power operated relief valve inservice tests to ensure valve opening timing is consistent with requirements from design documents. Corrective action documents were reviewed to ensure problems were identified and corrected in a timely manner.

b. Findings

No findings of significance were identified.

.2.12 Containment Spray Pump 3B

a. Inspection Scope

The team reviewed design documents, including drawings, calculations, procedures, and the design basis document to determine the design requirements for the containment spray pump 3B. Hydraulic analyses were reviewed to verify adequacy of net positive suction head and verify adequacy of surveillance test acceptance criteria for pump minimum discharge pressure at required flow rate. Inservice testing results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Pump actuation logic test results were reviewed to ensure the containment spray pump would start in accidents as described in the USAR. In addition, the licensee responses and actions related to IE Bulletin 88-04, “Potential Safety-Related Pump

Loss," were reviewed to assess implementation of operating experience related to pump minimum flow requirements, and pump to pump interaction.

The team performed a walkdown of the containment spray pump area and supporting equipment to determine whether the alignment was in accordance with design basis and procedural requirements. Preventive and corrective maintenance records were reviewed to ensure the pump was properly maintained. Finally, the team reviewed corrective action documents to ensure problems associated with the pump were appropriately identified and corrected.

b. Findings

No findings of significance were identified.

.2.13 Auxilliary Feedwater Steam Admission Valves, AOV 1045 and 1045 A&B

a. Inspection Scope

For steam admission valves AFW-1045 and 1045A/B, the team reviewed the USAR, technical specifications, design basis documents, drawings, and procedures to identify the design basis requirements for the valves. The valve testing procedures and valve specifications were reviewed to verify the design basis requirements, including worst case system and environmental conditions, were incorporated into the test acceptance criteria and equipment design. The adequacy and availability of the backup air source for valve operation was assessed by reviewing equipment history to verify that valves were adequately maintained and that identified equipment problems were resolved. Recent performance tests were reviewed to assure that the equipment capability was monitored and maintained to meet their design basis function. Additionally, a walkdown was performed to assess the observable material condition and verify the installed configuration was consistent with the design bases and plant drawings.

b. Findings

No findings of significance were identified.

.2.14 Auxilliary Feedwater Containment Isolation Valve, AOV 1107B & 1108B

a. Inspection Scope

For containment isolation valves AFW-1107B and 1108B, the team reviewed the design basis documents, technical specifications, USAR, drawings, and procedures to identify the design basis requirements for the valves. The valve testing procedures and valve specifications were reviewed to verify the design basis requirements, including worst case system and environmental conditions, were incorporated into the test acceptance criteria and equipment design. The adequacy and availability of the backup air source for valve operation was assessed by reviewing equipment history to verify that valves were adequately maintained and that identified equipment problems were resolved.

Recent performance tests were reviewed to assure that the equipment capability was monitored and maintained to meet their design basis function. Additionally, a walkdown was performed to assess the observable material condition and verify the installed configuration was consistent with the design bases and plant drawings.

b. Findings

No findings of significance were identified.

.2.15 Intake Structure

a. Inspection Scope

The safety-related functions of the intake structure is to provide water from the Missouri River that is required for component cooling and fire fighting at Fort Calhoun Station and to provide the structural support and environmental protection necessary to ensure the functional integrity of the raw water pumps and fire protection pumps under all operational and environmental conditions. The team performed several detailed walkdowns of the intake structure and reviewed design and event analyses to ensure that the structure would remain intact during design bases events. The team reviewed the Fort Calhoun Station Individual Plant Examination for External Events (IPEEE) report to determine the probability of external events that the intake structure was able to sustain. The review focused on the potential for damage to the intake structure caused by barge impact, seismic, and external flooding events. The barge impact and seismic events were selected based on previous inspection unresolved items and the flooding event was chosen based on significance of core damage if flooding mitigation features do not work.

b. Findings

b.1 Inadequate Assessment of Seismic Qualification of Raw Water Pumps

Introduction. The team identified a Green, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, the licensee did not adequately evaluate the seismic qualification of the anchorage of raw water pumps in the intake structure, to ensure that the pumps' anchor bolts imbedded in the floor would meet Seismic Class I standards.

Description. The team reviewed the Fort Calhoun Station USAR, Appendix F, Section 1.3.b.2 which describes the raw water system as a Seismic Class I system, and all supports, hangers, etc., associated with Class I equipment are also Class I.

During a walkdown of the intake structure, the team observed the raw water pump anchor bolts and seismic bracing of the raw water system piping attached to the pumps. The team noted that there were no longitudinal seismic restraints in the common raw water discharge header, and noted several disconnected seismic supports in the horizontal piping supports perpendicular to the piping header.

Following the walkdown, the team reviewed the seismic analysis of the raw water pumps contained in Screening Evaluation Worksheet, “Motor-Driven Pump AC-10A,” performed in February 1992. The seismic analysis for the raw water pumps included the analysis of the pump anchorage bolts embedded in the floor, and the piping nozzle loads applied to the pump anchorages. After reviewing the analysis, the team determined that the analysis was not conservative for the following reasons:

- The weight distribution of the pump/motor assembly in the analysis did not correctly apply the center of gravity of the pump to the loading analysis.

- The stress analysis of the anchors did not include the weight of the water in the piping.

- The stress analysis did not include the nozzle loads applied to the pump due to the weight of the discharge piping.

The team determined that failure of the anchors to remain in place would impact the capability of the safety-related raw water pumps to perform their design function following a seismic event. Failure of the raw water pumps to remain in place would result in a common cause failure of the raw water system to provide a cooling medium for the Seismic Class I component cooling water system.

As a result of the team’s concern, the licensee entered the issue into their corrective action program as CR 2009-3977 and performed a preliminary operability evaluation of the support components which determined that the pumps would remain operable following a safe shutdown earthquake. The team reviewed the evaluation, and concurred with the operability evaluation.

Following the inspection, the licensee performed a final seismic evaluation to assess the integrity of the pumps in accordance with the methodology described in the USAR. The final seismic evaluation concluded that the pumps with the seismic brace are acceptable for the required loadings.

Analysis. The team determined the failure to adequately evaluate the seismic qualification of the raw pumps was a performance deficiency because the evaluation did not meet Seismic Class I requirements. The finding is more than minor because it adversely affected the design control attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 worksheets, the finding was of very low safety significance (Green) because it was a design deficiency that did not result in actual loss of safety function. This finding was not assigned a crosscutting aspect because the underlying cause was not indicative of current performance.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, “Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §50.2 and as specified in the license application, for those structures, systems, and components for which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions." Contrary to this, from February 1992 to August 28, 2009, the licensee did not correctly translate Seismic Class I requirements into seismic evaluations of the raw water pumps to ensure that they would function during a seismic event. Because this violation is of very low safety significance and it was entered into the corrective action program as CR 2009-3977, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000285/2009006-01, "Inadequate Assessment of Seismic Qualification of Raw Water Pumps."

b.2 Inadequate Flood Protection for the Intake Structure

Introduction. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee failed to prescribe instructions into procedures that would ensure that the plant could be safely shutdown at the probable maximum flood elevation of 1009.3 feet mean sea level.

Description. Fort Calhoun Station is licensed and required to adhere to the 70 draft General Design Criteria published for comment in the Federal Register (32 FR 10213) on July 11,1967. General Design Criterion 2, "Performance Standards," states, "Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) Appropriate consideration for the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design."

The team determined license basis flood levels at Fort Calhoun Station by reviewing USAR Chapter 2.7, "Hydrology," USAR Chapter 9.8, "Raw Water System," and Technical Specifications 2.16, "River Level."

USAR Section 2.7.1.2 states, in part:

> "The design flood elevation of 1,006 feet based on a 0.1 percent probability flood is considered conservative. Without special provisions, the plant can accommodate flood levels of up to 1,007 [feet mean sea level]. Steel flood gates are permanently mounted above and adjacent to openings in structures containing equipment required for a safe and orderly plant shutdown. In the event of high water levels, these flood gates can be installed to provide protection to a level of 1,009.5 [feet mean sea level]. In the Intake Structure, protection to 1009.5 [feet] msl is accomplished with flood gates and sandbagging. The plant can be protected by sandbags, temporary earth levees and other methods to allow a safe shutdown with a flood elevation of 1,013 [feet mean sea level]."

USAR Section 9.8.6 states, in part:

> "Protection for the raw water pumps and their drives against floods is provided at three elevations as indicated on Figure 9.8-1. The pumps are permanently protected against any water level up to elevation 1007.5 feet by the Class I concrete substructure of the intake building. Protection is provided to elevation 1009.5 feet by sandbags around the traveling screen areas and by gasketed steel closures at exterior doorway openings in the intake structure reinforced concrete perimeter walls. Protection to elevation 1014.5 feet is provided by additional sandbags around the traveling screen areas, and by supplementing the intake structure perimeter walls with sandbags. The water level inside the intake cells can be controlled by positioning the exterior sluice gates to restrict the inflow into the cells."

Technical Specification 2.16 states

> "If the Missouri River level exceeds 1009 feet, the reactor will be placed in a cold shutdown condition using normal operating procedures. When the river level reaches elevation 1004.2 feet and rising, the emergency plan to protect the plant will be instituted."

During an intake structure walkdown in September 2009, the team observed two 14-inch diameter fire protection piping penetrations in the outer wall, with the bottom of the penetration near an elevation of 1008 feet mean sea level. The penetrations were not sealed, and had an air gap of about ½-inch between the wall and the pipe. Following the walkdown, the team requested the licensee provide a flooding analysis for the intake structure including an analysis of water entering the structure through the penetrations. The licensee's preliminary evaluation determined that all raw water pumps would be subject to flood water damage in one to four hours with a probable maximum flood level of 1009.3 feet mean sea level.

To determine if the penetrations would be sealed during a flood, the team reviewed Fort Calhoun Station Abnormal Operating Procedure (AOP) 1, "Acts of Nature," Revision 23 and procedure GM-RR-AE-1002, "Flood Control Preparedness for Sandbagging," Revision 8. AOP-1 directs the facility to use procedure GM-RR-AE-1002 which provides instructions for the installation of sandbag berms and other flood control barriers at the intake structure, auxiliary building, and turbine building when the flood levels reach heights between 1000 and 1014 feet mean sea level. Between flood levels of 1004 and 1007 feet mean sea level, the licensee will begin to install steel flood gates and build sandbag berms at the doors of the plant structures to protect to 1009.3 feet mean sea level. At a flood level of 1009 feet mean sea level, the procedure states that additional sandbags would be draped across flood gates to protect the intake structure to 1014 feet mean sea level as stated in Attachment 9.8 of procedure GM-RR-AE-1002.

The team identified, although flood protection would be installed in and around the intake structure, procedure GM-RR-AE-1002 did not prescribe steps to mitigate external flooding through the open penetrations.

The team determined that failure to ensure appropriate protection of the intake structure at increased flood levels would result in a common cause failure of the safety-related raw water pumps to safely shutdown the plant following a flooding event at 1009 feet mean sea level as stated Technical Specification 2.16, "River Level." Additionally, the team determined that the piping penetrations would need to be sealed and flooding protection strategies be revised to ensure the intake structure would be available for safe shutdown at 1009.5 feet mean sea level as stated in the USAR.

As a result of the team's concern, the licensee entered the issue into their corrective action program as CR 2009-4166 and CR 2009-6195, and verified that there are no other open penetrations in the building walls below the flood level of 1009.5 feet mean sea level. The licensee changed procedure GM-RR-AE-1002 to provide temporary sealing of the penetrations if predicted floods occurred before the permanent seals were installed. The licensee stated that the penetrations will be permanently sealed before the spring 2010 flood season.

Analysis. The team determined the failure to prescribe steps to protect the intake structure and raw water pumps from external flooding up to a flood height of 1009.5 feet mean sea level is a performance deficiency. Specifically, the licensee failed to prescribe steps into procedure GM-RR-AE-1002, "Flood Control Preparedness for Sandbagging," that would provide adequate protection to safely shutdown the plant at the flood level of 1009 feet mean sea level as stated in Technical Specification 2.16 and Abnormal Operating Procedure 1, "Acts of Nature."

This performance deficiency is more than minor because it adversely affected the Mitigating Systems Cornerstone attribute of external events and affected the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The team determined that the finding resulted in the degradation of equipment and functions specifically designed to mitigate a flooding initiating event. In addition, during a flooding event, the loss would degrade two or more trains of a multi-train safety system. Therefore, the finding was potentially risk significant to flood initiators and a Phase 3 analysis was required.

The senior reactor analyst completed a Phase 3 analysis using the plant-specific Standardized Plant Analysis Risk Model for Fort Calhoun, Revision 3.45; the Individual Plant Evaluation of External Events (IPEEE); and hand calculations. The exposure period of 1 year represented the maximum exposure time allowable in the significance determination process. The analysis covered the risk affected by the performance deficiency over a range of postulated external floods from 1008.4 to 1009.5 feet mean sea level. The flood hazard was developed using information provided by the U. S. Army Corp of Engineers and was confirmed by the licensee's risk analysts. Baseline failures at these flood elevations included an unrecoverable loss of offsite power, a loss of fuel transfer for Diesel Engine-Driven Auxiliary Feedwater Pump FW-54, and a loss of potable water. The performance deficiency resulted in a common cause failure of all Raw Water System pumps.

Given the postulated sequences and plant procedures, the analyst provided a screening factor of 0.1 for nonrecovery of the raw water pumps via diversion or pumping of the water coming into the intake structure. The analyst also provided a 0.1 nonrecovery screening value to account for alternative makeup sources of water to the emergency feedwater storage tank. Finally, the analyst estimated the failure rate for the minimum alternative mitigation equipment available to the licensee at 3.2 percent as provided in Attachment 2 of this inspection report.

The SPAR model provided that the primary accident sequences involve a loss of offsite power, loss of all auxiliary feedwater, and either a loss of containment cooling or a loss of once through cooling. The final change in core damage frequency was calculated to be 8.2×10^{-7} indicating that the finding was of very low safety significance (Green).

Enforcement. 10 CFR Part 50 Criterion V, "Instructions, Procedures, and Drawings," states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Contrary to this, from August 9, 1973, to September 8, 2009, the licensee failed to prescribe documented instructions into procedures that would ensure that the plant could be safely shutdown at an external flood elevation of 1009.3 feet mean sea level. Specifically, the licensee failed to ensure that station procedures would prescribe steps to mitigate external flooding through open wall penetrations in the intake structure, up to 1009.5 feet mean sea level. Because this violation was of very low safety significance and it was entered into the corrective action program as CR 2009-4166 and CR 2009-6195, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000285/2009006-02, "Inadequate Flood Protection for the Intake Structure."

b.3 Failure to Update Flood Protection for Safety-Related Buildings

Introduction. The team identified an unresolved item concerning external flood protection for plant areas considered vital to allow the reactor to achieve cold shutdown. Specifically, the issue concerns: (1) the ability of the licensee to protect the Fort Calhoun Station auxiliary building, intake structure, and turbine building basement from external floods up to flooding elevation 1013 feet mean sea level as stated in USAR and station procedures; and, (2) upon receiving new flooding information in November 2003, the licensee was required to update the USAR.

Description. The team determined flood elevations at Fort Calhoun Station by reviewing USAR Chapter 2.7, "Hydrology," USAR Chapter 9.8, "Raw Water System," and Technical Specifications 2.16, "River Level."

USAR Section 2.7.1.2 states, in part:

> "The design flood elevation of 1,006 feet based on a 0.1 percent probability flood is considered conservative. Without special provisions, the plant can accommodate flood levels of up to 1,007 [feet mean sea level]. Steel flood gates are permanently

mounted above and adjacent to openings in structures containing equipment required for a safe and orderly plant shutdown. In the event of high water levels, these flood gates can be installed to provide protection to a level of 1,009.5 [feet mean sea level]. In the Intake Structure, protection to 1009.5 [feet] msl is accomplished with flood gates and sandbagging. The plant can be protected by sandbags, temporary earth levees and other methods to allow a safe shutdown with a flood elevation of 1,013 [feet mean sea level]."

USAR Section 9.8.6 states, in part:

> "Protection for the raw water pumps and their drives against floods is provided at three elevations as indicated on Figure 9.8-1. The pumps are permanently protected against any water level up to elevation 1007.5 feet by the Class I concrete substructure of the intake building. Protection is provided to elevation 1009.5 feet by sandbags around the traveling screen areas and by gasketed steel closures at exterior doorway openings in the intake structure reinforced concrete perimeter walls. Protection to elevation 1014.5 feet is provided by additional sandbags around the traveling screen areas, and by supplementing the intake structure perimeter walls with sandbags. The water level inside the intake cells can be controlled by positioning the exterior sluice gates to restrict the inflow into the cells."

Technical Specification 2.16, "Basis," states:

> "The maximum Missouri River level of 1009 feet is the level at which the installed flood gates will protect the plant. Any increase in river level will require sand bagging to repel the water to a maximum flood level of 1014 feet or greater."

When the licensee determines it is necessary to protect the plant at flood levels greater than 1009.3 feet, the licensee implements procedure GM-RR-AE-1002, "Flood Control Preparedness for Sandbagging." Step 7.4 of the procedure states:

> "The primary focus for flood protection should be directed to those facilities which are considered vital with respect to nuclear safety and credited with flood protection in the IPEEE Flooding evaluation, Reference 2.3. These facilities shall be protected at the sacrifice of the other facilities if site conditions warrant. The vital facilities are: Auxiliary Building, Intake Structure, and Turbine Building Basement."

Reference 2.3 of procedure GM-RR-AE-1002 is the IPEEE for Fort Calhoun Station Section 5.2, "External Flooding," Table 5.2.3, which credits flood protection up to 1013.5 feet mean sea level by sandbagging for the switchyard, auxiliary building, intake structure, and turbine building. Table 5.2.3, "Impact of Periodic Flood due to Rain and Snow," comments that "severe core damage results if either intake or auxiliary building sandbagging fails."

Attachment 9.5 of procedure GM-RR-AE-1002 gives specific instructions that plant operators would use to protect from flood crest above 1009 feet mean sea level. The attachment notes that sandbags would be tied and draped over the top of floodgates to supplement the protection capability to the projected flood crest. Specifically, the attachment states, “Place additional sandbags on top of the floodgates to raise the protection against the expected crest of the flood.”

During the inspection, the team discussed, with the licensee, how protection of vital facilities against floods would occur at flood levels above the probable maximum flood level of 1009 feet mean sea level. As a result of this discussion, the team determined that stacking and draping sandbags at a height of four feet over the top of floodgates would be insufficient to protect the vital facilities. The sandbags would be insufficient because the cross section on the top of the floodgates is too small to support a stacked sandbag configuration that would retain four feet static head of water.

The team determined that the failure to protect the auxiliary building, intake structure, and turbine building to an external flood height of 1013 feet mean sea level is a performance deficiency. Specifically, the licensee failed to meet the self imposed standards of USAR Section 2.7.1.2 and USAR Section 9.8.6. In addition, the licensee failed to meet the self imposed standard of Step 7.4 in procedure GM-RR-AE-1002, “Flood Control Preparedness for Sandbagging.”

The licensee disagreed with the team’s determination that this is a performance deficiency because Fort Calhoun Station is designed to protect to the design basis probable maximum flood height of 1009.3 feet mean sea level. The licensee believes that any flooding above the probable maximum flood level is incredible because it would involve a dam break upstream of the site and dam failure was considered incredible based on Army Corps of Engineers letter to Omaha Public Power District, dated December 12, 1967. However, the team believes, because the licensee stated flood protection would occur up to 1013 feet mean sea level in the USAR and station procedures, the licensee self-imposed a standard to provide flood protection up to 1013 feet mean sea level. Additionally, in November 2003, the licensee received new information from the Army Corps of Engineers that determined flood levels could be potentially higher than what was evaluated in 1967.

Because more inspection is necessary to resolve this issue, the issue is considered an unresolved item pending further NRC Region IV review. The NRC Region IV review will determine:

1. If the failure to meet the self imposed standard of flood protection up to 1013 feet is a performance deficiency in accordance with NRC Manual Chapter 0612.

2. If a violation of NRC requirements is associated with the performance deficiency because the licensee did not update the flooding design basis when new information was received in November 2003.

This unresolved item is identified as URI 05000285/2009006-03, “Failure to Update Flood Protection for Safety Related Buildings.”

.3 **Results of Reviews for Operating Experience:**

.3.1 Inspection of Generic Letter 1988-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment"

a. Inspection Scope

The team reviewed Generic Letter 1988-14 associated with instrument air problems causing adverse effects on safety-related equipment. The review was performed to ensure that the licensee's actions were adequate to ensure that the safety-related equipment performed as required. The team reviewed the USAR, technical specifications, design basis documents, calculations, modification documents, maintenance documents, and surveillance testing to determine if the licensee had completed all of the items listed in their commitments to Generic Letter 88-14. Additionally, the team reviewed preventive maintenance documents to ensure that the licensee had included desiccant replacement and changing air filters in the instrument air system. In addition, the team reviewed an abnormal operating procedure and training on the procedure to ensure that the procedure met all operator requirements in the event of a loss of instrument air.

b. Findings

No findings of significance were identified.

.3.2 Inspection of Generic Letter 1988-17, "Loss of Decay Heat Removal"

a. Inspection Scope

The team reviewed the Generic Letter 1988-17, which involved actions taken to address identified deficiencies in procedures, hardware, and training related to loss of decay heat removal when the reactor is not at power. In the letter, the NRC requested status of the licensee's implementation of eight recommended expeditious actions. To verify that the licensee adhered to these actions, the team reviewed licensee response letters, NRC evaluation reports in response to the letters, NRC inspection reports, related operations procedures, instrument calibration procedures, and instrumentation vendor information on calibration practices. Additionally, the team reviewed the schedule and completion of six programmed enhancement recommendations.

b. Findings

No findings of significance were identified.

.3.3 Inspection of Information Notice 2005-30, "Safe Shutdown Potentially Challenges by Unanalyzed Internal Flooding Events and Inadequate Design"

a. Inspection Scope

The team reviewed the licensee's disposition of Information Notice 2005-30. This information notice discussed recent industry events where it was discovered that safe shutdown was potentially challenged by unanalyzed flooding events and inadequate design of safety-related systems. Specifically, the physical arrangement of safety-related systems essential to achieve safe shutdown made these systems vulnerable to flooding originating from failures of nonsafety-related systems located in the turbine building. The team reviewed the disposition of the Information Notice as documented by licensee in OE-2007-4005.

b. Findings

No findings of significance were identified.

.3.4 Inspection of Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems of Cause Plant Transients"

a. Inspection Scope

The team reviewed Information Notice 2002-12 and Generic Letter 2007-01, which documented the concern of inaccessible power cables. In addition, the team performed a walkdown of portions of the safety-related underground duct banks that contained the 5 kV cable for the raw water pump motors to assess the installation configuration, material condition, and potential vulnerability to hazards.

b. Findings

b.1 Inadequate Corrective Actions to Ensure the Reliability of the Raw Water Pump Power Cables

Introduction. The team identified a Green, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for failure to take effective corrective action following the discovery of water intrusion in manholes MH-5 and MH-31 in 1998, 2005 and 2009. Specifically, the licensee failed to take corrective action to establish an appropriate monitoring frequency that would mitigate potential common mode failure of raw water 5kV motor cables in underground ducts and manholes.

Description. The motors driving the raw water pumps, AC-10A through AC-10D, are powered through 5 kV cables routed through duct banks and manholes (MH-5 & MH-31) between the plant building and the intake structure.

USAR Section 8.5, Cable Installation, Section 8.5.1.f states:

> "The E prefixed cables inside the screenhouse [intake structure] and between the plant building and screenhouse are routed in separate conduits, tray sections, or in separate duct bank conduits."
>
> "There is one manhole between the pull boxes and the screen house. The cables are in cable trays and the routing is in conformance with the Cable and Conduit Schedule Notes (Figure 8.5-1). There is a 6" thick concrete wall separating cable trays with EA and EC cables from cable trays holding EB and ED cables."

On numerous occasions, between 1998 and 2009, the licensee discovered water in manholes MH-5 and MH-31. In response to the 1998 event, the licensee established a 5-year inspection schedule for MH-31, as documented in CR 199801719. The 5-year inspection schedule did not include MH-5 in the corrective action extent of condition.

In 2002, the licensee addressed Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," documented in CR 200200707, justifying the acceptability of lifetime submergence of the safety-related raw water 5 kV cables based on the cable manufacturer's design basis accident qualification report. As a corrective action to address Information Notice 2002-12, the licensee established a 5-year inspection schedule for MH-5.

In 2005, the licensee discovered a sufficient water level in MH-5 to completely submerge all but the top two cable trays, as documented in CR 200503247. Following the discovery of water in 2005, the licensee established an 18-month inspection schedule for MH-5. As a corrective action, engineering suggested at that time adding either a level alarm or a sump pump to MH-5.

During August 2009, MH-5 was opened on three separate occasions and water was pumped from the vault each time. The inspectors determined, at the rate of water intrusion during these occasions, the amount of water would submerge the 5 kV cable trays in an 18-month inspection schedule. At the time of this inspection neither an alarm nor a sump pump had been approved for installation to ensure the safety-related cables in MH-5 would stay above the water level.

During the inspection, the team identified that the environmental qualification report used to address Information Notice 2002-12, that justified the submergence of safety-related cables, was not conservative because the cable ratings were based on a 30-day containment spray, not an uncontrolled lifetime of submergence. Additionally, the team identified that the licensee had initially failed to include MH-5 in the corrective action for the flooding found in MH-31 in 1998; subsequently failing to determine if the 5-year inspection cycle was appropriate for MH-5 in 2002. Since 2005, the licensee postponed installation of proposed level alarm or sump pump in MH-5 multiple times. The postponement was based on fire protection requirements and budget constraints.

Based on the team's finding, the licensee entered the issue into the corrective action program as CR 2009-4216. The corrective action changed the manhole inspection schedule from an 18-month schedule to a quarterly schedule.

Analysis. The team determined that failure to take effective corrective action to ensure the reliability and capability of the safety-related cables powering the raw water pump motors is a performance deficiency. Furthermore, the finding was within the licensee's ability to foresee and correct because the licensee had multiple opportunities to correct the continuing challenge to the safety-related cables and raceway system for the raw water system. The finding is more than minor because it adversely affected the Mitigating Systems Cornerstone attribute of design control for ensuring the availability, reliability, and capability of systems that respond to Initiating Events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheets, the finding was of very low safety significance (Green) because it was a design deficiency that did not result in actual loss of safety function.

This finding has a crosscutting aspect in the area of human performance, decision making, because the licensee failed to use conservative assumptions in decision making and adopts a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, since 2005, the licensee decided to postpone installation of proposed water level control corrective actions and failed to appropriately monitor water intrusion in MH-5 and MH-31 multiple times [H.1(b)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." Contrary to this, from 1998 to September 11, 2009, the licensee failed to establish appropriate monitoring frequency to correct or mitigate water intrusion in cable vaults containing safety-related cables. Because this violation was of very low safety significance and it was entered into the corrective action program as CR 2009-4216, this violation is being treated as a noncited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000285/2009006-04, "Inadequate Corrective Actions to Ensure the Reliability of the Raw Water Pump Power Cables."

.4 Results of Reviews for Operator Actions:

The team selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. This included components and operator actions that had a risk achievement worth factor greater than two or Birnbaum value greater than 1E-6.

a. Inspection Scope

For the review of operator actions, the team observed operators during simulator scenarios associated with the selected components as well as observing simulated actions in the plant.

Inspection procedure 71111.21 requires a review of 3 to 5 relatively high-risk operator actions. The sample selection for this inspection was 5 operator actions.

The selected operator actions were:

- A bounding case major steam line break with feedwater isolation and containment air cooler failures (Scenario)

- A loss of offsite power with a fault of 125 Vdc Bus 2, the motor-driven auxiliary feedwater pump out of service, and a diesel drive auxiliary feedwater pump failure (Scenario)

- Establish emergency boration from outside the control room (Job Performance Measure)

- Cross-tie power supplies to open a power-operated valve during a loss of offsite power with an emergency diesel generator failure (Job Performance Measure)

- Align raw water system to shutdown cooling heat exchanger during a loss of coolant accident with a circulating cooling water system failure (Job Performance Measure)

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

The team reviewed a sample of problems that the licensee had identified previously and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions. In addition, condition reports written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific documents that were sampled and reviewed by the team are listed in the attachment.

4OA5 Other

.1 URI 05000285/2005011-05: Intake Structure Design

a. Inspection Scope

The team reviewed actions that the licensee completed to resolve concerns associated with the intake structure unresolved item which was reported in NRC Inspection Report 05000285/2005011. The licensee's action was to provide original intake structure design documentation for the seismic, tornado, and barge impact analysis. The licensee could not provide original design documentation, so licensee staff embarked on a reconstitution effort. The licensee has provided new calculations for the seismic design and tornado analysis; however, the barge impact was not completed.

To resolve this issue, the NRC inspectors will need to review the future barge impact analysis. From discussion, the licensee has agreed to complete the barge impact analysis by June 30, 2010. The unresolved item will remain open pending NRC review of this analysis.

b. Findings

b.1 Failure to Maintain Quality Records for the Intake Structure Design

Introduction. The team identified a Severity Level IV, noncited violation of Criterion XVII, "Quality Assurance Records," for failure to maintain original records as required by the Atomic Energy Commission's 70 draft General Design Criteria, July 1967. Specifically, the licensee failed to maintain the original records of the seismic and tornado analysis of the intake structure.

Description. Fort Calhoun Station is licensed and required to adhere to the 70 draft General Design Criteria published for comment in the Federal Register (32 FR 10213) on July 11,1967. Criterion 5, "Records Requirement," states, "Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor."

During an NRC inspection in 2005 (NRC Inspection Report 05000285/2005011), NRC inspectors requested documentation of the seismic analysis for the Fort Calhoun Station intake structure. At the time of the inspection, the information was not readily available. Since the information was not available, the licensee entered the concern into the corrective action program, as CR 200504345, to begin searching for the associated documents. The investigation concluded that the records were not retrievable.

After the licensee determined the documentation was not retrievable, the licensee reconstituted the seismic and tornado analysis of the intake structure. These analyses were available during the team's inspection.

Analysis. The team determined that the failure to meet the 70 draft General Design Criteria is a performance deficiency. This finding is assessed through traditional enforcement because the finding has the potential for impacting the NRC's ability to perform its regulatory function. Using Inspection Manual Chapter 0612, Appendix E, the finding is more than minor because the records were irretrievable. Using Supplement I of the NRC Enforcement Policy, this finding will be treated as a Severity Level IV violation. This finding was not assigned a crosscutting aspect because the underlying cause was not indicative of current performance.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records," states, in part, "Sufficient records shall be maintained to furnish evidence of activities affecting quality." Additionally, "Records shall be identifiable and retrievable." Contrary to this, prior to 2005, the licensee failed to maintain the original records for the seismic and tornado analysis for the intake structure. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as CR 2009-3769, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000285/2009006-05, "Failure to Maintain Quality Records of the Intake Structure Design."

b.2 Failure to Update Intake Structure Design in the FSAR

Introduction. The team identified a Severity Level IV, noncited violation for failure to update the final (updated) safety analysis report in accordance with 10 CFR 50.71(e). Specifically, the licensee failed to update Section 9.8, "Raw Water Systems," of the Fort Calhoun Station USAR after constructing a sheet pile alignment wall alongside the intake structure in 1982.

Description. USAR Section 9.8, "Raw Water Systems," states, "The intake structure is a massive concrete building set just back of the harbor line of the river." Furthermore, "The river bottom slopes downward from the bank to the thread of the channel, thus keeping boats and barges away from the actual harbor line. Any blow that could be struck by such a vessel would be a glancing one at worst on the armored wall noses and any damage to the structure itself is considered unlikely."

In 1982, the licensee installed a sheet pile alignment wall on the upstream side of the intake structure, as documented in Modification Request FC-81-106. This modification removed the slope from the river bottom. Additionally, recent sounding records indicate the river bottom near the intake structure is approximately the same depth as the center of the channel, thus, invalidating the USAR statement.

Analysis. The finding is a performance deficiency because the licensee failed to update USAR Section 9.8, as required by 10 CFR 50.71(e). The finding is assessed through traditional enforcement because the finding has the potential for impacting the NRC's ability to perform its regulatory function. The finding is more than minor because the finding is determined to have a material impact on safety. Specifically, it is not certain that a river vessel will strike the intake structure. Using Supplement I of the NRC Enforcement Policy, this finding will be treated as a Severity Level IV violation. This finding was not assigned a crosscutting aspect because the underlying cause was not indicative of current performance.

Enforcement. 10 CFR 50.71, "Maintenance of records, making of reports," paragraph (e) states, "Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." Contrary to this, since September 30, 1982, the licensee did not update USAR Section 9.8. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as CR 2009-3927, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000285/2009006-06, "Failure to Update Intake Structure Design."

4OA6 Meetings, Including Exit

On September 17, 2009, the team leader presented the preliminary inspection results to Mr. D. Bannister, Vice President and Chief Nuclear Officer, Fort Calhoun Station, and other members of the licensee's staff.

On December 28, 2009, the team leader conducted a telephonic final exit meeting with Mr. J. Reinhart, Site Vice President, and other members of the licensee's staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

4OA7 Licensee Identified Violations

None.

Attachments: Supplemental Information

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Reinhart, Site Vice President
D. Bannister, Vice President, Nuclear Operations
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D. Trausch, Assistant Plant Manager
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M. Elzway, Senior Nuclear Design Engineer – Electrical
D. Gage, Senior Instructional Technician
A. Chladil, Senior Operations Engineer
D. Rollins, Reliability Engineer
K. Dworak, Working Crew Leader – Electrical Maintenance
T. Bottum, System Engineer
E. Linzer, Licensed Operator
J. Carlson, Design Engineer
A. Filips, Design Engineer
A. Koenig, System Engineer
E. Matzke, Regulatory Compliance
J. Bock, Maintenance Support Clerk, Work Management
K. Hyde, Design Engineer
P. Cronin, Operations
K. Landis, Member, SARC

NRC Personnel

D. Loveless, Senior Reactor Analyst
J. Kirkland, Senior Resident Inspector
J. Wingebach, Resident Inspector
K. Manoly, Nuclear Reactor Regulation
R. Lantz, Chief, Operations Branch
W. Schaup, Acting Resident Inspector
L. Wilkins, Project Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285/2009006-03	URI	Failure to Update Flood Protection for Safety Related Buildings (Section 1R21.2.15)

Opened and Closed

05000285/2009006-01	NCV	Inadequate Assessment of Seismic Qualification of Raw Water Pumps (Section 1R21.2.15)
05000285/2009006-02	NCV	Inadequate Flood Protection for the Intake Structure (Section 1R21.2.15)
05000285/2009006-04	NCV	Inadequate Corrective Actions to Ensure the Reliability of the Raw Water Pump Power Cables (Section 1R21.3.4)
05000285/2009006-05	NCV	Failure to Maintain Quality Records of the Intake Structure Design (Section 4OA5.1)
05000285/2009006-06	NCV	Failure to Update Intake Structure Design (Section 4OA5.1)

Discussed

05000285/2005011-05	URI	Intake Structure Design (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Condition Reports

CR 199701234
CR 199701278
CR 200401046
CR 2004-2961
CR 2004-4093
CR 200501120
CR 200503247
CR 2005-4247
CR 2005-4652
CR 2006-2694
CR 2006-4706
CR 2006-5723
CR 200700618
CR 200700620
CR 200700707
CR 200700712
CR 200700725
CR 200700756
CR 2007-2056
CR 2007-2580
CR 2007-2731
CR 2007-3351
CR 2007-4858
CR 2008-1592
CR 2008-1788
CR 2008-3154
CR 2008-3283
CR 2008-3855
CR 2008-3864
CR 2008-4951
CR 2008-5064
CR 2008-5160
CR 2008-5529
CR 2009 4198
CR 2009-1698
CR 2009-2195
CR 2009-2205
CR 2009-2228
CR 2009-3707
CR 2009-3728
CR 2009-3786
CR 2009-3926
CR 2009-3930
CR 2009-3942
CR 2009-3957
CR 2009-3967
CR 2009-4216
CR 2009-4218
CR 2009-4219
CR 2009-4230
CR 2009-4240
CR 2009-4241
CR 2009-4271
CR 2009-4276
CR 2009-4277
CR 2009-4278
CR 2009-4293
CR 2009-4304
CR 2009-4311
CR 2009-0948
CR 2009-1202
CR 2009-2465
CR 2008-6353
CR 2008-6439
CR 2008-6850
CR 2009-0104
CR 2009-0349
CR 2008-6489
CR 2002-3065
CR 2009-0470
CR 1999-1602
CR 2002-3065
CR 2006-5427
CR 2006-5477
CR 2008-3282
CR 2008-7290
CR 2006-5477
CR 2007-0694
CR 2008-2304
CR 2008-2857
CR 2008-6824
CR 2008-3076
CR 2008-3600
CR 2006-3808
CR 2008-7228
CR 2009-0650
CR 2009-0656
CR 2008-0449
CR 2009-2750
CR 2008-2569
CR 2008-2729
CR 2009-1717
CR 2007-3763
CR 2006-3985

Calculations

EA 90-057, “Degraded Voltage Calc 4160/480 V,” Revision 7

EA 91-084, “Breaker-Fuse Coordination DG Loading Transient Analysis,” Revision 7

EA 99-006, “4160 Bus Fast Transfer Analysis,” Revision 9

EA92-072, “DG Loading Transient Analysis,” Rev. 5

EA-FC-01-011, “Low Temperature Overpressure Protection for 20 EFPY,” Revision 0

EA-FC-90-057, “Section 5.1, OPLS Setpoint Derivation,” Revision 9

EA-FC-91-42, “Calibration procedure Set point determination in support of MR-FC-89-013,” Revision 1

EA-FC-92-072, “DG Loading Transient Analysis,” Revision 5

EA-FC-99-005, “Load Parameters Database Documentation,” Revision 5

Eaton Metal Seismic Calculation, Tag FW-19 (EFWST), 04/01/70

FC 04002, “480 V Load Center Breaker Settings,” Revision 0

FC00220, “Switchgear and Cable Spreading Room Heat up Analysis,” Revision 0

FC01274, “Minimum Accumulator Pressure Required,” Revision 0

FC01414, “Exterior Wall Design, Diesel Generator Room, West Wall, East Wall,” Revision 0

FC03104, “Stress Analysis for Sub-System RW-309A,” Revision 0

FC04270, “Instrument Air Capacity,” Revision 1

FC05168, “Evaluation of Stroke Time on Valves YCV-1045, and YCV-1045A/B,” Revision 0

FC05292, “Design of Accumulator and Miscellaneous Control Air System Component mounting for MR-FC-88-17,” Revision 2

FC05346, “Accumulator Sizing for Auxiliary Feedwater Control Valves HCV-1107A/B, HCV-1108A/B, HCV-1105, HCV-1106, FCV-1368,” Revision 2

FC05374, “Starting Air for DG-1 and DG-2: Leakage Criteria, Air Dewpoint Criteria, 5 Start Capability,”

FC05587, “EDG Instruments / Relief Valve Setpoints,” Revision 1

FC05603, “DC Short Circuit,” Revision 6

FC05658, “CQE Design Basis Open Item Resolution,” Revision 4

FC05690, “Battery Load Profile and Voltage Drop,” Revision 5

FC05690, “Battery Load Profile and Voltage Drop,” Revision 7

FC05773, “Evaluation of Stroke Time on Valves YCV1045, YCV-1045A/B (Main Steam System,” Revision A

FC05839, “CCW Return Temperature foe DBA-A, Single Failure LOIA Accident Scenario with River Water Temperature equal to 90 degrees F and Minimum River Water Elevation,” Revision 0

FC05839, “CCW Return Temperature for DBA-A, Single Failure LOIA Accident Scenario with River Water Temperature equal to 90 degrees,” Revision 0

FC05905, “Auxiliary Building HVAC Cooling Load,” Revision 0

FC05916, “Operating Temperature Limits for DG-1 and DG-2,” Revision 4

FC06102, “Switchgear Room Heat up Analysis,” Revision 0

FC06148, “Auxiliary Feedwater Storage Requirements,” Revision 5

FC06284, “EFW Storage Tank Level Indication Total Loop Uncertainty Calc.,” Revision 1

FC06319, “USI A-46 / IPEEE Outlier Resolution and Detailed HCLPF for Tank FW-19,” Revision 0

FC06621, “Containment Air Cooler Thermal Hydraulic Analysis for Accident with LOOP,” Revision 4

FC06638, “Capacity of Fuel Oil Tank FO-38,” Revision 0

FC06639, “Containment Spray Pump Minimum Performance Requirement,” Revision 5

FC06642, “Uncertainty Calculations to Support ISI Testing,” Revision 3

FC06648, “FW-54 Hydraulic Performance Limits for Preventive Maintenance Testing,” Revision 0

FC06669, “Heat Removal Success Paths to Maintain RCS Temperature Below 300 Degrees F for FCS,” Revision 0

FC06877, “Ft. Calhoun Low Temperature Overpressure Protection Analysis in Support of Steam Generator Replacement,” Revision 1

FC07243, “FCS – LOCA Long Term Sump Temperature Analysis with No Spray,” Revision 0

FC07373, “IA Accumulator Pressure under Containment DBA Temperature Conditions,” Revision 0

FC07495, “Intake Structure Seismic Analysis and Pile Design,” Revision 1
FC07536, “FW-6 and FW-10 Suction and Discharge Piping Friction Loss (Proto-Flo Model),” Revision 0

FC6904, “Category 1 Air Operated Valve Operator margin Analysis,” Revision 4

FC-87-46, “Instrument Air Capacity,” 09/19/2005

FL01274, “Minimum Accumulator Pressure Required,” Revision 0
GSE Request – “Determine Which Valves with Air Accumulators are Required for Safe Shutdown,” 04/06/1988
I
C-RR-VX-0408, “Inspection and Repair of Pneumatic Valve Actuators,” 10/21/2008
OI-CA-1, “Compressed Air Normal Operation,” Revision 62

Westinghouse CN-OA-06-23, “Transient Analysis to Support PRA Criteria for Fort Calhoun Station with RSGs, RPZR and RUH.” Revision 0

Westinghouse CN-RRA-06-16, “MAAP Analyses Evaluating Level 2 Success Criteria and Operator Action Times for OPPD after NSSSRP,” Revision 0

Design Basis Documents

PLDBD-IC-32, “Instrumentation and Control Systems,” Revision 30
SDBD-COMP-300, “ERF Computer and QSPDS Computer,” Revision 20
SDBD-STRUC-503, “Intake Structure,” Revision 8
PLDBD-CS-51, “Seismic Criteria,” Revision 17
PLDBD-EV-70, “Site Meteorology,” Revision 6
SDBD-AC-RW-101, “Raw Water,” Revision 34
SDBD-DG-112, “EDGs,” Revision 23
SDBD-AFW-117, “Auxiliary Feedwater,” Revision 37
SDBD-SI-CS-131, “Containment Spray,” Revision 28
SDBD-EE-200, “120 VAC Vital Distribution,” Revision 15
SDBD-EE-201, “AC Distribution,” Revision 21
SDBD-EE-202, “DC Distribution,” Revision 18
SDBD-VA-AUX-138, “DBD-Auxiliary Building HVAC,” Revision 23
PLDBD-EE-21, “Electrical Equipment,” Revision 13
SDBD-CA-IS-105, “Instrument Air,” Revision 22
SDBD-DG-112, “Design Basis Document Emergency Diesel Generators,” Revision 23
SDBD-VA-CON-139, “Design Basis Document Containment HVAC,” Revision 32
SDBD-FM-AFW-117, “Auxiliary Feedwater,” Revision 37

Drawings

0223R0455, SH. 15, File 9958, “Raw Water Pump AC-10A Power and Control,” Revision 15

0223R0455, SH. 19, File 9960, “Raw Water Pump AC-10C Power and Control,” Revision 12

0223R0455, SH. 19, File 9962, “Aux Feed Pump Power and Control Circuit,” Revision 14

0223R0455, SH. 20, File 9963, “Aux Feed Pump Sequencer Circuit,” Revision 4

11405-A-278, “Intake Structure Floor Plan P&ID,” Revision 2

11405-E-1, File 12235, “Main One Line Diagram,” Revision 44

11405-E-143, Sh. 1, “Schematic SI-3A,” Revision 27

11405-E-143, Sh. 2, “Schematic SI-3B,” Revision 3

11405-E-143, Sh. 3, “Schematic SI-3C,” Revision 4

11405-E-145, Sh. 1, “Schematic VA-3A,” Revision 12

11405-E-145, Sh. 2, “Schematic VA-3B,” Revision 4

11405-E-145, Sh. 3, “Schematic VA-7C,” Revision 5

11405-E-145, Sh. 4, “Schematic VA-7D,” Revision 4

11405-E-18, Sh. 1, “480V Bus 1B3A, 1B3A-4A 7 1B4A,” Revision 16

11405-E-180, “Electrical Penetration G-1 Wiring Diagram Sheet No. 57,” Revision 1

11405-E-298, File 12563, “Intake Structure Plans and Sections,” Revision 17

11405-E-3, File 12239, “4.16 KV Auxiliary Power One Line Diagram,” Revision 21

11405-E-300, File 12565, “Intake Structure Details,” Revision 15

11405-E-319, File 12582, “Site Plan Underground Ducts and Manholes,” Revision 56

11405-E-320, File 12583, “Underground Ducts and Manholes Sec. and Details 1,” Revision 22

11405-E-321, File 12584, “Underground Ducts and Manholes Sec. and Details 2,” Revision 9

11405-E-4, File 12240, Sh. 1, “480 V Auxiliary Power One Line Diagram,” Revision 31

11405-E-5, File 12241, Sh. 2, “480 V Auxiliary Power One Line Diagram,” Revision 29

11405-E-8, Sh. 1, “125 Volt DC Misc Power Distribution Diagram P & ID,” Revision 62

11405-M0253, “Composite Flow Diagram Steam Generator Feedwater and Blowdown P&ID,” Revision 46

11405-M100, “Raw Water Flow Diagram P&ID,” Revision 97

11405-M-100, “Raw Water Flow Diagram P&ID,” Revision 97

11405-M-252 Sheet 2, “Flow Diagram Steam P&ID,” Revision 100

11405-M-253, “Steam Generator Feedwater and Blowdown P&ID,” Revision 46

11405-M-292, “Service Building Composite Sections and Details,” Revision 20

11405-M-294, “Intake Structure Composite Plan at El. 985’-0,” Revision 8

11405-M-295, “Intake Structure Composite Plan at El. 1007’-6,” Revision 17

13007.18-EC-18A, “Diesel Generator Enclosure Plan and Details,” Revision 6

136B2341, Sh. 24, “Elementary Diagram Electrical Control 480 V System,” Revision 18

136B2341, Sh. 25, “Elementary Diagram Electrical Control 480 V System,” Revision 16

136B2341, Sh. 26, “Elementary Diagram Electrical Control 480 V System,” Revision 16

161F531, File 9385, “Main Three Line Diagram,” Revision 46

2C-4825, “Raw Water Pump AC10A, B, C, D,” Revision 5

51B8608, “80 Actuator 657-ES Diaphragm Actuated Control Valve,” Revision C

B120F07001, “Start Air System Schematic DG-1 P&ID,” Revision 34

B120F07001, “Starting Air System Schematic DG-2 P&ID,” Revision 25

B-23866-414-307, “Elementary Wiring Diagram Electrical Pump Motor Control,” Revision 01

B-23866-414-307, “Elementary Wiring Diagram Pump Motor Control,” Revision 1

B-4022, File 20255, “Aux Feedwater Pump relay Panel AI-190, Elementary Diag.” Revision 3

D-4025, “Battery Room Ventilation Rm 54 & 55, MR #78-49,” Revision 0

D-4159, File 37777, “Schematic Diagram, Solenoid Operated Valves,” Revision 7

D-4252, “CQE Piping Isometric Seismic Subsystem # RW-309A,” Revision 2

D-841, File 41593, “C&D LC Cell Discharge Curve,” Revision 0

E-23866-210-110, “Reactor Coolant System P&ID,” Revision 19

E-23866-210-120, “Composite Flow Diagram – Chemical & Volume Control System P&ID,” Revision 42

E-23866-210-130, “Safety Injection and Containment Spray System Flow Diagram P&ID,” Revision 100

E-23866-210-130, Sheets 1-3, “Safety Injection and Containment Spray System Flow P&ID,” Revision 98

Eaton Drawing number 80055, “12’ Dim. X 23’ Dim. O.A.L. Diesel Fuel Oil Tank FO-1,” Revision 7

EM-1368/1369, SH. 1, File 15861, “Instrument and Control Equipment,” Revision 9

EM–770, Sh. 1, “Instrument and Control Equipment List”

File 12234, “Simplified One Line Diagram, UFSAR Figure 8.1,” Revision 131

M-11405 M-2 Sh. 3, “Auxiliary Building Heating and Ventilating Flow Diagram P & ID,” Revision 56

M-11405 M-88, “Auxiliary Building Ventilation, Elev. I011’-0” & 1013’-0,” Revision 31

NP207B9607, SH. 1, File 6456, “Transformer T1A-3 Name Plate,” Revision 4

NP207B9607, SH. 2, File 6456, “Transformer T1A-4 Name Plate,” Revision 4

OPPD File 03352, Revision Sheet 37564, “Design Drawings for HCV-482A and HCV-482B,” Revision 2

T-31895-1, File 45836, “AFW Pump Curve,” Revision 1

XC-545-S-107, “Seismic Restraints for Raw Water Pumps,” Revision 1

Engineering Reports

EA-FC-91-014, “Effect of Loss of Cooling Water on SI/CS Pumps,” Revision 1
Bulletin No. C-671-1, “Model ‘C’ Double Acting (DA) – Models 521-C, 721-C, 722-C, 731-C, 732-C, 733-C,” June 1971

EA-FC-94-003, “Alternate Seismic Criteria and Methodologies (ASCM),” Revision 0

EA-FC-07-010, “Use of Low Sulfur Fuel Oil,” Revision 1

“Failure Analysis of 4160 VAC Breaker Offset Rods,” 3/15/2007

Maintenance Work Orders

WO 00276516 01
WO 00175432 01
WO 00321934 01
WO 00312150 01
WO 00277812 01
WO 00244509 01
WO 00310812 01
WO 00241574 01
WO 00296907 01
WO 00265654 01
WO 00253100 01
WO 00286405 01
WO 00329489 00
WO 00087589 01
WO 00031060 01
WO 00087579 01
WO 00324114 01
WO 00333468 01
WO 00295611 01
WO 00298304 01
WO 00331811 01
WO 00168582 01
WO 00218003 01
WO 00217421 01
WO 00208932 01
WO 00252217 01
WO 00298630 01
WO 00262159 01
WO 00262311 01
WO 00262407 01
WO 00202383 01
WO 00255220 01
WO 00216528 01
WO 00201780 01
WO 00085367 01
WO 00024996 01
WO 00022219 02
WO 00021659 01
WO 298858-02
WO 317333-01
WO 00265982 01
WO 00265983 01
WO 00319018 01
WO 00320240
WO 00349416 01
WO 00349416 02
WO 00349416 03
WO 00349417 01
WO 00349417 02
WO 00349417 03
WO 00220060
WO 00307237
WO 00293474
WO 00258406
WO 00264502
WO 00294859
WO 00308333
WO 00303244
WO 00266477
WO 00261465
WO 00170253

Modifications

DCN 2823 MR-FC-97-027, “Electrical System Analysis,” Revision 0
EA-06-043, “Containment Fan Coolers Design Basis,” Revision 2
EC 06209, “Containment Air Cooling Coils Post-Accident Heat Removal Performance,”
EC 30663, “GSI-191 Implementation,” Revision 0
EC 31237, “Install Miscellaneous Point-of-Use Instrument Air Filters (Outage),” Revision 0
EC 35300, “DG Voltage Regulator/Exciter Replacement Scope Statement”
EC 37306, “Install miscellaneous Point-of –Use Instrument Air Filters,” Revision 03
EC 39208, “Replace Moore Lamb’s Wool Instrument Air Filters,” Revision 4

Operator Acton Items

“Human Reliability Worksheets,” Fort Calhoun Station, July 20, 2009
JPM-new, “Align Raw Water Backup Cooling to SDC HX AC-4A,” Revision 0, 09/07/2009
JPM-new, “Establish Once-Through-Cooling with 1A3 Deenergized,” Revision 0, 09/07/2009
PRA Document OPER-50, “Human Failure to Initiate Emergency Boration”
PRA Document OPER-53, “Operator Fails to Cross-Tie Buses for Second PORV”
PRA Document WHFFRWBKUPS, “Operator Fails to Line Up RW Backup Flow”
PRA Document XFW10O54, “Human Failure to Feed with FW-10 or -54 Following Loss of DC”
PTC No. 0008, “Emergency Boration form Outside the Control Room,” Revision 11, 08/07/2009
Simulator Evaluation Guide 2009CDBI-1, “UHE RC-2A in Containment,” Revision 1
Simulator Evaluation Guide 2009CDBI-2, “Loss of DC Bus #2,” Revision 0

White Paper by FCS PRA Staff, “Discussion of Human Failure Events OPER-53, WHFFRWBKUPS, and XFW10OR54,” 09/15/2009

Procedures

AOP-01, “Acts of Nature,” Revision 23

AOP-06, “Fire Emergency,” Revision 21

AOP-11, “Loss of Component Cooling Water,” Revision 14

AOP-12, “Loss of Containment Integrity,” Revision 6

AOP-16, “Loss of Instrument Bus Power,” Revision 16

AOP-19, “Loss of Shutdown Cooling,” Revision 14

AOP-30, “Emergency Fill of EFW Storage Tank”, Revision 9

AOP-31, “161 kV Grid Malfunctions,” Revision 9

ARP-AI-79/A79, “Annunciator Response Procedure A79 Local Annunciator A79 Air Dryer Panel,” October 18, 2007

ARP-CB-1,2,3/A2, “Annunciator A2 – Shutdown Cooling Flow HI-LO,” Revision 33

ARP-CB-1,2,3/A4, “Annunciator A4 – RCS Refueling Level LO,” Revision 23

CH-ANL-MI-0017, “Determination of Dewpoint by Sahara Moisture Analyzer,” August 24, 2006

EM-ST-EE-0001, “Battery 1 Monthly Surveillance,” Revision 13

EM-ST-EE-0002, “Battery 2 Monthly Surveillance,” Revision 12

EM-ST-EE-0003, “Battery 1 Quarterly Surveillance,” Revision 17

EM-ST-EE-0004, “Battery 2 Quarterly Surveillance,” Revision 17

EM-ST-EE-0005, “Battery 1 Capacity Discharge Surveillance Test,” Revision 21

EM-ST-EE-0006, “Battery 2 Capacity Discharge Surveillance Test,” Revision 19

EOP-00, “Standard Post Trip Actions,” Revision 24

EOP-03, “Loss of Coolant Accident,” Revision 35

EOP-06, “Loss of All Feedwater,” Revision 16

EOP-07, “Station Blackout”

EOP-20, “Functional Recovery Procedure,” Revision 23

ERF-VNV-RIA-001, “Digital Systems, Verification and Validation Plan (V&V),” Revision 0

FCSG-15, “Fort Calhoun Station Safety Manual,” Revision 25

FCSG-15-29, “Ladders and Powered Platforms,” Revision 3

FCSG-15-31, “Fall Protection Requirements,” Revision 7

FCSG-15-42, “Standing/Stepping on Plant Equipment,” Revision 0

FCSG-20, “Abnormal Operating Procedure and Emergency Operating Procedure Writer’s Guide,” Revision 7

GM-RR-AE-1002, “Flood Control Preparedness for Sandbagging”, Revision 9

IC-CP-01-0119, “Calibration of Reactor Coolant Shutdown Narrow Range Level, Loop L-119,” Revision 9

IC-CP-01-0119, “Calibration of Reactor Coolant Shutdown Narrow Range Level, Loop L-119,” Revision 10

IC-CP-01-0197, “Calibration of Reactor Coolant Shutdown Wide Range Level, Loop L-197,” Revision 9

IC-CP-01-1183, “Calibration of EFW Storage Tank FW-19 Level Loop L-1183,” Revision 17

IC-ST-AFW-0001, “Auto Initiation of AFW Functional Check of Initiation Circuits,” Revision 14

IC-ST-AFW-0006, “Auto Initiation of AFW Functional Check of Logic Matrices,” Revision 1

IC-ST-IA-3002, “CVCS Instrument Air Accumulator Check Valve Leakage Test,” 05/29/2008

IC-ST-MS-0010, “Channel Calibration of Steam Generator RC-2A and RC-2B Wide Range Pressure Loops A/P-913 and A/P-914,” Revision 14

IC-ST-SA-3001A, “DG-1 Starting Air Compressors Discharge Check Valves Exercise Test,” Revision 0

IR-RR-VX-0408, “Inspection and Repair of Pneumatic Valve Actuators,” Revision 21

MM-ST-DG-0001, “DG 1 Inspection,” Revision 74

MM-ST-DG-0002, “DG 2 Inspection,” Revision 46

NOD-QP-36, “Grid Operations and Control of FCS Switchyard,” Revision 18

OI-AFW-4, “AFW Startup and System Operation,” Revision 73

OI-AOV-01, “Air Operated Valve Manual Operation,” Revision 14

OI-CS-1, “Containment Spray-Normal Operation,” Revision 34

OI-EE-4, “120 VAC System Normal Operation,” Revision 42

OI-EG-3, “EMS Post-FCS-Trip 161 kV Voltage Prediction and Switchyard Status,” Revision 7

OI-FO-1, “Fuel Receipt (FO-1, -10, -27, -32, -43A and FO-43B),” Revision 27

OI-RC-2A, “RCS Fill and Drain Operations,” Revision 66

OI-RC-2A, Attachment 4, “RCS Draining,” Revision 66

OI-RC-6, “Reactor Coolant Pump Operation,” Revision 70

OI-RC-7, “RCS Pressure Control-Normal Operation,” Revision 16

OI-SC-1, “Shutdown Cooling Initiation,” Revision 48

OI-SC-3, “Alternate Shutdown Cooling Utilizing Containment Spray Pumps,” Revision 17

OP-3A, “Plant Shutdown,” Revision 78

OP-ST-AFW-3007, “AFW Test of Circuits and Components,” Revision 14

OP-ST-AFW-3010, “Surveillance Test Auxiliary Feedwater System Quarterly Category A and B Valve Exercise Test,” Revision 7

OP-ST-DG-0001, “DG 1 Check,” Revision 67

OP-ST-DG-0001, “EDG Surveillance,” Revision 67

OP-ST-DG-0002, “DG 2 Check,” Revision 63

OP-ST-MS-3001, “Surveillance Test main Steam System Category B and C Valve Exercise Test,” Revision 21

OP-ST-RW-3003, “Raw Water System Category A & B Interface Valve Exercise Test,” Revision 13

PE-ST-VX-3004, “Surveillance Test ASME Code Relief Valve Test for the DG Air Start System Receiver Tanks,” Revision 0

QAP-6.2, “Quality Assurance Plan Corrective Maintenance”

SM-M-101, “Maintenance Work Control,” Revision 85

SO-G-30, “Procedure Changes and Generation,” Revision 115

SO-G-74, “Fort Calhoun Station EOP/AOP Generation Program,” Revision 13

SO-M-101, “Maintenance Work Control,” Revision 85

SO-M-103, “System Cleanliness”

SO-O-1, “Conduct of Operations,” Revision 78

SO-O-21, “Shutdown Operations Protection Plan,” Revision 32

SO-O-41, “Control of Operator Aids and Emergency Equipment,” Revision 116

SO-O-46, “Post Trip Reviews,” Revision 13

SO-R-2, “Standing Order Condition Report and Corrective Action,” Revision 45

SP-CP-08-FW-6, “Calibration of Protective Relays for Aux. FW Pump FW-6,” Revision 7

TBD-AOP-07, “Evacuation of Control Room,” Revision 13

TBD-AOP-17, “Loss of Instrument Air,” Revision 13

TBD-AOP-17, “Loss of Instrument Air,” Revision 13

TBD-AOP-17, “Loss of Instrument Air,” Revision 13

TBD-EOP-20, “Functional Recovery Procedure,” Revision 23

Completed Surveillance Packages

IC-IS-IA-3008, “HCV-A/B and HCV-2899A/B Air Accumulator Check Valve Operability Test,” 03/30/2009
I
C-ST-AFW-3001, “Accumulator Check Valve Test for Aux. Feed Water Pump Minimum Flow Recirculation Valves,” 04/22/08

IC-ST-IA-3001, “Safety Injection Refueling Water Tank Air Accumulator Check,” 11/18/2008

IC-ST-IA-3001, “Safety Injection Refueling Water Tank Air Accumulator Check,” 02/10/2009

IC-ST-IA-3002, “CVCS Air Accumulator Check Valve Leakage Test,” 06/06/2008

IC-ST-IA-3003, “Raw Water Instrument Air Accumulator Check Valve Operability,” 04/07/2009

IC-ST-IA-3005, “Instrument Air Accumulator Check Valve IA-HCV-2987-C Operability Test,” 05/25/2008

IC-ST-IA-3006, “Refueling Outage Instrument Air Accumulator Check Valve Operability,” 11/17/1999

IC-ST-IA-3007, “Instrument Air Accumulator Check Valve Operability Test,” 10/31/2000

IC-ST-IA-3009, “Operability Test of IA-YCV-1045-C and Close Stroke Test of YCV-1045,” 12/04/2006

IC-ST-IA-3009, “Operability Test of IA-YCV-1045-C Instrument Air Accumulator,” 01/04/2006

IC-ST-IA-3009, “Operability Test of IA-YCV-1045C Instrument Air Accumulator,” 03/30/2009

IC-ST-IA-3200, “CVCS Instrument Air Accumulator Check Valve Leakage Test,” 05/29/2008

IC-ST-SA-3001, “Starting Air Compressors Discharge Check Valves Test,” 04/08/2008

IC-ST-SI-0002, “Channel Calibration of SIRWT Low Level Monitoring Switches, Loops, ABCD/L383,” 03/12/08

IC-ST-VA-0013, “Calibration of Containment Air Cooling and Filtering Unit Flow and Pressure Drop,” 05/12/2006

MM-PM-AFW-0004, “Fuel Oil Transfer Pump (FO 37) Maintenance,” 09/17/08

OP-PM-AFW-0004, “FW-54 Monthly Full Flow Operability Verification,” 07/23/09

OP-SI-ST-3022, “Room 22 Safety Injection/Containment Spray Pumps and Valves Exercise IST,” 07/22/09, 05/07/09, 02/10/09,

OP-ST MS-3001, “Main Steam System Category B and C Valve Exercise Test,” 04/04/2009

OP-ST-AFW-3009, “AFW Pump FW-6, Recirculation Valve, and Check Valve Tests,” 07/20/09, 01/16/08, 04/13/09

OP-ST-AFW-3010, “Auxiliary Feedwater System Category A and B Valve Exercise Test,” Revision 5

OP-ST-AFW-3010, “Auxiliary Feedwater System Category A and B Valve Exercise Test,” 05/28/2009

OP-ST-AFW-3011, “AFW FW-10 Steam Isolation, Valve and Check Valve Test,” 08/13/2008

OP-ST-ESF-0009, “Surveillance Test Channel B Safety Injection, Containment Spray and recirculation Actuation Signal Test,” Revision 53

OP-ST-ESF-0010, “Surveillance Test Channel B Safety Injection, Containment Spray and recirculation Actuation Signal Test,” Revision 53

OP-ST-RC-3004, “Power Operated Relief Valves (PORVs) Low Temperature Low Pressure Exercise Test (PCV-102-1 and PCV 102-2),” 03/04/05, 09/11/06, 04/22/08

OP-ST-SI-3003, “Low Pressure Safety Injection and Containment Spray System Pump and Check Valve Tests,” 04/22/08

OP-ST-VA-0008, “Containment Ventilation system Containment Fans and Dampers Exercise,” 11/06/2006

OP-ST-VX-3002, “Auxiliary Feedwater System Remote Position Indicator Verification,” 06/06/2008

PE-ST-VX-3004, “ASME Code Relief Valve Test for the DG Air Start System Receiver Tanks Valve SA-177,” 02/02/2008

PE-ST-VX-3004, “ASME Code Relief Valve Test for the DG Air Start System Receiver Tanks Valve SA-178”

SE-ST-AFW-3006, “Auxiliary Feedwater Pump FW-10, Steam Isolation Valve and CH,” 12/02/2006

SE-ST-AFW-3006, AFW Feedwater Steam Isolation Valve and Check Valve Test,” 06/06/2006

SE-ST-VA-0008, “Control Room charcoal Filter VA-64A Replacement,” Revision 6

SP-CP-08-FW-6, “Calibration of Protective Relays for Aux. Feedwater Pump,” 04/29/2008

IC-ST-AFW-3002, “Instrument Air Accumulator/Check Valve Operability Test,” 08/05/2008

IC-CP-01-0770A, “Calibration procedure Calibration of Battery Room 54 Air Flow, LOOP F-770A,” Revision 10

IC-CP-01-0770B, “Calibration procedure Calibration of Battery Room 54 Air Flow, LOOP F-770B,” Revision 8

File 11405, “Transformers T-1A-3 and T-1Q-4 Test Data,” May 1969

EM-ST-EE-0009, “Swing Charger,” 2/14/2008

OP-ST-ESF-0002, “DG 2 Integrated Loading Test,” 2/20/2008

EM-ST-EE-0005, “Battery 1 Discharge Test,” 5/04/2008

EM-ST-EE-0006, “Battery 2 Discharge Test,” 5/18/2008

PE-ST-VX-3011, “Inspect FW-658 (FW-19 Vacuum Breaker),” 06/05/08

EM-ST-EE-0003, “Battery 1 Quarterly Surveillance,” 9/29/2008

EM-ST-EE-0003, “Battery 1 Quarterly Surveillance,” 12/22/2008

OP ST-ESF-0022, “S1-2 Automatic Load Sequencer Test,” Revision 26

OP ST-ESF-0023, “S2-2 Automatic Load Sequencer Test,” Revision 26

EM-ST-EE-0003, “Battery 1 Quarterly Surveillance,” 3/17/2009

EM-CP-05-1B3A-7, Calibration Procedure, “Calibration of Containment Vent Fan VA-3A Circuit Breaker”

EM-CP-05-1B3B-4B-3, Calibration Procedure, “Calibration of Containment Spay Pump”

EM-CP-05-1B3B-4B-4, Calibration Procedure, “Calibration of Containment Vent Fan VA-7D Circuit Breaker”

EM-CP-05-1B3C-4C-3, Calibration Procedure, “Calibration of Containment Vent Fan VA-7C Circuit Breaker”

EM-CP-05-1B4B-1, Calibration Procedure, “Calibration of Containment Spay Pump SI-3B Circuit Breaker”

EM-CP-05-1B4C-8, Calibration Procedure, “Calibration of Containment Vent Fan VA-3B Circuit Breaker”

EM-ST-EE-0005, “Battery 1 Discharge Test,” 5/17/2009

EM-ST-EE-0006, “Battery 2 Discharge Test,” 5/28/2009

FC-793, “Diesel Fuel Analysis Form-Tanker,” 07/30/09, 08/03/09

OP-ST-DG-0002, “DG 2 Inspection,” 8/12/2009

OP-ST-DG-0001, “DG 1 Inspection,” 8/26/2009

EM-ST-EE-0003, “Battery 1 Quarterly Surveillance,” 8/31/2009

Operating Experience

2007-11513, “OE24508 – Power Operated Valve Manual Operator Functional Testing,” 09/28/2009

Licensing Documents

Letter from A. Bill Beach, DRP, US NRC, to T. L. Patterson (OPPD), “NRC Inspection Report 50-285/94-21,” November 17, 1994 (Document NRC-94-323)

Letter from A. Bill Beach, DRP, US NRC, to W. G. Gates (OPPD), “NRC Inspection Report 50-285/92-16,” September 19, 1992 (Document NRC-92-367)

Letter from Anthony Bournia, NRR, U.S. NRC, to W. Gary Gates (OPPD), “Programmed Enhancements for GL 88-17,” August, 10, 1990 (Document NRC-90-289)

Letter from Anthony Bournia, NRR, U.S. NRC, to W. Gary Gates (OPPD), “Programmed Enhancements for GL 88-17,” August, 10, 1990 (Document NRC-90-289)

Letter from Anthony Bournia, NRR, U.S. NRC, to W. Gary Gates (OPPD), “Programmed Enhancements for GL 88-17,” August, 10, 1990 (Document NRC-90-289)

Letter from David L. Wigginton, NRR, U.S. NRC, to W. G. Gates (OPPD), “GL 88-17 Enhancements – FCS Unit 1,” December 5, 1991 (Document NRC-91-361)

Letter from David L. Wigginton, NRR, U.S. NRC, to W. G. Gates (OPPD), “GL 88-17 Enhancements – FCS Unit 1,” December 5, 1991 (Document NRC-91-361)

Letter from David L. Wigginton, NRR, U.S. NRC, to W. G. Gates (OPPD), “GL 88-17 Enhancements – FCS Unit 1,” December 5, 1991 (Document NRC-91-361)

Letter from K. T. Morris (OPPD) to U.S. NRC, “Response to NRC GL 88-17,” February 10, 1989 (Document LIC-89-045)

Letter from K. T. Morris (OPPD) to U.S. NRC, “Response to NRC GL 88-17,” February 10, 1989 (Document LIC-89-045)

Letter from K. T. Morris (OPPD) to U.S. NRC, “Response to NRC GL 88-17,” February 10, 1989 (Document LIC-89-045)

Letter from Patrick D. Milano, NRR, US NRC, to Kenneth J. Morris (OPPD), “Comments on OPPD Response to GL 88-17, FCS,” May 18, 1989 (Document NRC-89-207)

Letter from Patrick D. Milano, NRR, US NRC, to Kenneth J. Morris (OPPD), "Comments on OPPD Response to GL 88-17, FCS," May 18, 1989 (Document NRC-89-207)

Letter from W. G. Gates (OPPD) to Martin Virgilio, U.S. NRC, "Request for Temporary Waiver of Compliance from Provisions for TS 2.6(1)a," January 29, 1992 (Document LIC-92-019R)

Letter from W. G. Gates (OPPD) to U.S. NRC, "Completion of GL 88-17 Actions," May 21, 1992 (Document LIC-92-170R)

Letter from W. G. Gates (OPPD) to U.S. NRC, "GL 88-17 (Reference 2) Enhancements," April 4, 1990 (Document PED-FC-90-560)

Letter from W. G. Gates (OPPD) to U.S. NRC, "GL 88-17 Enhancements," April 12, 1990 (Document LIC-90-0284)

Letter from W. G. Gates (OPPD) to U.S. NRC, "GL 88-17 Enhancements," September 4, 1991 (Document LIC-91-201R)

Letter from W. G. Gates (OPPD) to U.S. NRC, "GL 88-17 Enhancements," January 31, 1992 (Document LIC-92-170R)

Letter from W. G. Gates (OPPD) to U.S. NRC, "Supplement to OPPD Responses to GL 88-17 and Bulletin 80-12," June 8, 1990 (Document LIC-90-0483)

Letter from W. G. Gates (OPPD) to U.S. NRC, "Verification of Information Contained in the USAR (1/31/92)," January 31, 1992 (Document LIC-92-014R)

LIC-89-046, Letter to USNRC from OPPD, "Response to NRC Bulletin 88-04," 02/10/89

LIC-95-0130, Letter to USNRC from OPPD, Letter number, "Phase II Response to Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE)," 06/30/95

LIC-04-0129, "LER 2004-002," 12/17/2004

LIC-05-0175, "LER 2005-006," 09/22/1995

LIC-06-0025, "Response to GL 2006-02," 04/03/2006

LIC-07-0022, "LER 2007-002," 04/09/2007

LIC-07-0033, "Response to GL 2007-01," 05/07/2007

LIC-92-129R, "Response to NOV," 06/15/1992

Miscellaneous Documents

"Post Trip Review Report of the March 15, 2008 FCS Trip," 3/22/2008

4160 V Circuit Breaker MOC Offset Rod Replacement Schedule (Undated)

EDG Emergency Diesel Generator System Health Report, First Quarter 2009

EDS Electrical Distribution System Health Report, First Quarter 2009

EDSA Network FCS_MODEL Data Check Report, 08/28/09

FACTS for Cable EA150

FACTS for Cable EC107

FACTS for Cable EC109

FACTS for Cable EC111

FACTS for Cable EC152

FACTS for Cable EC153

FCS Cable Route Report for Cable EC125

FCS Cable Route Report for Cable EC152A

FCS System Health Report, “AFW,”1Q-2009

FCS System Health Report, “ECC Emergency Core Cooling,” 1Q-2009

FCS System Health Report, “EDGs,” 1Q-2009

FCS System Health Report, “RCS Reactor Coolant,” 1Q-2009

Gilbert Spec Sheet number H-24, “FO Tank number 1 Hold Down Bands,” Revision 1

Lesson Plan 7-11-1, “Auxiliary Feedwater System,” Revision 20

Lesson Plan 7-15-17, “Loss of Secondary Heat Removal Events,” Revision 6

Lesson Plan 7-17-28, “Mitigating Core Damage,” Revision 6

Lesson Plan 7-18-10, “Introduction to EOP’s and Standard Post Trip Actions,” Revision 10

Lesson Plan 7-18-12, “Loss of Offsite Power/Loss of Forced Circulation,” Revision 9

Lesson Plan 7-18-16, “Loss of All Feedwater,” Revision 11

Lesson Plan 7-18-18, “Functional Recovery Procedure AOP-20,” Revision 15

Letter from Alan R. Wheeler and Norm Provost (Invensys Foxboro) to OPPD, “Foxboro 100 and 200 Series FBM Field Calibration,” 09/02/2009

NED-09-0086 DEN, “Report on Periodic Trending of 161 kV Power,” 06/29/2009

NUC-001-02, “Nuclear Plant Interface Coordination,” 08/05/2009

Operability Evaluation for CR 2009-2195, 05/08/2009

OPPD Internal Memo, “Electric Operations and FCS Grid/Switchyard Interface Policy Agreement,” 11/05/2000

PED-SYE-92-1265, Memorandum (OPPD) from M. T. Frans to R. W. Short, “Closure of CID 920896/01, DHR System Monitoring, Visual and Audible Indications of Abnormal Conditions, Follow up Item 9216-01,” 12/03/1992

PED-SYE-93-0742, Memorandum (OPPD) from M. T. Frans to R. W. Short, “Closure of CID 920897/01, CET Monitoring,” 06/30/1993 (Document)

PED-SYE-93-0962, Memorandum (OPPD) from M. T. Frans to R. W. Short, “Closure of CID 920898/01, ‘Review of Alarm Setpoints for Low Reactor Coolant System Level, High Core Exit Temperature and Low Shutdown Cooling Flow Did Not Appear to be Appropriate for all Operating Conditions That Might Be Encountered’,” 09/16/1993

RFP 3153, “Section H, Technical specifications, 4160 V Breaker Replacement,” 9/8/2009

Screening Evaluation Worksheet (SEWS), “FO-2-1, Day Tank,” 11/25/93

Screening Evaluation Worksheet (SEWS), “FO-2-2, Day Tank,” 11/25/93

Screening Evaluation Worksheet (SEWS), “Motor-Driven Pump AC-10A,” 02/14/92

Simulator Scenario Guide 8-12-24, “Loss of Instrument Bus Power – 1 DC Bus,” Revision 10

Simulator Scenario Guide 8-13-14, “Complete Loss of Feedwater with Condensate Recovery,” Revision 4

Simulator Scenario Guide 8-13-15, “Total Loss of Feedwater – Once-Through Cooling,” Revision 4

Simulator Scenario Guide 8-13-4, “Loss of Offsite Power,” Revision 8

System Training Manual, “Component Cooling Water System,” Volume 8, Revision 35

System Training Manual, “Electrical Distribution System,” Volume 14, Revision 35

System Training Manual, “Emergency Diesel Generator System,” Volume 16, Revision 27

System Training Manual, “Raw Water System,” Volume 35, Revision 22

TD-D243.0090, “Instruction Manual for Installation and Maintenance of Consolidated Electromagnetic Relief Valve Type 31533VX-30 with Bellows,” Revision 5

U.S Atomic Energy Commission, “Safety Evaluation of the OPPD Fort Calhoun Station, Unit 1”, Supplement 1, 04/23/72

Upper Mississippi River System, U.S. Army Corps of Engineers Omaha District, “Flow Frequency Study Hydrology and Hydraulics, Appendix F, Missouri River,” November 2003

W-List Sh. 1, “Rockbestors Cable Schedule Ampacities,” Revision 7

W-List Sh. 2, “Rockbestors Cable Schedule Ampacities,” Revision 7

WO# 48769, “Fuel Oil Storage Tank Samples” and “Diesel Tanker Truck Receipt-dated 30-July-09,” 08/13/09

Vendor Manuals

TD G080.2800, “Voltage Regulator Manual,” Revision 1

TD G080.1250 GEH-2038E, “GE Control and Transfer Switch Type SBM,” Revision 1

TD G080.1920, “Instruction Manual for Power Circuit Breakers Type AK-2//2A-15, AK-2/3/2A/3A-25 and AKU-2/3/2A/3A-25”

TD K960.0020, “Instruction Manual for Kaeser Compressed Air Filters,” December 1993

TD F130.0330, “Instruction Manual for Type 657 Diaphragm Actuator Sizes 30-70 and 87,” Revision 5

117C3250, Sh. 10, “General Electric Drawing Switches A1-30A & A1-30B,” Revision 4

Failure Probability of the Alternative Mitigation Strategy

In the significance determination process evaluation documented in Section 1R21.2.15 of this inspection report, the analyst assumed that the minimum mitigation equipment available for all postulated flooding scenarios was the licensee's alternative mitigation strategy using portable gasoline pumps described in Procedure PE-RR-AE-1002, "Installation of Portable Steam Generator Makeup Pumps." The quantification of the failure probability used, 3.2 percent, is documented below.

The analyst noted that Emergency Plan Implementing Procedure EPIP-TSC-2, "Catastrophic Flood Preparation," requests that the Site Director/Shift Manager authorize the performance of Procedure PE-RR-AE-1002 when the Corps of Engineers notifies the licensee of an upstream dam failure or the expectation of flooding above Elevation 1009 feet mean sea level.

Procedure PE-RR-AE-1002 directs the installation of two portable gasoline engine-driven pumps as shown in the drawing in Figure 1. The first pump takes a suction directly on the flood waters in the turbine building via a suction strainer and fills the Emergency Feedwater Storage Tank via fire hoses and a permanently installed process flange. The second pump takes a suction from the tank via a second flange, and pumps to the steam generators through hoses, fittings, and flanges to the feedwater system.

The analyst utilized information available in Westinghouse LTR-RAM-II-09-085, "Reliability Analysis of Refilling the EFWST Using Portable Gasoline Pumps," to create the fault tree shown in Figure 1. The analyst used the failure data from Westinghouse LTR-RAM-II-09-085 as the best estimate parameter values. For Basic Event EFW-XHE-PORTABLE, "Operators Fail to Makeup to S/Gs using Gasoline Pumps," the analyst used the SPAR-H method for human reliability analysis.

The analyst evaluated the probability that operators fail to diagnose the need for and to implement the actions specified in Procedure PE-RR-AE-1002. The diagnosis portion included the licensee recognizing that Procedure PE-RR-AE-1002 was the appropriate recovery path given the circumstances. The analyst evaluated the entire implementation of the procedural requirements under the action portion of the human reliability analysis. The following performance shaping factors were adjusted from nominal:

<u>Time</u>:

> The analyst determined that implementation of Procedure PE-RR-AE-1002 would be at least 2 days after reactor shutdown. The reactor would most likely have been in cold shutdown. Therefore, the reactor would have had to heat up to pressure and temperature and boiled the steam generators dry prior to proceeding to postulated core damage. This would have taken up to 25 hours depending on initial conditions.
>
> <u>Diagnosis</u>: Because the flood would have been an ongoing condition, the analyst determined that the operators would have "an inordinate amount of time (a day or more) to diagnose the problem. Therefore, diagnosis credit was adjusted for "Expansive Time."

Action: The analyst noted that the licensee would not start implementing this procedure until all sources of clean water and plant process equipment had failed. The analyst estimated that performing this well-documented, yet unfamiliar task would take several hours. As such, there would be "greater than five times the time required" for installation of the equipment and providing a source of gasoline. However, the time available would not be more than 50 times the required time.

Stress:

The analyst determined that the licensee personnel performing diagnosis and the operators/maintenance personnel implementing the procedure would be under an "Extreme" level of stress. The onset of the station blackout and loss of all feedwater would be sudden (though possibly anticipated), because it would initiate from the failure of last ditch flood protection and the flooding of large portions of the plant. The stressing situation would have persisted for days as the flood waters would likely have been rising for weeks, requiring additional work hours and hard work. Workers would realize that they were surrounded by the flood waters and could not leave if conditions worsened. The catastrophic failure in this scenario would lead operators to understand that failure would result in core damage and permanent loss of the plant.

Complexity:

Diagnosis: The analyst determined that the diagnosis for this human reliability analysis was "Moderately Complex." There would be some ambiguity in what was to be diagnosed. Last minute flood protection efforts, expectation that alternatives might be viable, even the possibility that successful turbine-building flood protection would have to be torn down, complicate the diagnosis that this alternative mitigating strategy is the only viable solution.

Action: The analyst determined that implementing Procedure PE-RR-AE-1002 would be "Moderately Complex." Implementation requires installing a complex sequence of pumps, hoses and spool pieces. Coordination with the plant operating state is necessary to ensure that the steam generator safeties can be blocked open. Coordination of the two operating pumps is constantly a concern because the Emergency Feedwater Storage Tank must be full to prevent the second pump from binding.

Experience:

The SPAR-H evaluates the experience and training of operators and personnel based on the years of experience and broad consideration of training for the actions involved in the accident scenario. The analyst evaluated this and determined that operators had the nominal skills to assemble and operate the pumps and equipment necessary. Therefore, the analyst determined that the experience for both diagnosis and action would be "Nominal."

Procedures:

Diagnosis: As stated above, Procedure EPIP-TSC-2 directs the attention of the technical support center personnel to look at Procedure PE-RR-AE-1002. The procedure addresses, but does not delineate that interfacing systems should be isolated prior to flood-related loss of shutdown cooling. The procedure directs that the main steam safety valves be opened to maintain steam pressure as close as possible to atmospheric. The procedures for diagnosis are available and enhance performance, better than many alternative mitigating strategy procedures in the industry. However, they do not meet the definition of diagnostic procedures. Therefore, the analyst determined that the procedures were of "Nominal" level.

Action: The analyst determined that Procedure PE-RR-AE-1002 was available and enhances performance. The equipment is mostly staged in containers in Room 81, and drawings are readily available to assist the operators/maintenance personnel in installing the equipment. Therefore, the analyst determined that the procedures were of "Nominal" level.

Ergonomics:

Diagnosis: The analyst determined that most of the diagnosis would be done from the main control room and the technical support center. The big picture indication of plant status would be available until after most of the scenarios would have required diagnosis. (Extremely high flooding would also take out the direct current system). Therefore, the analyst determined that the ergonomics for diagnosing this condition were "Nominal."

Action: The analyst determined that the ergonomics for implementing Procedure PE-RR-AE-1002 were "Poor." Operators and maintenance personnel would be working under emergency lighting conditions. In most scenarios installed emergency lighting would have failed from battery depletion. Workers would be performing tasks around and in flood waters and in dank conditions. Operations would involve routine handling of gasoline. No steam generator indication is available at the location.

The Table 2 provides the calculations used to apply the performance shaping factors and the odds ratio. The resulting HRA non-recovery value was 2.0 percent.

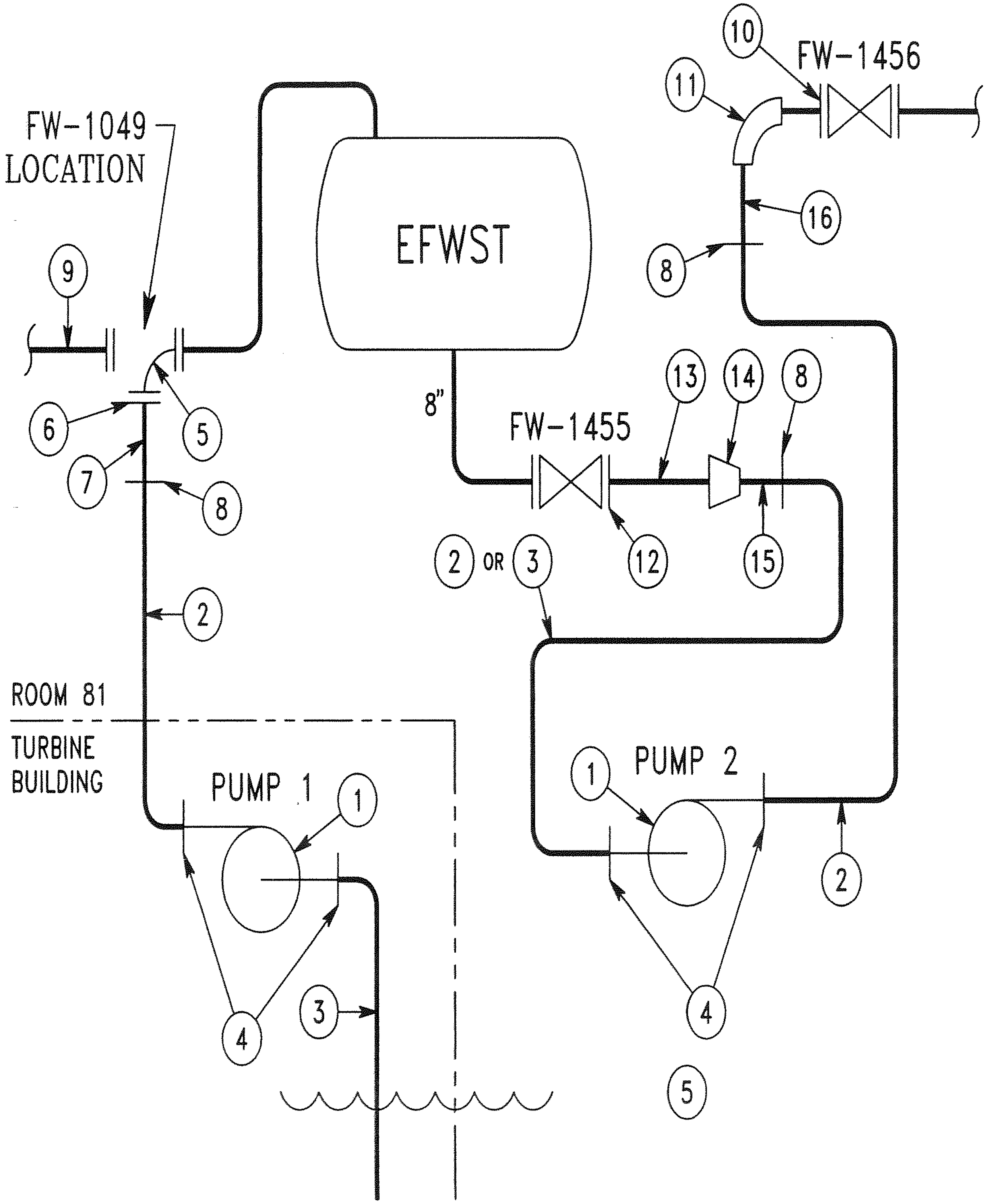
TABLE 2 Refill Steam Generator Using Gasoline Pumps				
Performance Shaping Factor	**Diagnosis**		**Action**	
	PSF Level	**Multiplier**	**PSF Level**	**Multiplier**
Time:	Expansive Time	0.01	>= 5 times	0.1
Stress:	Extreme	5.0	Extreme	5.0
Complexity:	Moderately Complex	2.0	Moderately Complex	2.0
Experience:	Nominal	1.0	Nominal	1.0
Procedures:	Nominal	1.0	Nominal	1.0
Ergonomics:	Nominal	1.0	Poor	10.0
Fitness for Duty:	Nominal	1.0	Nominal	1.0
Work Processes:	Nominal	1.0	Nominal	1.0
	Nominal	1.0E-02		1.0E-03
	Adjusted	1.0E-03		1.0E-02
	Odds Ratio	1.0E-03		9.9E-03
	Composite	0.1		10
	Failure to Refill Steam Generators Using Gasoline Pumps Probability:			**1.1E-02**

The analyst solved the fault tree, shown in Figure 2, and determined that the probability that this alternate mitigation strategy would fail was 3.23E-02. Because this strategy would be available regardless of the performance deficiency, the analyst multiplied both baseline and case conditional core damage probabilities by this value.

<u>Discussion of Licensee's Value</u>:

The licensee provided a total failure rate for the alternative mitigating strategy of 1.3E-02 in the IPEEE. The analyst's value of 3.23E-02 was calculated by evaluating the potential for equipment failure, as well as a human reliability analysis.

The analyst evaluated Westinghouse LTR-RAM-II-09-085 that analyzed a similar recovery action for the licensee. This evaluation referenced an Analysis ST-95-0147 developed by ABB Combustion Engineering, Inc., dated March 10, 1995. This report used the EPRI HRA Calculator with the HCR/ORE (cognitive) and THERP (execution) modules to evaluate the human portion of the nonrecovery. The published value was 1.0E-2, with a 10 percent conservatism added to 1.1E-2. Westinghouse LTR-RAM-II-09-085 then modified this value to 1.1E-3 because they were evaluating only the action to fill the emergency feedwater storage tank. Therefore, the analyst determined that the original value of 1.0E-2 was more applicable to the current evaluation and corroborated the analyst's HRA value of 1.1E-02.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LOCATION
EFWST
FW-1456
FW-1455
8"
2 OR 3
ROOM 81
TURBINE
BUILDING
PUMP 1
PUMP 2
1
2
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Figure 1

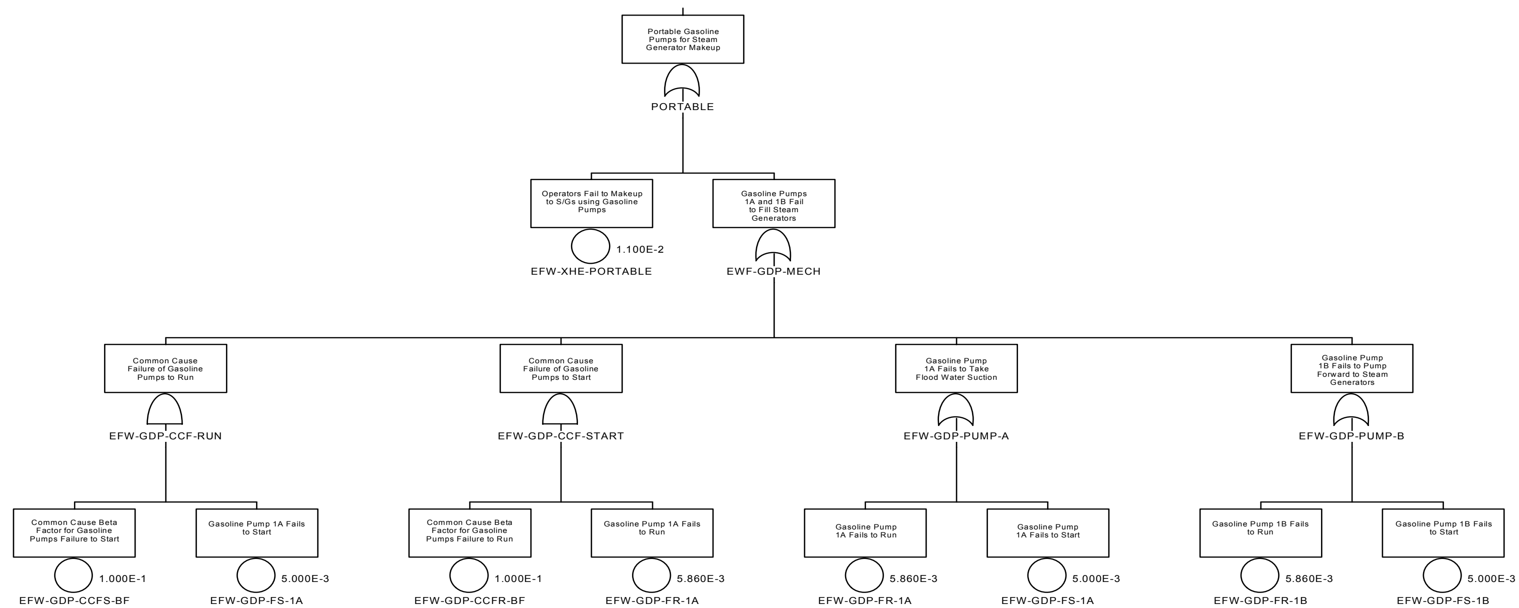

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Figure 2