UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

May 29, 2007

Duke Power Company LLC
d/b/a Duke Energy Carolinas, LLC
ATTN: Mr. J. R. Morris
Site Vice President
Catawba Site
4800 Concord Road
York, SC 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION - COMPONENT DESIGN BASES INSPECTION - NRC INSPECTION REPORT 05000413/2007006 AND 05000414/2007006

Dear Mr. Morris:

On April 19, 2007, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Catawba Nuclear Station Units 1 and 2. The enclosed inspection report documents the inspection findings which were discussed on April 19, with you and other members 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 inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified three findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because each was entered into your corrective action program, the NRC is treating the findings as non-cited violations consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these non-cited violations you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Catawba Nuclear Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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 Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mike Cain, Acting Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos.: 50-413, 50-414
License Nos.: NPF-35, NPF-52

Enclosure: NRC Inspection Report 05000413/2007006 AND 05000414/2007006
w/Attachment: Supplemental Information

(cc w/encl cont'd — See page 3)

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Sincerely,

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Docket Nos.: 50-413, 50-414
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-413, 50-414
License Nos.:	NPF-35, NPF-52
Report Nos.:	05000413/2007006, 05000414/2007006
Licensee:	Duke Power Company, LLC
Facility:	Catawba Nuclear Station
Location:	4800 Concord Road York, SC 29745-9635
Dates:	March 19 - April 19, 2007
Inspectors:	R. Moore, Lead Inspector M. Yeminy, Contractor R. Lewis, Reactor Inspector W. Fowler, Reactor Inspector G. Skinner, Contractor E. Riggs, Resident Inspector J. Hamman, Inspector Trainee
Approved by:	M. Cain, Acting Chief, Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000413/2007006; 05000414/2007006; 03/19/2007 - 03/23/2007, 04/02/2007 - 04/-6/2007, 04/16/2007 - 04/19/2007; Catawba Nuclear Station, Units 1 and 2; Component Design Bases Inspection.

This inspection was conducted by a team of four NRC inspectors and two NRC contractors. Three green non-cited violations, were identified during this inspection. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP 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 and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for failure to perform adequate corrective action associated with an air entrainment issue in the auxiliary feed water system (CA) pump suction line identified in PIP C-97-01579. The corrective actions in PIP 97-01579 were inadequate in that they did not address the potential impact of the air entrainment on the swap over instrumentation for the assured water supply located in the suction line upstream of the pumps. The licensee entered this deficiency into their corrective action program.

  This finding is more than minor because the engineering calculation error which failed to include the potential impact of the air entrainment on the RN/CA swap over pressure switches resulted in a condition in which there was reasonable doubt on the operability of the CA pumps. The finding is of very low safety significance because the licensee's engineering evaluations performed during the inspection determined that there was no adverse impact on the pressure switches and therefore no loss of the CA pumps capability for short term heat removal. (Section 1R21.2.5)

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for failure to perform adequate and timely corrective actions to resolve a potential equipment design deficiency of the 1DGBB battery and distribution which provided the alternate power supply to the 125 VDC Vital I&C distribution panel 1EDF. The licensee entered this deficiency into their corrective action program.

  This finding is more than minor because it affects the mitigating systems cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events in that 125 VDC distribution center 1EDF provides control power to critical equipment such as the 4.16kV vital bus which aligns power to ECCS pumps and

valves. The finding is associated with the cornerstone attribute of design control. This finding is of very low safety significance because the team identified no occurrence, since this issue was identified on July 20, 2006, in which the station was aligned in the vulnerable condition relying on the alternate power supply to 1EDF. Additionally, the normal power supply, the vital battery, is a highly reliable power source and the alignment to the alternate power source requires manual action. Therefore there was no loss of the 1EDF safety function to provide adequate vital I&C control power for safe shutdown of the plant. This finding involved the crosscutting area of Problem Identification and Resolution because the evaluation, specifically the operability assessment, was inadequate and contributed the inadequacy of subsequent corrective actions. (Section 1R21.2.12)

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, for failure to follow procedure NSD 319, Vendor Technical Information Program, Rev. 2, which requires performance of technical impact reviews of maintenance and surveillance procedures due to vendor manual changes and technical updates. The licensee entered this deficiency into their corrective action program.

  This finding is more than minor because procedure inconsistencies were identified between the reactor trip breaker vendor manual and procedure SI/0/A/5100/002, Reactor Trip Breaker Surveillance Procedure, Rev. 18, which indicated that the licensee routinely failed to perform engineering evaluations on similar issues. The finding was determined to be of very low safety significance because there was no loss of the reactor trip breaker safety function to open on a scram signal. (Section 1R21.2.15)

B. Licensee-identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the licensee's Probabilistic Risk Assessment (PRA). In general, this included components and operator actions that had a risk achievement worth factor greater than two or Birnbaum value greater than 1 X10-6. The components selected were located within several safety related systems. The sample selection included 15 components, 5 operator actions, and 5 operating experience items. Additionally, the team reviewed three modifications by performing activities identified in IP 71111.17, Permanent Plant Modifications, Section 02.02.a. and IP 71111.02, Evaluations of Changes, Tests, or Experiments.

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 due to modification, or 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 action, repeated maintenance, maintenance rule (a)1 status, Regulatory Issue Summary 05-020 (formerly GL 91-18) 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. An overall summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

.2 Results of Detailed Reviews

.2.1 Air Operated Valve (AOV) 1RN-351 (KC Heat Exchanger 1B RN Control Valve)

a. Inspection Scope

The team reviewed the design basis documentation, including the system design basis document (DBD), supporting calculations, drawings, and the updated final safety analysis report (UFSAR) to identify the design bases function of AOV 1RN-351.

The team reviewed the design, operation, and routine maintenance of the valve assembly to assess the positioner component reliability to position the valves for decay heat removal and accident conditions. Additionally, the team reviewed the independence of the control power to assess the potential for common cause failure of the valves. The team reviewed AOV calculations which established the maximum expected differential pressure and surveillance procedures which implemented testing of this valve to assure that it can function under accident conditions. The failure mode of the valve positioner was reviewed to verify it was consistent with the design basis assumptions. Preventative maintenance and surveillance program documentation governing this component was evaluated against service history records to ensure that indications of degraded performance were being identified, evaluated and trended and that the valve was being operated and maintained in accordance with industry and manufacturer's recommendations. The team walked down the valve to observe material conditions.

b. Findings

No findings of significance were identified.

.2.2 Component Cooling Heat Exchangers (KC Hx)

a. Inspection Scope

The team reviewed the heat exchanger specification information, design basis information and supporting calculations to identify the heat removal requirements and capability of the KC heat exchangers to remove the required heat load. This included the tube plugging limits, basis for the limits and the number of tubes presently plugged. The maintenance, inspection, and performance testing were reviewed to verify the capability of the heat exchangers to remove the design heat load as well as the adequacy of flow testing for both the shell side and tube side of the heat exchangers. The calibration of instrumentation used for the KC Hx heat capacity testing was reviewed to verify that appropriately calibrated instrumentation was used for testing. The team reviewed the licensee's use of the fouling factor parameter to assess their capability to identify the heat exchanger's performance degradation, the correct use of the design temperature of the ultimate heat sink, and the projection of test results to accident parameters. In addition, the team verified that instrument uncertainty was taken into account and that sufficient margins exist. The team reviewed the trending of the performance of the heat exchangers as well as the frequency of the thermal testing and schedule for visual inspection and cleaning. The team reviewed the station's overall implementation of GL 89-13, Service Water System Problems Affecting Safety-Related Equipment, to verify that requirements applicable to the KC Hx were addressed.

b. Findings

No findings of significance were identified.

.2.3 ND-36B, NI-173A, -178B (Decay Heat Removal System (ND) suction and injection discharge to RCS)

a. Inspection Scope

The team reviewed MOV calculations to verify that design bases, system conditions, and allowable degraded voltage conditions were used as design inputs to size the actuators and establish set point values. Additionally, the translation of design information into MOV test procedure acceptance criteria was reviewed. Maintenance documentation was reviewed to verify that MOVs were periodically tested and that appropriate torque switch settings were maintained. Maintenance history and corrective action history were reviewed to assess the capability to identify component degradation. The team reviewed the minimum required and the maximum allowable thrust and torque as well as stall torque to assess the adequacy of motor sizing for the system application.

The inspectors reviewed elementary and single line diagrams for 1ND-36B (ND Pump B Suction Isolation Valve), 1NI-173A (ND Header A to NC Loops C and D Cold Legs Isolation Valve) and 1NI-178B (ND Header B to NC Loops A and B Cold Legs Isolation Valve) as well as design basis documentation for the component power supply and control considerations in order to ensure that accident conditions were adequately addressed through the design. Operating and abnormal operating procedures were reviewed to ensure that the condition and configuration of controls reflect and support the attributes identified in the design bases. The team reviewed the elementary and schematic diagrams of the valve motor control circuit configurations to verify that the circuitry satisfied the logic presented in the design basis documentation. Interlocks were reviewed to ensure that system architecture adequately supported the design bases under accident conditions. A walk down was performed to observe material conditions and verify valve alignment was consistent with plant operating conditions.

b. Findings

No findings of significance were identified.

.2.4 1RN-291 (KC Hx 1A RN Control Valve)

a. Inspection Scope

The team reviewed the design base documentation, including the DBD, supporting calculations, drawings, and the UFSAR to identify the design base function of AOV 1RN-291. Maintenance, modification, and corrective action history of the AOV was reviewed to verify that component degradation would be identified. The team reviewed the design, operation, and routine maintenance of the valve assembly to assess its reliability to position the valve for decay heat removal and accident conditions. Additionally, the team reviewed the independence of the control power to assess the potential for common cause failure of the KC Hx control valves. The team reviewed AOV calculations which established the maximum expected differential pressure and surveillance procedures which implemented testing of this valve to assure that it could

function under accident conditions. The failure mode of the valve positioner was reviewed to verify it was consistent with the design basis assumptions. Preventative maintenance and surveillance program documentation governing this component was evaluated against service history records to ensure that indications of degraded performance were being identified, evaluated and trended and that the valve was being operated and maintained in accordance with industry and manufacturer's recommendations. The team walked down the valve to observe material conditions.

b. Findings

No findings of significance were identified.

.2.5 Turbine Driven Auxiliary Feedwater Pump (CAPT)

a. Inspection Scope:

The team reviewed the design basis documentation, pump vendor manual and related vendor correspondence, drawings, and the UFSAR to identify design, maintenance, and operational requirements related to pump flow and developed head, achieved system flow, net positive suction head (NPSH), vortex formation and prevention, minimum flow requirements, and runout protection. These requirements were reviewed for pump operation with the source of water originating from the auxiliary feedwater condensate storage tank (CACST) and upper storage tank (UST). Design calculations as well as documentation of in-service, periodic surveillance tests, and flow balances were reviewed to verify that design performance requirements were met. The team also performed alternate calculations to assess the adequacy of calculations assessing the magnitude of air ingestion. Maintenance, in-service testing, corrective action, and design change histories were reviewed to assess the potential for component degradation and resulting impact on design margins and performance.

The team reviewed the adequacy and reliability of the CA suction source swap over function from the preferred (non-safety related) to the assured (safety related) water source. This included review of corrective actions in PIP C-097-01579 which identified a safety issue related to air entrainment from the CACST and its impact on the operation of the CA pumps. The team reviewed the design features of the service water/auxiliary feedwater (RN/CA) swap over pressure instrumentation which included a reset and five second time delay to assess the potential impact on this equipment from the CACST air entrainment into the CA pump suction piping. The team performed an alternate calculation to assess the margin available to avoid emptying the suction piping.

The team reviewed the elementary and schematic diagrams of the CAPT control circuit configurations to verify that the circuitry satisfied the logic presented in the design basis documentation. In addition, the team walked down portions of the CA system to verify that the installed configuration was consistent with design basis information and visually inspected the material condition of the pumps. The team reviewed the licensee's actions to verify the timed operator actions for abnormal operation to verify that the time to perform the actions was consistent with design basis assumptions.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for failure to perform adequate corrective action associated with an air entrainment issue in the CA pump suction line identified in PIP C-97-01579. The corrective actions in PIP C-97-01579 were inadequate in that they did not address the potential impact of the air entrainment on the swap over instrumentation for the assured water supply located in the suction line upstream of the pumps.

Description: During the review of the design of water sources to supply the turbine driven CA pump, the team identified a potential adverse impact of air entrainment from the CACST on the CA suction line pressure switch which provides the signal to align the RN to the CA pumps. The initial water supply to the safety related CA pumps is provided by a preferred, non-safety related source, and upon depletion or loss of the preferred source, the pump suction swaps over to the assured, safety related source. The preferred source is condensate grade water provided by the CACST containing approximately 42,500 gallons, an UST containing approximately 85,000 gallons, and the main condenser with approximately 170,000 gallons. The CACST and the UST lines combine at a common vertical header to form the CA suction line with the RN line entering the suction line below the intersection. To align the 170,000 gallons from the main condenser, manual operator actions are required. The assured source is provided by the RN and is initiated by pressure switches in the CA suction line. These pressure switches use a two-out-of-three logic to open valves that align the CA suction line with the RN system.

By design, the CACST empties completely before the UST water is available upon the opening of a check valve. Because the UST is subjected to the main condenser vacuum, and until the water level in the CACST is completely depleted and the water level in the suction line is further lowered, there isn't sufficient pressure to open the check valve. The piping and valves in the preferred source lines are non-safety related. As the CACST empties, air is drawn into the suction piping due to the formation of a vortex. The licensee initially reviewed the impact of this air entrainment on CA pump operation in 1997 via PIP 97-01579 and implemented modifications to reduce the vulnerability of the CA pumps. The licensee's initial evaluation of the air entrainment issue is documented in calculation CNC-1223.42-00-0054, Analysis of CA System Suction Sources, Rev. 1. The initial review of the air entrainment condition and the subsequent piping modification did not address the potential adverse impact of the air entrainment on the RN/CA swap over pressure switches. The team determined the air entrainment at the tank during vortexing, using the Knauss Jost "Swirling Water Problems at Intakes" chart, to be approximately 16 percent. Using the ideal gas law and accounting for decreased air volume due to the long length of piping between the CACST and the pumps the air entrainment at the pump suction was estimated to be approximately 7 percent. The team’s concern was that the potential cavitation of the pump at this air entrainment condition and the resulting fluctuations in turbine and motor power would cause pressure perturbations in the CA pumps’ suction line that would

induce a reset of the pressure switches. The design of the pressure switches includes a reset feature and a five second time delay.
The pressure switches, with a two out of three logic, were set to send the initiation signal at a pressure corresponding to a water level in the CA piping at approximately the 560 foot elevation. At this set point there is a 4.5 second margin to assure the RN/CA valves open to provide water to the CA pumps before the suction lines to all three pumps completely empties, causing pump damage due to loss of suction. Considering the five second time delay feature, there was virtually no margin to account for a reset of the pressure switches. The licensee's previous analysis had not identified or addressed the potential of failure of the assured water source alignment due to pressure switch reset. The team concluded that the licensee's failure to identify and address the potential impact of air entrainment on the RN/CA swap over pressure switches during analysis related to PIP C-97-01579 was inadequate corrective action.

During the inspection, the licensee evaluated the potential air entrainment from depletion of the CACST and consulted the CA pump vendor regarding the mechanism of pump cavitation causing pressure perturbations in the CA suction line. The pump vendor provided a letter stating that the conditions of seven percent air entrainment calculated by the team, would not cause pressure perturbations in the CA suction line. Based on this information the team concluded that there was no operability concern for the CA pumps resulting from the impact of the air entrainment on the RN/CA swap over pressure switch.

Analysis: Failure to perform adequate corrective action related to a 1997 deficiency associated with air entrainment in the auxiliary feed water system (CA) pump suction line identified in PIP C-97-01579 is a performance deficiency. This finding is related to the mitigating systems cornerstone. This finding is more than minor in accordance with MC 0612, in that the engineering calculation error which failed to include the potential impact of the air entrainment on the RN/CA swap over pressure switches resulted in a condition in which there was reasonable doubt on the operability of the CA pumps. The item is of very low safety significance (Green) because the licensee's engineering evaluations performed during the inspection determined that there was no adverse impact on the pressure switches and therefore no loss of the CA pumps capability for short term heat removal.

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, states, in part, that measures shall be established and implemented to assure that conditions adverse to quality such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformance are promptly identified and corrected. Contrary to the above, on May 8, 1997, measures were not implemented to assure conditions adverse to quality were promptly identified in that the impact of air entrainment on the RN/CA swap over pressure switches was not addressed when the CA suction line air entrainment issue was addressed in PIP C-97-01579. Because this failure to comply with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, is of very low safety significance and has been entered into the licensee's corrective action program, PIP C-07-01579, this violation is being identified as an NCV, consistent with Section VI.A. of the NRC

Enforcement Policy: NCV 05000413,414/2007006-01, Inadequate Corrective Action for CA System Air Entrainment Issue Identified in PIP C-97-01579.

.2.6 1NS-43A,-38B (ND pump discharge to containment spray system)

a. Inspection Scope

The team reviewed the design basis documentation, drawings, valve vendor manual, and the UFSAR to identify the design and operational requirements for the ND pump discharge to containment spray (NS) valves 1NS-43A, and -38B. The team reviewed the MOV mechanical analysis calculation for the valves to verify that the design bases, system pressure conditions, and degraded voltage conditions were used in developing and translating diagnostic setup requirements and acceptance criteria into the MOV diagnostic test. The team reviewed the minimum required and the maximum allowable thrust and torque as well as stall torque. The vendor manual and documentation of preventive maintenance activities were reviewed to verify preventive maintenance inspection, actuator lubrication, and stem lubrication were consistent with vendor recommendations. Maintenance documentation was reviewed to verify that MOVs were periodically tested and that appropriate torque switch settings were maintained. Maintenance history and corrective action history were reviewed to assess the capability for identification of component degradation.

The team reviewed elementary and single line diagrams for 1NS-43A (ND Pump A to NS Spray Header Containment Isolation Valve) and 1NS-38B (ND Pump B to NS Spray Header Containment Isolation Valve) as well as design basis documentation for the component power supply and control considerations in order to ensure that accident conditions were adequately addressed through the design. Operating and abnormal operating procedures were reviewed to ensure that the configuration of controls reflect and support the attributes identified in the design bases. Interlocks were reviewed to ensure that system architecture adequately supported the design bases under accident/event conditions. A walk down was performed to ensure valve position was consistent with plant operating conditions.

b. Findings

No findings of significance were identified.

.2.7 1CA-20,-27, 32 (CA pump mini-flow valves)

a. Inspection Scope

The team reviewed the design base documentation, drawings**,** pump vendor manual and related vendor correspondence, valve vendor manual, and the UFSAR to identify design, maintenance, and operational requirements for the CA pump mini-flow valves. This included review of system layout drawings for the auxiliary feedwater system and condensate system to verify the minimum recirculation system flow would not be reduced or blocked with the current design. The team reviewed vendor letters and

design basis documents to verify the inclusion of minimum flow requirements for both intermittent and long-term recirculation in response to NRC Bulletin 88-04. Also, flow rate indications for minimum flow during in-service testing were compared against design documents to verify the current installation provided adequate minimum flow. Walkdowns were conducted to verify the valve installation was oriented as designed and the minimum flow indication equipment was installed in an appropriate manner.

b. Findings

No findings of significance were identified.

.2.8 1KC-40B (KC pump mini-flow valve)

a. Inspection Scope

The team reviewed design basis and vendor documentation to identify the minimum flow requirements for the component cooling water pumps and reviewed testing procedures and piping layout drawings to verify the requirements were being satisfied with the current testing. The team reviewed the licensee's contingency actions for locking open the mini-flow valves to assure pump operability was not adversely impacted and runout protection was maintained. Test documentation was reviewed to verify that equipment degradation was monitored. Maintenance and corrective action documentation was reviewed to verify that equipment problems were adequately resolved. Field walkdowns were performed to assess observable material conditions.

b. Findings

No findings of significance were identified.

.2.9 Standby Shutdown Facility (SSF) Reactor Coolant Makeup Pump

a. Inspection Scope

The team reviewed NPSH calculations and installed equipment to verify head losses due to acceleration head would not impact the pump's capability to perform its design basis function of providing reactor coolant pump seal injection in a station blackout (SBO) event. Operating procedures were reviewed to verify inclusion of contingency actions associated with isolating the makeup pump suction line if a line break occurs. Facility elevation and piping layout drawings were reviewed and independent calculations were performed to verify adequate time was available to isolate the line break to assure the minimum spent fuel pool level was maintained. The team reviewed the calculations which verified the SSF pump safety function to inject adequate borated water to attain plant hot shut down conditions to verify appropriate pump operating parameters were used as design inputs. Maintenance test documentation was reviewed to verify that potential component degradation was monitored.

The inspectors reviewed elementary and single line diagrams for this component as well as design basis documentation for the power supply considerations to ensure that accident conditions were adequately addressed through the design. Service history for the power supply path was reviewed to evaluate potential degradation aspects were being identified, evaluated and trended. Operating and abnormal operating procedures were reviewed to ensure that the configuration of pump controls reflect and support the attributes identified in the design bases.

b. Findings

No findings of significance were identified.

.2.10 Emergency Diesel Generator (EDG) Ventilation Dampers and Fans

a. Inspection Scope

The team reviewed the design basis documentation, drawings, vendor manual, and the UFSAR to identify the design and operational requirements for the EDG ventilation system dampers and fans. The team reviewed the capability of the exhaust and intake dampers to withstand tornado loading by reviewing the licensee's structural loading model applied to this equipment. Additionally, the weak link analysis for the fans and dampers was reviewed to verify sub-component strength, hinges and blades, was adequate for potential stresses from anticipated natural events. The team reviewed the EDG room cooling fan specifications and test information to verify the adequacy of fan sizing. Maintenance and corrective action documentation was reviewed to verify that equipment degradation was monitored and that equipment problems were adequately resolved. The team reviewed the EDG space heat load calculations and system design capability to verify the room temperatures were maintained below a temperature that could impact the reliability of equipment and sub-components necessary for EDG operation. The review included heat load calculations, fan specification data sheet, room cooling test results, maximum ambient temperature, temperature of the diesel's combustion air, and the safety classification of the rooms' thermostats. The team walked down the accessible fan and damper equipment, and the associated inlet and outlet ductwork to assess the material condition of the system components.

b. Findings

No findings of significance were identified.

.2.11 EDG Turbo Charger

a. Inspection Scope

The team reviewed the UFSAR and design basis documents to select a sample of critical parameters for the emergency diesel generators intake and exhaust system to verify the turbo chargers ability to perform its design function of providing cool and clean compressed intake air for the diesel to meet generator load requirements.

Procurement records and requirements were reviewed to verify original equipment manufactures (OEM) turbo charger intake filters were being procured according to site procedures and quality controls.
The team included reviews of exhaust thermocouple calibration procedures to verify thermocouples are being calibrated and replaced if necessary in order to provide reliable indications of air filter performance. Turbo charger intercooler water flow rates and temperatures were reviewed to verify the coolers ability to provide the design heat removal rates during operation and assurance that they can perform their design function after major equipment overhauls. System walk downs were performed to assess observable material conditions.

b. Findings

No findings of significance were identified.

.2.12 125 VDC Vital Instrumentation and Control (I&C) Distribution Center (1EDF)

a. Inspection Scope

The team reviewed design basis documents, the UFSAR and system single line and elementary diagrams to identify the design basis functions of vital I&C distribution center 1EDF. The team reviewed the design documentation to verify the capability of the normal and alternate power supplies to support 1EDF loading during a design basis accident (DBA). Operating and abnormal procedures were reviewed to identify system configurations. Inspectors performed a walk down of the power distribution path to ensure alignment was consistent with plant operating conditions and that operational aspects were consistent with design bases. The team reviewed the translation of design base information and vendor specifications into equipment test acceptance criteria. Recent test results for the distribution center and sub-components were reviewed to ensure all results were consistent with requirements. Preventative maintenance program documentation governing this component was evaluated against service history records to ensure that indications of degraded performance were being identified, evaluated and trended. The licensee's response to several industry operating experience items were reviewed to ensure that applicability and action determinations were adequate.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for failure to perform adequate and timely corrective actions to resolve a potential equipment design deficiency of the 1DGBB battery and distribution which provided the alternate power supply to the 125 VDC Vital I&C distribution panel 1EDF.

Description: During review of design documentation to verify the adequacy of the alternate power supply to 1EDF, the team noted there was no analysis to support the adequacy of the 125 VDC Auxiliary Power System (EPQ) as the alternate power supply to support the 125 VDC Vital I&C distribution panel loading during a DBA. The licensee had previously identified this lack of design documentation on July 20, 2006, in PIP C-06-05322. The PIP corrective actions included plans to perform an analysis of voltage drop from the EPQ supply and guidance to the maintenance organization to avoid configurations that would depend upon the alternate power supply. Operations was not informed of the potential vulnerable configuration. The PIP documented an operational "assessment" that concluded there was no operability concern. This evaluation of operability appeared to be based primarily on engineering judgment with no documented technical input. No follow up technical evaluation was performed to verify the operability assessment. An apparent cause performed subsequent to the operability assessment identified a general concern for the adequacy of the supply for connected loads and distribution centers as well as a recommendation to avoid configurations where the Vital I&C System (EPL) battery is isolated from its associated distribution center (EDE or EDF) except when the associated train is out of service. This recommendation was not reevaluated by operations personnel for incorporation in light of the aforementioned operability assessment's conclusion. The final analysis of the adequacy of the power supply was scheduled for completion one year from the identification date. During the inspection, the team requested the licensee to provide the preliminary analysis of the battery and distribution configuration to 1EDF. The preliminary analysis provided to the team on April 2, 2007, indicated that 1DGBB was inadequate to supply the 1EDF loads for the first minute of a DBA due to voltage loss conditions. The team concluded that the corrective actions for this issue were inadequate in that Operations was not notified of the vulnerable configuration and the one year lead time to perform the analysis of the potential equipment design deficiency was not timely commensurate with its safety significance.

The team noted there were previous opportunities to identify that the alternate power supply was inadequate. A vital battery modification implemented in CNCE-61191, Cable Replacement from EDA to EDE and from EDD to EDF to Ease Voltage Drop Considerations to the Loads Fed from EDE and EDF, dated July, 1997, replaced the cable between the vital battery and 1EDF and other 125 VDC vital I&C distribution centers with larger cable due to voltage drop concerns identified in July of 1996. Although the same undersized cable was used on the alternate power supply, it was not evaluated nor was the adequacy of the alternate sources to supply these distribution centers evaluated. The EDG batteries were replaced with larger batteries in 2005 and 2006 (CNCE-11447, Replace Unit 1 EPQ 125VDC DG Auxiliary Power Battery, dated May, 2005, and CNCE-21447, Replace Unit 2 EPQ 125VDC DG Auxiliary Power Battery, dated March, 2006) as a result of adverse EDG battery performance trends identified in PIP C-03-06703. No review of the battery or distribution hardware was performed to support the battery modification or verify its capability as an alternate power supply to 1EDF.

Analysis: Failure to perform adequate and timely corrective actions to resolve a potential equipment design deficiency of the 1DGBB battery and distribution network is identified as a performance deficiency. This finding is more than minor because it affects the mitigating systems cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events in that 125 VDC distribution center 1EDF provides control power to critical equipment such as the 4.16kV vital bus which aligns power to ECCS pumps and valves.
The finding is associated with the cornerstone attribute of design control. This finding is of very low safety significance (Green) because the team identified no occurrence, since the potential design deficiency was identified on July 20, 2006, in which the station was aligned in the vulnerable condition relying on the alternate power supply to 1EDF. Additionally, the normal power supply, vital batteries, is a highly reliable power source and the alignment to the alternate power source requires manual action. There was no loss of the 1EDF safety function to provide adequate vital I&C control power for safe shutdown of the plant. This finding involved the crosscutting area of Problem Identification and Resolution [Aspect 15] because the evaluation, specifically the operability assessment, was inadequate and contributed the inadequacy of subsequent corrective actions.

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, states, in part, that measures shall be established and implemented to assure that conditions adverse to quality such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, on July 20, 2006, a condition adverse to quality was not promptly identified and corrected. Specifically, PIP C-06-05322, dated July 20, 2006, identified that the alternate power supply to the 125 VDC Vital I&C distribution center 1EDF was not validated as adequate to supply the distribution center vital loads and no action was taken to determine the adequacy of the power supply or inform operations of the potential vulnerable configuration. Preliminary analysis provided on April 2, 2007 indicated that the power supply was inadequate to supply the 1EDF loads for the first minute of a DBA due to voltage loss conditions. This failure to comply with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, is of very low safety significance and has been entered into the licensee's corrective actions program, PIP C-06-05322, which was revised to include corrective action sequence 6 addressing this inadequate corrective action. This violation is being identified as a Non-cited Violation (NCV), consistent with Section VI.A. of the NRC Enforcement Policy: NCV 05000413,414/2007006-02, Failure to Perform Adequate and Timely Corrective Action to Identify and Resolve an Equipment Design Deficiency of the Alternate Power Supply for the 125 VDC Vital I&C Distribution Center 1EDF.

.2.13 Engineered Safeguards (ESG) Auxiliary Relays HA, LE (ESGAX1, ESGAX2)

a. Inspection Scope

The team reviewed the vendor manual, specifications, and recent equipment technical bulletins to identify vendor maintenance requirements. Maintenance documentation was reviewed to verify incorporation of vendor recommendations. Completed maintenance

and surveillance documentation was reviewed to verify that anomalies were properly documented and resolved. Maintenance and surveillance schedules were reviewed to verify that vendor and Technical Specification periodicity requirements were met. Maintenance and corrective history documentation was reviewed to verify that equipment performance trending was performed and adverse conditions were corrected.

The team performed an industry data base search for relay performance problems to assess applicability to the ESG auxiliary relays. A walkdown of the equipment area was performed to assess potential adverse conditions or equipment hazards.

b. Findings

No findings of significance were identified.

.2.14 EDG Sequencer

a. Inspection Scope

The team reviewed the design base documentation, UFSAR, and sequencer wiring drawings to identify the design base requirements for the EDG sequencer. Sub-component refurbishment schedules were reviewed to verify equipment was adequately maintained. An industry data base search was performed to identify problems related to the sequencer sub-components and to assess applicability to station equipment. The licensee vendor contact program as related to sequencer sub-components was reviewed to verify associated technical information was received and assessed. The corrective action history of the sequencer sub-components was reviewed to verify that identified problems were resolved. The seismic qualification of the sub-components was reviewed to verify that the equipment was qualified for the appropriate seismic requirements.

b. Findings

No findings of significance were identified.

.2.15 Reactor Trip Breakers

a. Inspection Scope

The team reviewed the DS breaker maintenance manual and vendor technical update information for the Westinghouse DS-416 reactor trip breakers to determine whether vendor requirements have been incorporated into station maintenance and surveillance procedures. Completed maintenance and surveillance documentation was reviewed to verify that anomalies were properly documented and resolved. Maintenance and surveillance schedules were reviewed to verify that vendor and Technical Specification periodicity requirements were met. Maintenance and corrective history documentation was reviewed to verify that equipment performance trending was performed and adverse conditions were corrected.

The team reviewed the station surveillance guidance for breaker trip force testing to verify compliance with the maintenance manual recommendations and that acceptance criteria were consistent with the manual. A walkdown of the equipment area was performed to assess potential adverse conditions or equipment hazards.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion V, Procedures, Instructions and Drawings, for failure to follow procedure NSD 319, Vendor Technical Information Program, Rev. 2, which requires performance of technical impact reviews of maintenance and surveillance procedures due to vendor manual changes and technical updates.

Description: The team identified inconsistencies between the licensee's reactor trip breaker surveillance procedure, SI/0/A/5100/002, and the applicable vendor information. Westinghouse Maintenance Program Manual for Safety Related Type DS Low Voltage Metal Enclosed Switchgear, dated March, 1999, was incorporated into Catawba Vendor Manual CNM 1399.40-0016.001, MPM-DS Breaker Maintenance Program Manual for 1E Low Voltage Metal Encl Swgr, Revision 0, in April 2001. Changes to the March 1999 Westinghouse manual identified in Technical Bulletin W-TB-00-01-R0, Westinghouse DS Circuit Breaker Issues, 04/24/00, had not been incorporated into the Catawba vendor manual. Vendor Manual CNM 1399.40-0016.00, and the unincorporated Westinghouse Technical Bulletin W-TB-00-01-R0 contained changes for which there was no technical impact review and which were inconsistent with the performances of Surveillance Procedure SI/0/A/5100/002 on 9/28/06 and 1/19/07. The following procedure/manual inconsistencies were identified:

1. The Undervoltage Trip Attachment (UVTA) Dropout Voltage Test described in Vendor Manual Section 11-5-1, Step 5 was omitted from the surveillance procedure.

2. Undervoltage Trip Attachment (UVTA) Force Check Verification method in surveillance procedure Section 3.2.18 did not accurately reflect, in form nor content, that in the vendor manual, Section 11-5.3, as updated by Westinghouse Technical Bulletin W-TB-00-01-R0.

3. Breaker trip load force testing values in Section 3.2.7 of the procedure were assessed on an average of three tests instead of the worst case of three tests as described in Section 11-4 of the vendor manual.

4. Applicable maintenance procedures did not provide criteria for the replacement of the reactor trip breakers or their sub-components based on the number of cycles specified in Table 5-1 of the vendor manual.

There were no technical impact review evaluations to assess the acceptability of these deviations from the vendor technical manual. The licensee has entered this item into the corrective action program as PIPs C-07-01648 and C-07-01649.

Analysis: The team concluded that the failure to perform technical impact reviews of reactor trip breaker manual changes and technical updates was a performance deficiency. This finding is more than minor in a accordance with MC 0612. It affects the mitigating systems cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events and is associated with the attribute of procedure quality, in that procedure inconsistencies were identified in procedure SI/0/A/5100/002, Reactor Trip Breaker Surveillance Procedure, Rev. 18, which indicated that the licensee routinely failed to perform engineering evaluations on similar issues. The finding was determined to be of very low safety significance (Green), using the safety significance determination process (SDP) phase 1 worksheet, because there was no loss of the reactor trip breaker safety function to open on a scram signal.

Enforcement: 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings states in part that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions and procedures. Duke corporate procedure NSD 319, Vendor Technical Information Program, Rev. 2, requires that vendor notifications, updates, and revisions receive a technical impact review to address revising maintenance, surveillance or test procedures. Contrary to the above, activities affecting quality were not accomplished in accordance with prescribed procedures, in that a technical impact review was not performed for reactor trip breaker vendor technical updates and manual changes. Changes to the March ,1999, revision of the Westinghouse Maintenance Program Manual for Safety Related Type DS Low Voltage Metal Enclosed Switchgear, and a vendor update dated 04/24/2000, received no technical impact review. Subsequently, there was no justification for identified deviations between the vendor manual and the maintenance procedure. This issue is identified in the licensee's corrective action program as PIPs C-07-01648 and C-07-01649. Because this failure to comply with 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, is of very low safety significance and has been entered into the licensee's corrective action program, it is being identified as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000413,414/2007006-03, Failure to Follow Procedure for Analyzing the Impact of Updated Vendor Technical Information on Reactor Trip Breaker Maintenance and Inspection Procedures.

.3 Review of Low Margin Operator Actions

a. Inspection Scope

The team performed a margin assessment and detailed review of a sample of risk significant, time critical operator actions (TCOAs). Where possible, margins were determined by the review of the assumed design basis and UFSAR response times and performance times documented by job performance measure (JPM) results within

operator time critical task verification tests. For the selected operator actions, the team performed a walk through of associated Emergency Procedures (EPs), Abnormal Operating Procedures (AOPs), Annunciator Response Procedures (ARPs), and other operations procedures with appropriate plant operators and engineers to assess operator knowledge level, adequacy of procedures, availability of special equipment when required, and the conditions under which the procedures would be performed. Detailed reviews were also conducted with risk assessment engineers, engineering safety analysts, training department leadership, and through observation and utilization of a simulator training period to further understand and assess the procedural rationale and approach to meeting the design basis and UFSAR response and performance times. TCOAs in response to the following events were reviewed:

- loss of instrument air (VI)
- initiation of SSF Makeup flow to the RCP seals
- manual throttling of auxiliary feedwater flow on a loss of instrument air
- isolation of the steam supply to the turbine driven auxiliary feedwater pump from a ruptured steam generator
- isolation of an intersystem LOCA

b. Findings

No findings of significance were identified.

.4 Review of Industry Operating Experience

a. Inspection Scope

The team reviewed selected operating experience issues that had occurred at domestic and foreign nuclear facilities for applicability at the Catawba Nuclear Station. The team performed an independent applicability review and issues that appeared to be applicable to the Catawba Nuclear Station were selected for a detailed review. The issues that received a detailed review by the team included:

- 10 CFR 21 on Rotork Controls
- High ND suction pressure condition (PIP C-05-2259)
- Low EDG Frequency - impact on safety related pumps
- GL-83-28, Salem ATWS Event - related to reactor trip breaker reliability
- GL 89-13 Service water system problems - related to KD Hx

b. Findings

No findings of significance were identified.

.5 Review of Permanent Plant Modifications

a. Inspection Scope

The team reviewed three modifications related to the selected risk significant components in detail to verify that the design bases, licensing bases, and performance capability of the components have not been degraded through modifications. The adequacy of design and post modification testing of these modifications was reviewed by performing activities identified in IP 71111.17, Permanent Plant Modifications, Section 02.02.a.
Additionally, the team reviewed the modifications in accordance IP 71111.02, Evaluations of Changes, Tests, or Experiments, to verify the licensee had appropriately evaluated them for 10 CFR 50.59 applicability. The following modifications were reviewed:

- CD 200611, Install Bypass Line Around 2FW-28 and 2FW-56, dated 1/18/07.
- CE 61191, Cable Replacements, EDA to EDE, and EDD to EDF, dated 7/16/98.
- CNCE-73131, Replacement of Obsolete HFB Molded Case Circuit Breakers, dated 6/16/04.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4AO6 Meetings

On April 19, 2007, the team presented the inspection results to Mr. Pitesa, Station Manager, and other members of the licensee staff. The team returned all proprietary information examined to the licensee. No proprietary information is documented in the report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Brewer, Safety Assurance Manager
B. Ferguson, Mechanical/Civil Engineering Manager
G. Hamrick, Engineering Manager
R. Hart, Regulatory Compliance Manager
C. Kidd, Primary Systems Supervisor
K. Phillips, Operations Support Manager
B. Pitesa, Station Manager
T. Simril, Balance of Plant Supervisor

NRC

M. Cain, RII, Acting Chief, Engineering Branch 1
A. Sabisch, Senior Resident Inspector
G. Williams, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Open/Closed

05000413,414/2007006-01	NCV	Inadequate Corrective Action for CA System Air Entrainment Issue Identified in PIP C-97-01579 (Section 1R21.2.5)
05000413,414/2007006-02	NCV	Failure to Perform Adequate and Timely Corrective Action to Identify and Resolve an Equipment Design Deficiency of the Alternate Power Supply for the 125 VDC Vital I&C Distribution Center 1EDF. (Section 1R21.2.12)
05000413,414/2007006-03	NCV	Failure to Follow Procedure for Analyzing the Impact of Updated Vendor Technical Information on Reactor Trip Breaker Maintenance and Inspection Procedures (Section 1R21.2.15)

DOCUMENTS REVIEWED

Calculations

CNC-1381.05-00-0011, 125 VDC Vital Instrumentation and Control Power System Battery and Battery Charger Sizing Calculation, Rev. 8
CNC-1381.05-00-0149, 125 VDC Vital I&C Power System (EPL) Voltage Drop Calculation, Rev. 04B
CNC-1205.19-00-0170, Operability Eval. of Rotork Part 21 for Switch Mechanisms, Rev. 1
CNC-1205.41-00-0026, 1/2RN-291, 351 Required Force Evaluation Supporting the Air Operated Valve (AOV) Program, Rev. 1
CNC-1212.00-0013, Diesel Generator Building HVAC Calculations, Rev. 25
CNC-1223.42-00-001, Confirmation of CA System RN Transfer Scheme Adequacy, Rev. 17
CNC-1223.42-00-0058, Auxiliary Feedwater System Operability Calculation (Framatome), 6/29/2000
CNM 1210.04-0276.001, I/M, Model 580A-0 Differential Pressure Indicating Switch, Rev. D4
CNC-1211.00-00-0036, SSF Temperature Calculation / Auxiliary Feedwater Pump Room Area, Rev. 2
CNC-1381.05-00-0118, Station Blackout Coping Study, Rev. 4
CNC-1223.12-00-0063, Acceptance Criteria Verification for PT/1(2)/A/4400/01, ECCS Flow Balance, Rev. 8
CNC-1210.04-00-0055, Surge Tank Level Setpoint Calculation, Rev. 6
IP/1/A/3140/003 A, Calibration Procedure Auxiliary Feedwater System Train A and Train B Loss of Suction to Pumps, Rev. 033
PT/1/A/4250/003 D, RN to CA Pumps Suction Transfer Periodic Test, Rev. 050
CNC-1223.42-00-0054, Analysis of CA System Suction Sources, Rev. 1
CNC-1205.19.00-0039, Generic Letter 89-10 Calculations for ND System ND036B, Rev. 7
CNC 1205.19-00-0033, GL 89-10 Set-Up Calculation for valve NI173A and NI178B, Rev. 2
Calculation CNC-1223.24-00-0018, Acceptable RN Flow and Fouling in the KC heat exchangers, Rev. 4
CNC-1223.04-00-0070, Standby Makeup Pump NPSH and Suction Damper Evaluation
CNC-1223.04-00-0072, Reactor Cooling Pumps No. 1 Seal Leakoff Annunciator Setpoint for Unit 1 and Unit 2, Rev. 1
CNC-1223.04-00-0072 Att. G, Reactor Coolant Pumps No. 1 Seal Leakoff Annunciator Setpoint for Unit-1 and Unit-2, Rev. 1
CNC-1205.19-00-0034, Generic Letter 89-10 MOV Calculation for NV System Valves: 1(2)NV236B and 1(2)NV877, Rev. 1
CNC-1381.05-00-0198, Catawba Unit 1 ETAP Power Station Auxiliary Power Station Auxiliary Power System Voltage Study, Rev. 3
CNC 1205.19-00-0034, GL 89-10 Set-Up Calculation for NV System Valves: 1(2)NV236B and 1(2)NV877, Rev. 2
CNC 1205.19-00-0047, GL 89-10 Set-Up Calculation for Valve 1(2)NS038B and 1(2)NS043B, Rev. 5
CNC-1381.05-00-0162, Voltage Analysis of Motor Starter and Interposing Relay Coils at Catawba, Rev. 1

CNLD-0114-01.01, Logic Diagram Diesel Generator Load Sequencing System A-Train (EQB), Rev. 14
CNLD-0114-01.02, Logic Diagram Diesel Generator Load Sequencing System A-Train (EQB), Rev.14

Operating Procedures

AOP-036, Safe Shutdown Following a Fire, Rev. 39
AOP-036.02, Fire Area 1-A-BAL-A, Rev. 3
AOP-036.05, Fire Area 1-A-CSRB, Rev. 4
AOP-036.08, Fire Area 1-A-SWGRB, Rev. 3

Operations Training Related Documents

Standby Shutdown Facility Lesson Plan, Rev. 32
Safety Injection System Lesson Plan, Rev. 41
Air Systems Lesson Plan, Rev. 34
Auxiliary Feedwater System Lesson Plan, Rev. 53
Emergency Procedures Intro Lesson Plan, Rev. 6

Procedures

IP/0/A/3850/023, Molded Case Circuit Breaker Inspection and Testing Procedure, Rev. 083
IP/0/A/3820/040, AOV Diagnostic Testing Using the Viper diagnostic System, Rev. 003
OP/1/A/6350/008, 125 VDC/120 VAC Vital Instrument and Control Power System, Rev. 054
OP/1/A/6350/006, 125 VDC Diesel Auxiliary Power, Rev. 032
PT/1/A/4700/012, SSF Control Panel Functional Verification, Rev. 003
PT/1/A/4200/018, NI System Power Disconnect Test, Rev. 005
PT/0/A/4400/008A, RN Flow Balance Train A, Rev. 052
PT/1/A/4700/012, Standby Shutdown Facility (SSF) Control Panel Functional Verification Unit 1, Rev. 003
IP/0/B/3820/002B, Rotork Actuator Preventative Maintenance, Rev. 028
IP/0/A/3820/004, Operating Checkout of Limitorque and Rotork Valve Actuators, Rev. 303
IP/0/A/3820/004A, MOV Diagnostic Testing, Rev. 046
IP/0/A/3820/004B, Guidelines for Differential Pressure (DP) Testing of Motor Operated Valves, Rev. 008
IP/0/A/3820/004C, Troubleshooting/Root Cause Failure Analysis of MOVs, Rev. 008
IP/0/A/3820/007, Maintenance of Rotork Actuators, Rev. 045
IP/0/A/3820/009, Removal, Replacement and Field Set-Up of Rotork Actuators, Rev. 065
IP/0/A/3820/038, Measuring AOV Thrust Loads Using Valve Vision, Rev. 005
IP/0/A/3820/040, AOV Diagnostic Testing Using the Viper Diagnostic System, Rev. 003
PT/1/A/4400/001, ECCS Flow Balance, Rev. 039
IP/0/A/3816/010, Barton Model 580 and 581 DP Switch Calibration, Rev. 028
IP/1/A/3140/003 A, Calibration Procedure Auxiliary Feedwater System Train A and Train B Loss of Suction to Pumps, Rev. 33
PT/1/A/4250/003 A, Auxiliary Feedwater Motor Driven Pump 1A Performance Test, Rev. 059
PT/1/A/4250/003 E, CA System Discharge Control Valve Throttling Procedure, Rev. 032

AP/1/A/5500/006, Loss of S/G Feedwater, Rev. 036
PT/2/A/4400/006 C, KC Heat Exchanger 2A Heat Capacity Test, Rev. 015
PT/1/A/4400/006 D, KC Heat Exchanger 1B Heat Capacity Test, Rev. 019
PT/1/A/4400/006 C, KC Heat Exchanger 1A Heat Capacity Test, Rev. 023
PT/1/A/4250/003 C, Turbine Driven Auxiliary Feedwater Pump Performance Test, Rev. 94
PT/1/A/4250/003 B, Auxiliary Feedwater Motor Driven Pump 1B Performance Test, Rev. 47
PT/1/A/4250/003 A, Auxiliary Feedwater Motor Driven Pump 1A Performance Test, Rev. 59
PT/1/A/4200/007 C, Standby Makeup Pump #1 Performance Test, Rev. 37, 11/01/06
PT/1/A/4200/007 C, Standby Makeup Pump #1 Performance Test, Rev. 37, 2/07/07
MP/0/A/7150/097, Standby Makeup Pump Pulsation Dampers Preventative Maintenance Inspection, 6/29/05
MP/0/A/7150/097, Standby Makeup Pump Pulsation Dampers Preventative Maintenance Inspection, 5/31/06
PT/1/A/4400/003 E, Component Cooling Miniflow Verification, Rev. 7
IP/1/B/3630/001 A, D/G-1A Engine Intake and Exhaust System (VN), 6/14/06
IP/1/B/3630/001 B, D/G-1B Engine Intake and Exhaust System (VN), 7/18/06
MP/0/A07400/042, Diesel Engine Turbo charger Removal and Replacement, Rev. 26
NSD 219, Instrument and Electrical Device Calibration Out of Tolerance, Rev. 3
SI/0/A/5100/002, Westinghouse DS-416 Air Circuit Breaker Inspection and Maintenance, Rev. 18
IP/1/A/3200/001 A, Solid State Protection System (SSPS) Train A Periodic Testing, Rev. 007
IP/1/A/3200/001 B, Solid State Protection System (SSPS) Train B Periodic Testing, Rev. 005
IP/1/A3670/001 A, Calibration Procedure for D/G-1A Load Sequencer Timers (EQB), Rev. 035
PT/1/A/4200/009 A, Auxiliary Safeguards Test Cabinet Periodic Test, Rev. 228
PTS/1/A/4200/009, Engineered Safety Features Actuation Periodic Test, Rev. 176
PT/0/A/4700/061, Time Critical Operator Action Review, Rev. 3
AP/1/A/5500/006, Loss of S/G Feedwater, Rev. 36
AP/2/A/5500/006, Loss of S/G Feedwater, Rev. 28
AP/1/A/5500/017, Loss of Control Room, Rev. 47
AP/2/A/5500/017, Loss of Control Room, Rev. 42
ABG/1/5500/017, loss of Control Room, Rev. 0
AP/0/A/5500/020, Loss of Nuclear Service Water, Rev. 36
AP/1/A/5500/021, Loss of Component Cooling, Rev. 35
AP/2/A/5500/021, Loss of Component Cooling, Rev. 29
ABG/1/5500/021, Loss of Component Cooling, Rev. 1
AP/0/A/5500/022, Loss of Instrument Air, Rev. 25
EP/1/A/5000/E-3, Steam Generator Tube Rupture, Rev. 31
EBG/1/5000/E-3, Steam Generator Tube Rupture, Rev. 21
EP/2/A/5000/E-3, Steam Generator Tube Rupture, Rev. 27
EP/1/A/5000/ECA-0.0, Loss of All AC Power, Rev. 31
EP/2/A/5000/ECA-0.0, Loss of All AC Power, Rev. 30
EBG/1/5000/ECA-0.0, Loss of All AC Power, Rev. 10

Design Changes/Modifications

CNCE-61191, Cable Replacement Between 1EDA and 1EDE, and Between 1EDD and 1EDF, 7/16/97
CNCE-73131, Replacement of Obsolete HFB Molded Case Circuit Breakers, 6/16/04
CNCE-2929, Alteration of 1ND-002A and 1ND036B Control Circuits to Allow Interlock with FW and NS Valves to be Dependent Upon Valve Position Only, 11/19/90
C-00-03039, Reactor Trip Breaker Maintenance Program Manual Update, 10/12/00

Plant Investigation Reports (PIPs)

C-06-05322	Deficiency Identified in Configuration Management of the EPQ (125 VDC Diesel Auxiliary Power) System Due to Lack of a Documented Specific Voltage Drop Calculation
C-06-02348	Review of Westinghouse TB-06-02
C-02-01881	Evaluation of HFD Use in Obsolete HFB Applications
C-01-06109	Vendor Notes HFB Obsolescence
C-04-02668	Rotork Controls Part 21 Notification to Duke Power
C-04-04205	2NS-018A Valve Indication Fails to Intermediate on Functional Verification Stroke
C-03-05824	2NS-018A Valve Indication Fails to Intermediate During IMV
C-05-07464	1EMXS Inadvertently De-energized Due to Failure of Valve 1NV-865A
C-06-07763	1NS-38B Failed to Stroke During IWV Testing
C-05-01926	Unplanned Entry Into TS 3.8.1 Due to D/G 1A Breaker Tripping
C-06-07815	Evaluate Applicability of Low Suction Pressure to Catawba CA System
C-97-01579	A Potential CA System Operability Problem
C-07-00656	There is no evaluation of the impact of the allowed ranges of EDG frequency.
C-07-00735	Evaluate the Effects of +/- 2% EDG Frequency on MOVs
C-04-00844	Conservative Error Found in Calculation of NS HX 1B Fouling Factor
C-02-06278	Reactor Trip Bypass Breaker 2B Would Not Close
C-03-06331	Welds on Rx Trip Breaker Cubicle 1BYB Starting to Crack
C-04-05111	Train A Reactor Trip Breaker Did Not Open as Expected During Test
C-06-02025	Simulator Anomaly Involving Sequencer Load, Shed, and Reload
C-06-02083	U1 Reactor Trip Breaker "B" Spuriously Opened During Testing
C-06-03982	Diesel Stop Button Pressed Shortly After Emergency Start
C-06-04620	Review of Unit 2 Alarm Log During Unit 2 LOOP
C-06-05592	Determine Effect of Transferring the 1B ASP to Local on the 1B Load Sequencer
C-06-05921	Events Recorder Data Was not Recorded for Rx Trip Breaker
C-06-06342	Aux Relay for DFCS/CFP De-energized, 09/07/2006
C-06-06543	Events Recorder Did Not Record Valid Trip Time for Reactor Trip Breaker 2RTA
C-06-07751	Unanticipated 1A Blackout Signal Received During Outage Activities
C-06-08122	Disparity Between Nomenclature in Design Documents and Field
C-07-00114	Events Recorder Did Not Record Valid Trip Time for Reactor Trip Breaker 2RTA
C-07-01648	Enhance the Reactor Trip Breaker Maintenance Procedure (SI/0/A/5100/002) Relative to steps for Breaker Trip Force Testing
C-98-01712	High Contact Resistance for the SSPS P-4 Function
G-02-00206	Track Vendor Recontact Activities Performed by OEA in 2002

G-03-00277 Track Vendor Recontact Activities Performed by OEA in 2003
G-04-00379 Track Vendor Recontact Activities Performed by OEA in 2004
G-05-00392 Track Vendor Recontact Activities Performed by OEA in 2005
G-06-00445 Track Vendor Recontact Activities Performed by OEA in 2006
G-99-00247 Generic management attention item from August 1999 Nuclear Safety Review Board (NSRB) meeting
C-98-0188 Evaluate the impact of an operating experience item on TCOAs for a steam generator tube rupture
C-98-0197 Evaluate the impact of an operating experience item on TCOAs for initiating SSF Makeup Pump flow to RCP seals in 10 minutes
C-98-1062 A comprehensive list of TCOAs is needed
C-00-0484 Evaluate applicability of PIP M-99-3778, some locally operated valves used during abnormal or emergency events are not tested or in PM programs
C-00-2077 Evaluate applicability of PIP M-00-0263, use of break away locks on some time critical valves
C-01-1036 Documents the implementation of NSRB recommendations made in PIP G-99-0247
C-03-2905 Evaluate the restoration of instrument air or alignment of backup nitrogen as a TCOA
C-03-2908 Evaluate the manual throttling of auxiliary feedwater to the intact steam generators following a loss of instrument air as a TCOA
C-03-2913 Evaluate the closure of ND cold leg injection valves to isolate an ISLOCA as a TCOA
C-04-2648 The completion time for TCOAs has not been evaluated with respect to the length of time required to obtain safety gear
C-04-3291 Discrepancies noted during the performance of PT/0/A/4700/061, Time Critical Operator Action Review
C-04-3629 Operator are unclear on use of PPE during emergency situations
C-05-1033 OEM clarification of the volume in reactor coolant pumps
C-05-1113 The NRC residents identified two issues relating to TCOAs
C-06-6477 The interface between security and operations should be examined to ensure an appropriate level of communications and planning is occurring

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01128619 01, 1NI178B; Anneal & PFM MOV Thrust, 11/26/2006
01128620 01, 1NI173A; Anneal & PFM MOV Thrust, 11/26/2006
01721486 01, 1ND036B; R/R Rotork Act. / Center Col. Oil Leak, 12/4/2006
98263619 01, CE10737 WU20, 1NS038B, PFM MOV 89-10 Test, 10/26/2000
01702576 01, CD100548, 1NS043A; PCKING PM, R/R W/40NAX1-86 ACT, MOV Test, 11/21/2006
98579974 01, 1IRE: PM Reactor Trip Breaker A, 1/18/07
98579975 01, 1IRE: PM Reactor Trip Breaker B, 11/2/06
98683051 01, 1EQB - Cal 1A DG Load Seq Timers

98683235 01, 1IRE: PM Reactor Trip Breaker A, 9/28/06
98683236 01, 1IRE: PM Reactor Trip Breaker B, 9/7/06
98736158 03, 2IRE: PM Reactor Trip Bypass Breaker A

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CNEE-0151-01.32, Elementary Diagram Safety Injection System (NI) ND Hdr to NC Col Legs C and D Isolation Valve 1NI173A, Rev. 13
CNEE-0151-01.45, Elementary Diagram Safety Injection System (NI) ND Hdr to NC Col Legs A and B Isolation Valve 1NI178B, Rev. 13
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CNEE-0159-01.05, Elementary Diagram Containment Spray System (NS) Safety Injection Spray Isolation Valve 1NS043A, Rev. 14
CNEE-0159-01.16, Elementary Diagram Containment Spray System (NS) NS-Sys. Containment Isolation Valves 1NS29A, 1NS32A, and 1NS43A, Rev. 12
CNEE-0159-01.06, Elementary Diagram Containment Spray System (NS) Safety Injection Spray Isolation Valve 1NS038B, Rev. 14
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Load Logic Voltage Sensing Reset & Load Shed Circuits, Rev. 6
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PIPs initiated due to CDBI activity:

C-07-01413	Inconsistencies between SSF procedures
C-07-01414	Electrical calculation not updated following modification
C-07-01428	No rechargeable flashlights in SSF control room
C-07-01437	No validation of lighting adequacy for operator actions in SSF
C-07-01564	Inconsistent temperature information between EQCM and heat load calculation for CAPT space
C-07-01579	Low design margin for RN/CA pressure switch. Preferred CA water source transfer scheme did not address pressure switch time delay feature or potential adverse suction piping conditions
C-07-01643	Minor Error in Emergency Procedures Lesson Plan
C-07-01648	Inconsistent guidance between Rx trip breaker vendor manual and Rx trip maintenance Procedure
C-07-01649	Rx trip Breaker cycling of breaker attachments not tracked as recommended by vendor for cycle life concerns.
C-07-01671	Typographical error in Loss of VI procedure
C-07-01889	Testing the unused breaker pole of 3-pole breaker used in 2-Pole (DC) application
C-06-05322	Deficiency Identified in Configuration Management of the EPQ (125 VDC Diesel Auxiliary Power) System Due to Lack of a Documented Specific Voltage Drop Calculation (existing PIP revised)
C-07-01893	Interpretation of BTP PSB-1, position 2