UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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January 9, 2004

R. T. Ridenoure
Division Manager - Nuclear Operations
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SUBJECT: FORT CALHOUN STATION - NRC INSPECTION REPORT 05000285/2003-011

Dear Mr. Ridenoure:

On November 21, 2003, the US Nuclear Regulatory Commission (NRC) completed the onsite portion of an inspection at your Fort Calhoun Station. In-office inspection was continued through December 12, 2003, to review issues associated with the potential impacts of a long-term loss of instrument air. The enclosed report documents the inspection findings, which were discussed on November 21, 2003, with you and members of your staff.

This inspection was an examination of 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified two findings of very low safety significance (Green). This report also documents one finding concerning the ability to operate with a prolonged loss-of-instrument air. This finding has potential safety significance greater than very low significance. The latter finding does not present an immediate safety concern because the procedure was changed to address the issue appropriately during the inspection. The NRC has also determined that violations are associated with each of these findings. Two violations are being treated as a noncited violations, consistent with Section VI.A of the Enforcement Policy. The other violation will be dispositioned once a significance determination has been completed. The noncited violations are described in the subject inspection report. If you contest the violations or significance of the noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial(s), to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

Charles S. Marschall, Chief
Engineering and Maintenance Branch
Division of Reactor Safety

Docket No. 50-285
License No. DPR-40

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-285
License:	DPR-40
Report No.:	05000285/2003-011
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	Fort Calhoun Station FC-2-4 Adm. P.O. Box 399, Hwy. 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Dates:	November 3 through December 12, 2003
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Accompanying Personnel:	J. Adams, Reactor Inspector, Engineering and Maintenance Branch
Approved By:	Charles S. Marschall, Chief Engineering and Maintenance Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000285/2003-11; 11/03 - 12/12/2003; Fort Calhoun Station; Evaluation of Changes, Tests, or Experiments, and Safety System Design and Performance Capability

The NRC conducted an inspection with six regional inspectors. The inspection identified two Green noncited violations. The significance of most findings is indicated by its color (Green, White, Yellow, 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 3, dated July 2000.

<u>NRC-Identified and Self Revealing Findings</u>

Cornerstone: Barrier Integrity

- <u>TBD</u>. The team identified a violation for an inadequate procedure having potential safety significance greater than very low significance. Abnormal Operating Procedure AOP-17, "Loss of Instrument Air," Revision 5, a procedure required by Technical Specification 5.8 and Regulatory Guide 1.33, did not provide sufficient information for operators to respond to a prolonged loss of instrument air. Select valves were equipped with air accumulators or backup nitrogen supplies to maintain the valves operable after a loss of instrument air. The team identified some specific scenarios where valves would reposition to their failed position once the accumulators were exhausted and cause undesirable flow diversions in emergency core cooling systems.

  This finding was unresolved pending completion of a Phase 3 significance determination because the conditions of concern could not be evaluated using a Phase 2 assessment. This finding was more than minor because it affected objectives of both the mitigating systems and a barrier integrity cornerstones. Specifically, failure to maintain valves in their required positions following a loss of instrument air had the potential to impact the containment barrier function and mitigating system equipment availability. (Section 1R21.2)

Cornerstone: Mitigating Systems

- <u>Green</u>. A noncited violation of Technical Specification 5.8, “Procedures,” and Criterion III of Appendix B to 10 CFR Part 50 was identified for inadequate procedures. Abnormal Procedure AOP-17 requires operators to monitor select backup nitrogen supply bottle pressures with the intent to replace the bottles as necessary to maintain the pressure supply to the air operated valves. The valves affected were containment spray header isolation valves and the safety injection and refueling water tank outlet valves. The supply of spare nitrogen bottles was not procedurally controlled and was found to be insufficient to implement the procedure. This issue was entered into the licensee’s corrective action program under Condition Report 200305298.

This finding was more than minor because the barrier integrity cornerstone objective of maintaining the containment as a physical barrier to the release of radionuclides was affected by the procedure quality attribute. Specifically, the lack of spare nitrogen bottles had the potential to affect the leakage out of containment via the emergency core cooling system after a loss of instrument air. The finding screened as being of very low safety significance because it represented only a potential degradation of the radiological barrier function. (Section 1R21.2)

- Green. The team identified a finding of very low safety significance involving a noncited violation of Criterion III of Appendix B to 10 CFR Part 50 for the failure to correctly translate design information into calculations. Containment Piping Penetrations M-9 and M-12 contained steam generator drain lines with valves that were normally locked closed prior to plant startup, trapping cold water. The licensee did not consider the possible substantial pressure increase when the associated steam generators reach normal operating conditions in two calculations that assessed containment piping penetrations for potential over pressurization, EA-FC-90-082 and FC05994. The licensee concluded that the installed valves would allow enough seat leakage to prevent over-pressurizing the penetration, but this small leakage capability constitutes a design feature which is required to be documented and maintained. The licensee determined that the two calculations need to be revised. This finding was entered into the licensee's corrective action program under Condition Report 200305161.

  This finding affected the containment barrier cornerstone because of the potential for the loss of integrity of piping penetrating the containment vessel. This finding was more than minor because it was similar to Example 2.f of Appendix E of Manual Chapter 0612, in that the engineering staff had to perform a reanalysis and an operability evaluation due to this condition. This issue had very low safety significance because it did not represent an actual open pathway from the containment. (Section 1R21.5)

REPORT DETAILS

1 REACTOR SAFETY

Introduction

The NRC conducted an inspection to verify that the licensee adequately preserved the facility safety system design and performance capability and that the licensee preserved the initial design in subsequent modifications of the system selected for review. The scope of the review included any necessary nonsafety-related structures, systems, and components that provided functions to support safety functions. This inspection also reviewed the licensee's programs and methods for monitoring the capability of the selected systems to perform the current design basis functions. This inspection verified aspects of the initiating events, mitigating systems, and barrier cornerstones.

The licensee based the probabilistic risk assessment model for the Fort Calhoun Station on the capability of the as-built safety systems to perform their intended safety functions successfully. The inspectors determined the area and scope of the inspection by reviewing the licensee's probabilistic risk analysis models to identify the risk significant systems, structures, and components. The inspectors established this according to their ranking and potential contribution to dominant accident sequences and/or initiators. The inspectors also used a deterministic approach in the selection process by considering recent inspection history, recent problem area history, and all modifications developed and implemented.

The team reviewed in detail the components and subsystems that perform the containment functions, and safety-related portions of the instrument air system. The primary review prompted parallel review and examination of support systems, such as, electrical power, instrumentation, cooling water, and related structures and components. The instrument air system is a non-safety grade system which supports the operation of engineered safeguards equipment. Air-operated valves and dampers in engineered safeguards systems are to be designed to fail in their accident-required (safe) position or be equipped with safety-grade air-accumulators or nitrogen backup systems. The team reviewed the safety-grade portion of the instrument air system for a sample of air-operated valves and level instrumentation bubblers. The team reviewed design basis documents, abnormal and emergency operating procedures, system analyses, drawings, calculations, and testing procedures for a sample of valves and instrumentation equipped with air accumulators or nitrogen backup bottles.

The team assessed the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that the licensee used for the selected safety system and the necessary support systems during normal, abnormal, and accident conditions. Acceptance criteria used by the NRC inspectors included NRC regulations, the technical specifications, applicable sections of the Updated Safety Analysis Report, applicable industry codes and standards, and industry initiatives implemented by the licensee's programs.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

The team reviewed six licensee-performed safety evaluations to verify that licensee personnel had appropriately considered the conditions under which changes to the facility or procedures or the conduct of tests or experiments may be conducted without prior NRC approval. The subject evaluations had been performed since the last NRC inspection of these activities, which was documented in NRC Inspection Report 05000285/2002-03.

The team reviewed an additional 10 licensee-performed safety-evaluation screenings in which licensee personnel determined that evaluations were not required, to ensure that the exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59, "Evaluations of Changes, Tests, or Experiments." The team also reviewed five licensee-performed safety-evaluation applicability determinations in which licensee personnel determined that screenings, as allowed by the regulations, were not required.

The team reviewed and evaluated the most recent safety-evaluation program self-assessment and 10 corrective action documents written since the last NRC inspection of this area to determine whether there were sufficient in-depth analyses of the program to allow for the identification and subsequent resolution of problems or deficiencies.

A list of the specific documents reviewed is provided in the Attachment to this report. The required sample sizes for this biennial inspection are 5 to 7 licensee evaluations required by 10 CFR 50.59; and 10 to 15 changes, tests, or experiments that were screened out.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (71111.21)

.1 System Requirements

a. Inspection Scope

Inspection Procedure 71111.21 requires that one or two risk-significant systems be reviewed. The team inspected the following attributes of the containment and safety-related portions of the instrument air systems: (1) process medium (water, steam, and air), (2) energy sources, (3) control systems, and (4) equipment protection. The team examined the procedural instructions to verify instructions were consistent with actions required to meet, prevent, and/or mitigate design basis accidents. The team also considered requirements and commitments identified in the Updated Safety Analysis Report, technical specifications, design basis documents, and plant drawings.

b. Findings

Introduction: The team was unable to verify that the component cooling water system was able to provide the required flow to safety-related components, or that the containment air coolers were capable of removing their design heat loads. This is an unresolved item pending additional review of analyses and/or testing results by the NRC.

Description: The containment coolers are designed to remove heat from containment during accident conditions. They are specifically credited in the Fort Calhoun Updated Safety Analysis, Chapter 14, for limiting the peak containment pressure and temperature to within design limits during a main steam line break accident.

The team attempted to verify that the containment coolers were capable of removing the design heat load necessary to maintain containment pressures and temperatures within design limits by reviewing design and testing data. However, licensee engineers were unable to supply documentation that adequately demonstrated this capability to the team. Specifically, the licensee engineers could not show:

- The minimum component cooling water flow that was necessary for the coolers to remove the design heat load, or

- Test data, which demonstrated that the component cooling water system performance was verified to achieve the necessary flows to each of the vital loads under accident conditions

In addition, the team determined that some nonsafety-related loads cooled by component cooling water were intended to be isolated during accidents to allow more flow to vital loads. However, the isolation valves for some of these non-vital loads were fail-open valves, operated by instrument air without backup accumulators. Since the instrument air system is neither safety-related nor seismically qualified, it cannot be considered to be available under accident conditions. The team was concerned that the potential reduction in component cooling water flow to vital loads caused by flow diversions to non-vital loads could prevent fulfilment of one or more safety functions. For example, the team noted that the spent fuel pool cooling system would divert about 1100 gpm of component cooling water flow from vital loads on a loss of instrument air pressure. This represented a substantial portion of the total system flow, where one or two pumps would each be expected to supply 4,300 gpm flow to the system.

The licensee engineers provided information on flow modeling of the component cooling water system. The team reviewed the following calculations related to component cooling water flow rates:

Calculation FC 05669, "Component cooling water Flow Rates," Revision 3

Calculation FC 05660, "Check of Component cooling water Flow Model Against Measured Data," Revision 0

Calculation FC 06723, "Component cooling water and RW Input Data for ABB-CE Containment Analysis," Revision 0

Calculation FC 05780, "Component cooling water System Maximum DBA Heat Removal Rate," Revision 1

Due to time constraints and lack of documentation, the team was not able to verify the validity of the results of this modeling, nor whether they demonstrated adequate flow to attain the required heat removal in vital loads. However, the team was concerned that the model results were never validated by direct comparison to actual system performance while the system was aligned as it would be during the limiting accident.

In fact, the team noted that the component cooling water system had never been tested in an accident alignment to demonstrate that the required system flow rates to vital loads could be achieved. The team reviewed the following construction-era tests and noted that the system alignments did not reflect accident alignments:

- POTP.22, "Fort Calhoun Unit No. 1 Pre-operational Test Procedure Component Cooling Water System," dated April 10, 1972

- OT-PP-22, "Fort Calhoun Station Unit No. 1 Post-Core Hot Functional Test Procedure," dated July 6, 1973

On December 4 and 5, 2003, conference calls were held between licensee representatives and Region IV to discuss why the licensee concluded that the system would perform its intended function in light of the fact that it had never been tested to demonstrate that capability. The basis for reaching that conclusion was documented in Condition Report 200305471. The team reviewed the issue and concluded that the licensee had a reasonable basis to conclude the system was operable, primarily because the system was designed to have a large excess capacity. However, the standard for an operability evaluation is that the licensee must have a reasonable expectation that the system will perform as intended when a question about that capability exists. The licensee is required by various regulations, most notably 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” to establish design control measures, including tests, which demonstrate that design requirements have been correctly translated into actual plant performance capability. The team concluded that additional inspection is necessary to determine whether this requirement has been satisfied and documented in the design of the component cooling water system.

The licensee initiated Condition Report 200305293, which acknowledged that there were no flow tests that would verify the component cooling water flow rates developed in calculations.

<u>Analysis</u>: The potential risk associated with this issue will be evaluated upon completion of additional inspection. This item is unresolved pending receipt of additional information from the licensee concerning the results of their review of the maximum required flow rate and the heat removal requirements to meet design basis accident requirements. No immediate safety concern existed, as discussed above, because the system had a large excess capacity.

Enforcement: Appendix B to 10 CFR Part 50, Criterion III, "Design Control," states that measures shall be established to assure the design basis as defined in paragraph 50.2 and as specified in the license application, for those systems and components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Additional information from the licensee was needed in order to determine whether the component cooling water system was adequately designed and verified to be capable of supplying adequate flow to vital loads during accident conditions, and to determine whether the containment coolers were capable of removing the required heat from containment during accident conditions. This issue will be tracked as an unresolved item (URI 05000285/2003011-01) pending additional inspection of this issue.

.2 System Condition and Capability

a. Inspection Scope

The team reviewed testing procedures for the selected systems to verify that the capabilities of the systems were verified periodically. The team also reviewed the systems' operations by conducting system walkdowns; reviewing normal, abnormal, and emergency operating procedures; and reviewing the Updated Final Safety Analysis Report, technical specifications, design calculations, drawings, and procedures.

b. Findings

b1. Inadequate Nitrogen Backup Supplies

Introduction

The team identified a finding of very low safety significance involving a noncited violation of Technical Specification 5.8, "Procedures," and Criterion III of Appendix B to 10 CFR Part 50. Specifically, the team identified that the procedure for loss of instrument air included the requirement to maintain backup nitrogen supplies for select valves without sufficient procedural guidance to assure the required replacement nitrogen bottles would be available.

Description

Abnormal Procedure AOP-17, "Loss of Instrument Air," Revision 5, requires that the pressure of the backup nitrogen bottles for four valves equipped with safety grade nitrogen backup systems be checked every 2 hours. These valves are LCV-383-1 and LCV-383-2 (safety injection and refueling water tank (SIRWT) outlet valves); and HCV-344 and HCV-345 (containment spray header isolation valves). All of these valves fail open on loss of air pressure, and the backup nitrogen supplies are provided to hold the valves closed.

Procedure AOP-17 did not contain steps to replace the backup nitrogen bottles. Technical Basis Document TBD-AOP-17, clearly stated that the intent was to maintain these valves functional by checking and replacing the nitrogen bottles as necessary. The Updated Safety Analysis Report (USAR), Section 6.3.4.2, explains that one of the containment spray header isolation valves is required to stay shut to prevent pump runout if less than the full three containment spray pumps are running.

The team concluded that the SIRWT outlet valves were required to stay shut to prevent the loss of emergency core cooling system water from containment back to the tank.

Therefore, the team concluded that performance of Abnormal Procedure AOP-17, Loss of Instrument Air, for a prolonged period would require that replacement nitrogen bottles be available. The team identified that licensee personnel did not establish a dedicated reserve of spare nitrogen bottles for this purpose. The procurement program allowed the general inventory to be as low as three nitrogen bottles before reordering. Therefore, a sufficient supply of spare nitrogen bottles to allow even one round of replacements per Abnormal Procedure AOP-17 was not assured of being available.

Analysis

The performance deficiency associated with this finding is that the procedure for responding to a loss of instrument air contained the intent to maintain some valves functional for a mission time that exceeded the capacity of the installed backup nitrogen supply. The procedure was inadequate, in that it did not contain explicit instructions to replace the nitrogen bottles prior to losing the function, and the station did not assure that spare bottles would actually be available to perform this action.

This issue was more than minor because the ability of nitrogen backup supplies to maintain valves in their required positions affected the procedure quality attribute of the barrier integrity cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Failure to maintain Valves LCV-383-1 and LCV-383-2 in their required positions has the potential to impact the containment barrier integrity by allowing increased leakage out of containment via the emergency core cooling systems during the containment sump recirculation mode of operation.

This issue was determined to have very low safety significance. The finding screened as being of very low safety significance because it represented only a potential degradation of the radiological barrier function

The two systems affected by this finding were mitigating systems. However, the team concluded that the impact on the mitigating system cornerstone was minor. For the containment spray valves, emergency operating procedures provided guidance to verify that pumps were not in a runout condition. For the SIRWT suction valves, check valves provided protection against gross loss of water from containment, so the mitigating system function would not be challenged. For both, time was available for the condition to be recognized and the bottles replaced.

Enforcement

Technical Specification 5.8.1 and Regulatory Guide 1.33, Revision 2, Appendix A, require written procedures for combating emergencies and other significant events, including loss of instrument air. Abnormal Operating Procedure 17 implements this requirement at Fort Calhoun Station.

Criterion III of Appendix B to 10 CFR Part 50 states, in part, that "[m]easures 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 to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

Contrary to the above, the measures established to assure that design information was correctly translated into procedures were inadequate, in that, the requirement to maintain backup nitrogen supplies to select valves on loss of instrument air was identified but a method of assuring the necessary spare nitrogen bottles to implement the requirement was not incorporated into a procedure. The engineering staff issued Condition Report 200305298 and entered this finding into the corrective action program.

Because of the very low safety significance of the finding, and because the finding has been entered into the corrective action program, the team treated this as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000285/2003011-002)

b.2 Inadequate Abnormal Operating Procedure

a. Introduction

The team identified a violation with undetermined significance of Technical Specification 5.8 and Criterion III of Appendix B to 10 CFR Part 50. Specifically, the team identified that the abnormal operating procedure for loss of instrument air did not contain sufficient information for addressing a loss of instrument air over an extended period of time.

Description

Abnormal Procedure AOP-17 provides guidance for attempting to restore instrument air and the alignment of air-operated valves if system pressure cannot be restored to normal. The Fort Calhoun Station USAR accident analysis assumes that the nonsafety grade portions of the instrument air system will not be available. Valves and instruments, which will require air pressure for continued operation are provided safety grade air accumulators or backup nitrogen supplies. Abnormal Procedure AOP-17, Attachment E, “Air Operated Valves Operable Following Loss of Instrument Air,” lists valves with backup air accumulators or nitrogen supplies, the length of time the valves will remain operable, and the failure position of the valves on loss of pressure.

The Appendix E failure positions of the valves on loss of pressure are considered to be the fail-safe positions of the valves. The establishment of the fail-safe position of a valve was determined by the accident mitigating function of its system. The desired valve position is dependent on the assumed plant conditions being addressed. The long-term desired position of a valve after its backup air or nitrogen supply is exhausted may not be the same as its fail-safe position for initially responding to a design basis accident. Further, the analyses and calculations establishing the air accumulator or backup nitrogen operability times for air-operated valves do not fully address the required post-accident operating times for their respective systems.

For example, Valves HCV-385 and HCV-386 isolate the safety injection and containment spray pump recirculation to the SIRWT. Theses valves are normally open and remain open during the emergency core cooling system injection mode for a loss-of-coolant accident. Upon transfer to the containment sump recirculation mode, these valves are closed. These valves are equipped with air accumulators and are operable for 13 hours after the loss-of-instrument air and fail open on loss-of-air pressure.

Reopening Valves HCV-385 and HCV-386 during the containment sump recirculation mode would divert emergency core cooling system inventory from the containment building sump to the SIRWT. The reduction in containment sump inventory available to the safety injection and containment spray pumps would impact long-term emergency core cooling system operation due to insufficient net positive suction head for continued pump operation. The increased leakage from the containment to the SIRWT would be greater than the limit used in the analysis of USAR Section 14.15.8, "Radiological Consequences of a LOCA," therefore, the resulting dose assessment would also be increased.

Abnormal Procedure AOP-17 did not include any information about the expected effects of the valves listed in Appendix E going to their failure position after the backup air or nitrogen supplies are exhausted, or actions neccessary to address expected undersirable effects.

Analysis

The performance deficiency associated with this finding is the issuance of a procedure, which does not adequately address the long-term effects of a loss-of-instrument air. The team considered this finding to be more than minor because both the mitigating systems cornerstone and the barrier integrity cornerstone objectives were affected by the procedure quality attribute.

The ability of air accumulators and backup nitrogen supplies to maintain valves in their required positions affects the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The ability of air accumulators and backup nitrogen supplies to maintain valves in their required positions affects the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radio nuclide releases caused by accidents or events.

The finding of inadequate procedures is associated with the procedure quality attribute of both the mitigating systems and barrier integrity cornerstones. The attribute is applicable to operating (post-event) procedures (e.g., abnormal operating procedures and emergency operating procedures), which includes Abnormal Procedure AOP-17.

Failure of air accumulators and backup nitrogen supplies to maintain valves in their required positions following a loss of instrument air has the potential to impact the availability and reliability of multiple plant systems. The analyses for loss of instrument air do not demonstrate the valves will remain in their required positions throughout the periods their respective systems are required to operate. Further, the procedures do not address potential system operability problems and required operator actions beyond the periods that the air accumulators and backup nitrogen supplies will maintain pressure.

This issue was not an immediate safety concern because the licensee took compensatory measures to ensure the affected valves would be manually positioned prior to their accumulators being depleted by making a procedure change.

The team found that this issue resulted from a performance deficiency with the potential to be greater than very low safety significance (Green) because of the potential loss of accident mitigation equipment associated with this finding.

<u>Enforcement</u>

Technical Specification 5.8.1 states, in part, that "[w]ritten procedures and administrative policies shall be established, implemented and maintained covering the following activities: a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978; . . ." Regulatory Guide 1.33, Appendix A, lists typical safety-related activities that should be covered by written procedures. Appendix A, Section 6, “Procedures for Combating Emergencies and Other Significant Events,” includes loss-of-instrument air.

Criterion III of Appendix B to 10 CFR Part 50 states, in part, that "[m]easures 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 to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

Contrary to the above, the measures established to assure that design information was correctly translated into procedures were inadequate, in that, the requirement to maintain air operated valves in their desired positions on loss of instrument air was not addressed beyond the initial system responses when backup air accumulators and nitrogen supplies are available. The engineering staff issued Condition Report 200305311 and entered this finding into the corrective action program.

Because of the final safety significance of the finding has not yet been determined, and because the finding has been entered into the corrective action program, the team treated this as an unresolved item. (URI 05000285/2003011-003)

.3 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of problems for the selected systems identified by the licensee in the corrective action program to evaluate the effectiveness of corrective actions related to design issues. The sample included open and closed condition reports for the past 3 years and are listed in the attachment to this report. Inspection Procedure 71152, "Identification and Resolution of Problems," was used as guidance to perform this part of the inspection. Older condition reports that were identified while performing other areas of the inspection were also reviewed.

b. Issues and Findings

No findings of significance were identified.

.4 System Walkdowns

a. Inspection Scope

The team performed walkdowns of the accessible portions of the selected systems and required support systems. Inspectors focused on the installation and configuration of switchgear, motor control centers, manual transfer switches, field cabling, raceways, piping, components, and instruments. During the walkdowns, the team assessed:

- The placement of protective barriers and systems;
- The susceptibility to flooding, fire, or environmental conditions;
- The physical separation of trains and the provisions for seismic concerns;
- The accessibility and lighting for any required local operator action;
- The materiel condition and preservation of systems and equipment; and
- The conformance of the currently-installed system configurations to the design and licensing bases.

b. Findings

No findings of significance were identified.

.5 Design Review

a. Inspection Scope

The team reviewed the current as-built instrument and control, electrical, and mechanical design of the selected systems. These reviews included an examination of design assumptions, calculations, required system thermal-hydraulic performance, electrical power system performance, protective relaying, control logic, and instrument setpoints and uncertainties. The team also performed selected single-failure evaluations of individual components and circuits to determine the effects of such failures on the capability of the system to perform its design safety functions. The team also reviewed the licensee's calculations and methodology for ensuring the component cooling water system was protected against seismic, flooding, fire, and high energy line break events.

The team reviewed calculations, drawings, specifications, vendor documents, Updated Final Safety Analysis Report, technical specifications, emergency operating procedures, and temporary and permanent modifications.

b. Findings

Introduction

The team identified a finding of very low safety significance involving a non-cited violation of Criterion III of Appendix B to 10 CFR Part 50 for the failure to correctly translate design information into calculations to prevent potential over-pressurization of isolated containment penetrations.

Description

Fort Calhoun Station engineers performed Calculations EA-FC-90-082, "Potential Over-pressurization of Containment Penetration Piping Following a Main Steam Line Break in Containment," Revision 3; and FC05994, "Over-pressure Analysis of Penetrations M-7, M-9, & M-12 Under MSLB & LOCA Conditions," Revision 1, in response to Generic Letter 96-06, "Assurance of Equipment Operability And Containment Integrity During Design-basis Accident Conditions," Supplement 1. The team found that the calculations did not consider the effects of normal plant operation on the assumed starting pressures for Penetrations M-9 and M-12.

The subject piping in Penetration M-9 is a steam generator drain located between locked closed isolation Valves FW-181 and FW-686. For Penetration M-12, the steam generator drain piping is between locked closed isolation Valves FW-179 and FW-687.

These penetrations were evaluated in Calculations EA-FC-90-082 and FC-05994. The team found that those calculations used a starting pressure that was non-conservative compared to what could be expected under normal operating conditions. After the team identified this issue, a design engineer initiated Condition Report 200305161. The engineer determined that the two calculations needed to be revised. The engineer also performed an operability assessment for the current condition and concluded that the penetrations were operable. The team reviewed the licensee's analysis, compared it to vendor-supplied valve performance data, and concluded that the existing design would prevent over pressurization of the pipe.

Analysis

The team found the containment barrier cornerstone was affected because of the potential of for the loss-of-integrity of piping penetrating the containment vessel. The issue affected the attributes of design control and human performance. The team considered this finding more than minor since the finding was similar to Example 2.f of Appendix E of Manual Chapter 0612, "Power Reactor Inspection Reports," June 20, 2003, in that, the engineering staff had to perform a re-analysis and an operability evaluation due to this condition.

The team found that this issue resulted from a performance deficiency of very low safety significance. The team assessed this finding as Green because did not represent an actual open pathway from the containment. The finding was also found to be of very low safety significance because there was no actual loss of the containment barrier. Fort Calhoun Station personnel implemented corrective actions to ensure continued operability.

Enforcement

Criterion III of Appendix B to 10 CFR Part 50, "Design Control," states, in part, that "[m]easures 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 to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. . . . The design control measures shall provide for verifying or checking the adequacy of design, such as, by the performance of design reviews, [or] by the use of alternate or simplified calculational methods . . ."

Contrary to the above, the measures established to perform such verification were inadequate, in that, engineering personnel failed to correctly translate the normal plant values into calculations to determine the effects of an accident condition inside containment on the internal pressures of two piping penetrations. An engineer initiated Condition Report 200305161 and entered this finding into the corrective action program.

Because of the very low safety significance of the finding, and because the finding has been entered into the corrective action program, the team treated this as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000285/2003011-004)

.6 Safety System Inspection and Testing

a. Inspection Scope

The team reviewed the program and procedures for testing and inspecting selected components in the component cooling water system. The review included the results of surveillance tests required by the technical specifications and selective review of Class 1E control circuits for capability to test system functions.

b. Findings

No findings of significance were identified.

.7 Administrative Control of Containment Integrity

a. Inspection Scope:

The team inspected the licensee’s administrative practices for controlling containment integrity during nominal reactor conditions. They reviewed USAR (Section 5.9, “Structures, Containment Penetrations”) and technical specification (Technical Specification 2.6 “Containment System”) requirements governing the establishment/surveillance of containment integrity and implementing operating instructions (Operating Instruction OI-CO-5, “Containment Integrity,” Revision 21; and Operating Instruction OI-CO-1 “Containment Closeout,” Revision 25). The team compared containment penetration figures in the USAR with the checklists included in Operating Instruction OI-CO-5 to determine whether the checklists adequately ensure containment integrity following an outage. They also performed walkdowns to verify that the USAR figures accurately reflect as-built conditions and to inspect the materiel condition of the penetrations. They compared current containment isolation valve lineups with the recorded lineups on the checklists and reconciled differences.

b. Findings:

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On November 21, 2003, the inspectors presented the inspection results to Mr. R. Ridenoure and other members of his staff, who acknowledged the findings. A followup telephone call was made on December 12, 2003, with Mr. G. Cavenaugh to status the inspection findings. The inspectors confirmed that proprietary information which was examined during this inspection was returned to the licensee.

ATTACHMENT

KEY POINTS OF CONTACT

Licensee personnel

D. Bannister, Plant Manager
G. Cavanaugh, Supervisor, Station Licensing
R. Clemens, Division Manager, Nuclear Assessments
M. Core, Manager, Systems Engineering
A. Hackerot, Supervisor, Systems Analysis
K. Hyde, Supervisor, Design Engineering
A. Koenig, System Engineer
R. Lentz, Senior Licensing Engineer, 10 CFR 50.59 Coordinator
S. Lindquist, Operations
R. Luikens, Operations
C. Linden, Air-Operated Valve Program Coordinator
J. Nejad, Corporate Health Physics
R. Phelps, Division Manager, Nuclear Engineering
R. Ridenoure, Division Manager, Nuclear Operations
G. Seier, Procurement Engineering
D. Taylor, Design Engineer Mechanical

NRC Personnel

J. Kramer, Senior Resident Inspector
L. Willoughby, Resident Inspector

LIST OF ITEMS OPENED AND CLOSED

Opened

05000285/2003011001	URI	Verify Design Flow and Heat Removal Capability of Component Cooling Water (Section 1R21.1)
05000285/2003011003	URI	Failure to Provide Means to Assure Proper Emergency Core Cooling System Alignment During Prolonged Loss of Instrument Air (Section 1R21.2).

Opened and Closed

05000285/2003011002	NCV	Inadequate Procedure for Long-term Loss of Instrument Air (Section 1R21.2).
05000285/2003011004	NCV	Failure to Translate Design Basis for Preventing Containment Penetration Over-pressurization into Design Basis (Section 1R21.5).

LIST OF DOCUMENTS REVIEWED

Calculations

NUMBER	TITLE	REVISION
	Generic Air Accumulator's Using Propane Tanks Built to DOT Spec 4BA-240	0
EA-FC-90-082	Potential Overpressurization of Containment Penetration Piping Following a Main Steam Line Break in Containment	3
EA-FC-93-022	MFIV Stroke Time Evaluation/Containment Response	0
EA-FC-95-012	Effect of Post-DBA CCW Temperature Transient on Components	0
EA-FC-98-034	Electrical Penetration Design Basis Verification for Continuous Load Current, Overload Protection and Short Circuit Protection ELM-2.1-L1SC1	0
EA-FC-02-002	Containment Response to MSLB with Gothic	0
FC-88-017	Addition of a Third AFW Pump	May 4, 1990
FC01274	Minimum Accumulator Pressure Required	0
FC01438	Air Accumulator Capacity For IA-41, 42, 43, and 44	1
FC02007	Accumulator Sizing	2
FC04074	N2 Cylinder Sizing for HCV-438B/D, LCV-383-1/2, HCV-344/345 to Provide 1000 Hr Valve Actuation Supply	C
FC05346	Accumulator Sizing for Auxiliary Feedwater Control Valves HCV-1107A&B, HCV-1108A&B, HCV-1105, HCV-1106, FCV-1368, FCV-1369	1
FC05375	Pressure Between Aux. Feedwater Penetration Valves After a Main Steam Line Break	0

Calculations

NUMBER	TITLE	REVISION
FC05525	Containment Penetration Overpressure Analysis for M-7, M-8, M-11, M-15, M-18, M-19, M-53, M-80 Post-LOCA or MSLB	2
FC05660	Check of CCW Flow Model Against Measured Data	0
FC05669	CCW Flow Rates	3
FC05691	Air Accumulator Operable Time Requirements	2
FC05693	Component Cooling Water System Design Heat Loads and Flows	0
FC05780	CCW System Maximum Design Basis Heat Removal Rate	1
FC05977	ABB-CE Evaluation of Containment Spray Pump Net Positive Suction Head Accounting for Sump Subcooling	1
FC05994	Overpressure Analysis of Penetrations M-7, M-9, & M-12 Under MSLB & LOCA Conditions	1
FC06040	N2 Cylinder Sizing for HCV-400 Series Valves	0
FC06209	Containment Air Cooling Coils Post-Accident Heat Removal Performance	0
FC06676	Post-RAS NPSH Adjustments for CS and HPSI Pumps	0
FC06708	YCV-1045 Maximum Allowable Stroke Time Evaluation	5
FC06723	CCW and RW Input Data for ABB-CE Containment Analysis	0
FC06731	Containment Basement Water Level	1
FC06734	Injection Phase NPSH Adjustments for CS and HPSI Pumps	0
FC06742	Accumulator Sizing and Seismic Support for HCV-298 Accumulator	0

Calculations

NUMBER	TITLE	REVISION
FC06747	SI Pump Room (Room 21 & 22) Heat-up During Pump Operation Computer Analysis: GOT-5.0(QA)C-PI	1
FC06927	Analysis of Liner Panel of Fort Calhoun (Unit 1) Containment Vessel	1
O-MPS-CALC-008	Containment Spray Flow Rates in Various System Configurations	0

Condition Reports (CRs)

199701715	200102108	200201836	200303701	200305161	200305284
199901893	200102462	200201983	200303771	200305224	200305291
200100470	200102723	200203460	200304395	200305245	200305293
200100494	200103362	200204397	200304448	200305248	200305298
200101368	200200613	200204398	200304535	200305251	200305311
200101484	200200716	200300377	200304557	200305261	200304957
200101509	200200973	200300977	200305054	200305267	200305275
200101675	200201396	200301548	200305056	200305268	200305516
200101820	200201494	200302523	200305088	200305271	200305674
200101845	200201531				

Design Basis Documents

NUMBER	TITLE	REVISION
PLDBD-IC-30	Instrumentation Installation	6
PLDBD-IC-32	Instrumentation and Control Systems	18
SDBD-CA-IA-105	Instrument Air	16
SDBD-AC-CCW-100	Component Cooling Water	29
SDBD-AC-RW-101	Raw Water	24
SDBD-DG-112	Emergency Diesel Generators	18
SDBD-FW-AFW-117	Auxiliary Feedwater	26

Design Basis Documents

NUMBER	TITLE	REVISION
SDBD-MS-125	Main Steam and Turbine Steam Extraction	17
SDBD-SI-CS-131	Containment Spray	16
SDBD-SI-132	High Pressure Safety Injection	12
SDBD-SI-LP-133	Low Pressure Safety Injection System	15
SDBD-VA-CON-139	Containment HVAC	16
SDBD-VA-SI-CS-131	Containment Spray	16

Drawings

NUMBER	TITLE	REVISION
177B2371, Sh 316A	MCC 3B1 Data Sheet	17
11405-E-180, Sh 24	Electrical Penetration C-6 Wiring Diagram	1
11405-E-180, Sh 25	Electrical Penetration C-6 Wiring Diagram	5
11405-M-1, Sh 2	Containment Heating, Cooling & Ventilating Flow Diagram P&ID	27
11405-M-1, Sh 1	Containment Heating, Cooling & Ventilating Flow Diagram	73
11405-M-40, Sh 1	Auxiliary Coolant Component Cooling System Flow Diagram P&ID	36
11405-M-42, Sh 1	Nitrogen, Hydrogen, Methane, Propane and Oxygen Gas Flow Diagram P&ID	87
11405-M-263	Composite Flow Diagram Compressed Air P&ID	26
11405-M-263, Sh 1	Flow Diagram Compressed Air P&ID	64

Drawings

NUMBER	TITLE	REVISION
11405-M-263, Sh 2	Flow Diagram Compressed Air P&ID	18
11405-M-264	Composite Flow Diagram Instrument Air Diagram	26
11405-M-264, Sh 1	Instrument Air Diagram Auxiliary Building & Containment P&ID	60
11405-M-264, Sh 2	Instrument Air Diagram Turbine Building & Intake Structure P&ID	42
11405-M-264, Sh 3	Instrument Air Diagram Riser Details P&ID	46
11405-M-264, Sh 4	Instrument Air Diagram Riser Details P&ID	42
11405-M-264, Sh 5	Instrument Air Diagram Riser Details P&ID	42
161F615	SIRW Tank LO Level 2/4 Matrix - Initiation Matrix A & B Schematic	6
2325-7682, Sh 1	Medium Voltage Power Penetration Assy's	H
2325-7682, Sh 4	Medium Voltage Power Penetration Assy's	C
2325-7683	Coaxial and Triaxial Penetration Assemblies	F
2325-7684	Low Voltage Power, Control, Instrumentation, and Thermocouple Penetration Assemblies, Sheet 1 of 83	5
A-1388	Composite Model 12-W-204 Safety Valve	1
D-4062	Containment Hydrogen Purge System - Control Cable Isometric Layout Sheet 1 of 2	1
D-4069	Jack Screw Component Cooling Water Control Valves	0
D-4246	CQE Piping Isometrics Seismic Subsystem #VA-3026A	3
E-23866-210-130, Sh 1	Safety Injection and Containment Spray System	82

Drawings

NUMBER	TITLE	REVISION
E-23866-210-130, Sh 2	Safety Injection and Containment Spray System	61
EM-400, Sh 1	Instrument and Control Equipment List	20
EM-743/744, Sh 1	Instrument and Control Equipment List	5
EM-745, Sh 1	Instrument and Control Equipment List	6
EM-783/786, Sh 1	Instrument and Control Equipment List	8
EM-783/786, Sh 2	Instrument and Control Equipment List	8
GE 161F615	Containment High Press-2/4 Matrix Initiation Matrix - A & B Schematic	8
SPEC No. 11.30	Auxiliary Coolant System Control Valves	12

Miscellaneous Documents

NUMBER	TITLE/DESCRIPTION	REVISION
	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 52 to Facility Operating License No. DPR-40 Omaha Public Power District Fort Calhoun Station, Unit No. 1 Docket No. 50-285	October 14, 1980
	Safety Evaluation Related to Amendment No. 121 for Fort Calhoun Station	April 26, 1989
	Visual Examination of Data Form VT-3, IWE Containment Vent Systems and Surfaces	September 17, 2003
EA-FC-91-014	Effect of Loss of Cooling Water on SI/CS Pumps	1
LIC-79-0113	Evaluation of Fort Calhoun Safety Injection Pump Room Temperature Following a Loss of Coolant Accident	September 6, 1979

Miscellaneous Documents

NUMBER	TITLE/DESCRIPTION	REVISION
LIC-96-0164	Initial Response to NRC Generic Letter 96-06	October 30, 1996
LIC-96-0179	Licensee Event Report 96-012 Revision 0 for the Fort Calhoun Station	December 11, 1996
LIC-97-006	Response to Generic Letter 96-06 (TAC Number M96813)	January 24, 1997
LIC-98-0095	Response to Request for Additional Information Related to Generic Letter 96-06 Response (TAC No. M96813)	July 24, 1998
LIC-98-0132	Revised Response to Generic Letter 96-06 (TAC Number 96813)	October 21, 1998
LIC-99-0056	Additional Information to Support NRC Review of OPPD Response to Generic Letter 96-06 (TAC Number M96813)	June 22, 1999
LIC-02-0121	10 CFR 50.59 Report and Updated Safety Analysis Report (USAR) Revision for Fort Calhoun Station	November 15, 2002
LIC-03-0001	Fort Calhoun Station Unit No. 1 License Amendment Request, Containment Pressure Analysis using the Gothic Computer Code	January 27, 2003
LIC-03-0103	Response to Requests for Additional Information on License Amendment Request	August 1, 2003
NPM-220	Commercial Grade Item Evaluations	3
OSAR No. 87-10	GSE Request - Determine Which Valves with Air Accumulators are Required for Safe Shutdown	April 6, 1988
PDB-9	Relief Valve Program, Section 1.2.21, ‘VA-287/288'	9
PE Evaluation 18328	Valve, Check, Poppet, Body & Poppet 316SS/A479, Tube Conn ½ IN (In & Out), Working Pressure 3000 PSIG, Cracking Press 1 PSI	0

Miscellaneous Documents

NUMBER	TITLE/DESCRIPTION	REVISION
Standing Order SO-O-21	Shutdown Operations Protection Plan	7
Technical Specification 2.6	Containment System	Amendment No. 185
Topical Report AAF-TR-7101	Design and Testing or Fan Cooler Filter Systems for Nuclear Applications	February 20, 1972
Volume 19	Engineered Safeguard Controls System Training Manual	22

Procedures

NUMBER	TITLE	REVISION
AOP-11	Loss of Component Cooling Water	7
AOP-12	Loss of Containment Integrity	4
AOP-17	Loss of Instrument Air	5
ARP-AI-65A/A65A	Annunciator Response Procedure A65A Control Room Annunciator A65A Containment/RCGVS	11
ARP-CB-1,2,3/A1	Annuniciator Response Procedure	21
EOP-03	Loss of Coolant Accident	24
EOP-05	Uncontrolled Heat Extraction	18
EOP-20	Functional Recovery Procedure	10, 11
FCSG-23	Guideline-10 CFR 50.59 Resource Manual	2
IC-CP-01-0744	Calibration of Containment Pressure Loop P-744	1
IC-CP-01-0745	Calibration of Containment Pressure Loop P-745	1

Procedures

NUMBER	TITLE	REVISION
IC-CP-01-0786	Calibration of Containment Narrow Range Pressure Loop P-786	7
IC-PM-EX-1000	Verification of Electrical Penetration Nitrogen Seal Pressure	2
IC-ST-AFW-3002	Instrument Air Accumulator/Check Valve Operability Test	4
IC-ST-IA-3001	Safety Injection Refueling Water Tank Air Accumulator Check	6
IC-ST-IA-3002	CVCS Instrument Air Accumulator Check Valve Test	8
IC-ST-IA-3005	Instrument Air Accumulator Check Valve IA-HCV-2987-C Operability Test	12
IC-ST-IA-3009	Operability Test of IA-YCV-1045-C Instrument Air Accumulator	13
IC-ST-VA-0003	Channel Calibration of Containment Wide Range Pressure Loop P-783	3
IC-ST-VA-0013	Calibration of Containment Air Cooling and Filtering Units Flow and Pressure Drop	March 15, 2001
IP-PM-CCW-0350	Backup Nitrogen Supply Systems	3
IP-ST-SI-0002	Channel Calibration of SIRWT Low Level Monitoring Switches	7
NL-14	Instructions for Processing Proposed Changes to the Updated Safety Analysis Report and Inclusion Into the 50.59 Report	5
NOD-QP-3	10 CFR 50.59 Reviews	23
NOD-QP-16	Updated Safety Analysis Report (USAR)	17
OI-CC-1	Component Cooling System Normal Operation	46

Procedures

NUMBER	TITLE	REVISION
OI-CO-1	Containment Closeout	25
OI-CO-4	Containment Closure	42
OI-CO-5	Containment Integrity	21
OI-NG-1	Nitrogen System Normal Operation	19
OP-PM-CCW-0901	Preventive Maintenance Procedure CCW Inservice Flush of Containment Cooling Coils VA-1A/B and VA-8A/B	1
OP-PM-RW-0001	Raw Water System Interface Valve Actuation Test	3
OP-ST-AFW-3010	Auxiliary Feedwater System Category A and B Valve Exercise Test	3
OP-ST-CCW-3005A	Component Cooling Category A & B Valve Exercise Test	2
OP-ST-CCW-3005B	Component Cooling Category A & B Valve Exercise Test	2
OP-ST-ESF-0001	Diesel Auto Start Initiating Circuit Check	21
OP-ST-MS-3001	Main Steam System Category B and C Valve Exercise Test	18
OP-ST-SI-3002	Safety Injection System Category A, B and C Valve Exercise Test	16
OP-ST-VA-0008	Containment Ventilation System Containment Fans and Dampers Exercise Test	6
OT-PP-2	Post-Core Hot Functional Test Procedure	July 11, 1993
PE-ST-VX-3010	ASME Section XI Code Relief Valve Test for the Hydrogen Purge Ventilation System	3

Procedures

NUMBER	TITLE	REVISION
POTP-22	Pre-Operational Test Procedure Component Cooling Water System	April 27, 1972
SP-CP-08-D1-TC	Functional Checkout of the protective Relays in Diesel Generator Number One Circuit	11
SO-R-2	Condition Reporting and Corrective Action	24

Safety Evaluations

EC 26232 EC 26606 EC 28515 EC 33214 EC 33320 EC 33321

Safety Evaluation Screenings

EC 28131	EC 33197	EC 33401	EC 33571	EC 33602
EC 30288	EC 33282	EC 33417	EC 33589	EC 33626

Safety Evaluation Applicability Determinations

EC 29917 EC 30762 EC 32934 EC 33043 EC 33208

Surveillance Test Results

NUMBER	TITLE	REVISION
OP-ST-ESF-0009	Recirc. Actuation Signal Logic and Switch Test	October 6, 2003
OP-ST-ESF-0019	Channel A SI, CS and Recirc. Actuation Signal Test	October 22, 2003
PE-ST-VX-3010	ASME Section XI Code Relief Valve Test for the Hydrogen Purge Ventilation System, Valve VA-288	October 23, 1988

Surveillance Test Results

NUMBER	TITLE	REVISION
PE-ST-VX-3010	ASME Section XI Code Relief Valve Test for the Hydrogen Purge Ventilation System, Valve VA-287	November 4, 1988 and October 25, 2001

UFSAR

SECTION	DESCRIPTION	REVISION
5.9	Structures, Containment Penetrations	7
6.2	Engineered Safeguards - Safety Injection System	13
6.3	Engineered Safeguards -Containment Spray System	9
6.4	Engineered Safeguards -Containment Air Cooling and Filtering System	4
8.4	Electrical Systems - Diesel-Generators	9
9.4	Auxiliary Systems - Auxiliary Feedwater System	12
9.7	Auxiliary Systems - Component Cooling Water System	7
9.8	Auxiliary Systems - Raw Water System	13
9.12	Auxiliary Systems - Compressed Air System	8

Work Orders

00013736	00125219	00151217	00157817
00123775	00125303	00157699	00157900
00123812	00125394		