UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

October 20,2003

EA-03-191

Duke Energy Corporation
ATTN: Mr. D. M. Jamil
Site Vice President
Catawba Nuclear Station
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION - EXERCISE OF DISCRETION NRC INTEGRATED INSPECTION REPORT 05000413/2003004 AND 05000414/2003004

Dear Mr. Jamil:

On September 27, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Catawba Nuclear Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 9, 2003, with you and members of your staff.

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

This report documents one NRC-identified finding of very low safety significance (Green) which was determined to be a violation of NRC requirements. However, because of the very low safety significance and because the issue was entered into your corrective action program, the NRC is treating the finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. While a second violation of NRC requirements was also identified, we have concluded that Catawba's actions did not contribute to the degraded condition; therefore, no performance deficiency was identified. Based on these facts, I have been authorized, after consultation with the Office of Enforcement, to exercise enforcement discretion in accordance with section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violation. An evaluation was performed and we have determined that this was an issue of very low safety significance. If you contest the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC, 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC, 20555-0001; and the NRC Resident Inspector at the Catawba Nuclear 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 (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/

Robert Haag, Chief
Reactor Projects Branch 1
Division of Reactor Projects

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

Enclosure: Integrated Inspection Report 05000413/2003004 and 05000414/2003004
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-413, 50-414
License Nos:	NPF-35, NPF-52
Report No:	05000413/2003004, 05000414/2003004
Licensee:	Duke Energy Corporation
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	4800 Concord Road York, SC 29745
Dates:	June 29, 2003 - September 27, 2003
Inspectors:	E. Guthrie, Senior Resident Inspector M. Giles, Resident Inspector R. Carroll, Senior Project Engineer (Sections 1R04.2, 1R05.2) Paul Fillion, Reactor Inspector (Section 4OA5.2) Randy Moore, Senior Reactor Inspector (Section 4OA5.1) Mike Scott, Senior Reactor Inspector (Section 1R12.2)
Approved by:	R. Haag, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000413/2003-004, IR 05000414/2003-004; 6/29/2003-9/27/2003; Catawba Nuclear Station, Units 1 and 2; Other Activities.

The inspection covered a three month period of inspection by resident inspectors and announced regional inspections by one project engineer and three reactor inspectors. One non-cited violation (NCV) was identified. 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 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation for failure to comply with 10 CFR 50 Appendix B, Criterion 3, Design Control, due to inadequate design measures. Specifically, the licensee failed to assure adequate relief valve sizing to prevent exceeding the design pressure of the component cooling water (KC) piping in the event of a reactor coolant pump (RCP) thermal barrier rupture.

  This finding represented a performance deficiency because it involved the licensee’s failure to assure the design adequacy of the KC relief valve to protect the piping from exceeding design limits in the event of a RCP thermal barrier leak. This finding is more than minor because it affects the Reactor Safety Cornerstone, Barrier Integrity attribute of design control and affects the associated objective. The inadequately sized relief valve represents a potential open path way in the physical integrity of the reactor containment. The NRC performed a phase three significance determination screening analysis and concluded the finding is of very low safety significance. (Section 4OA5.1)

B. Licensee-identified Violations

None

Report Details

Summary of Plant Status:

Unit 1 began the inspection period operating at 100 percent Rated Thermal Power (RTP). Power was reduced to 95 percent RTP on August 14, 2003, due to Loop 1A hot leg temperature detector failures. On August 29, 2003, the Unit tripped due to the failure of a pressurizer pressure instrument. The Unit was returned to 100 percent RTP on September 10, 2003, and remained there for the rest of the inspection period.

Unit 2 began the inspection period operating at 100 percent RTP. On July 26, 2003, power was reduced to 45 percent to replace the ‘C’ reactor coolant loop, channel 2 flow transmitter. The Unit was returned to 100 percent RTP on July 27, 2003. On August 16, 2003, power was reduced to approximately 80 percent RTP due to the failure of two refueling water storage tank level transmitters. The Unit was returned to 100 percent RTP on August 16, 2003, and remained there for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee’s preparation for adverse weather associated with high ambient temperature conditions. The review included an assessment of the station Engineering Support Program Hot Weather Protection; Electrical Systems and Equipment Engineering Corrective Work Orders Required for Hot Weather; and the Protection Functional Equipment Group Heat Protection Model Work Order. Two risk significant systems were selected for this inspection: nuclear service water (RN) and emergency diesel generator (EDG) systems. The inspectors walked down portions of the systems on August 27, 2003. The inspectors also conducted interviews with engineering personnel to discuss station administrative and procedural guidance and controls, which provided protective measures for the RN and EDG systems.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors verified the critical portions of equipment alignments for selected trains that remained operable while the redundant train was inoperable. The inspectors reviewed plant documents to determine the correct system and power alignments, and the required positions of select valves and breakers. The inspectors verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact mitigating system availability. The inspectors verified the following partial system alignments and reviewed the associated listed documents:

- Unit 1 train A Nuclear Service Water (RN) while train B was out of service for backwash strainer retubing (OP/0/A/6400/006C, Nuclear Service Water System, Tagout: 03-01929)
- 1B EDG with the 1A EDG out of service (OP/1/A/6350/002, Diesel Generator Operation)
- Unit 2 RN supplying Unit 1 while a Unit 1 RN pump out of service (Nuclear Service Water System, OP/0/A/6400/006 C, Enclosure 4.11, Operability Actions with One RN Pump and/or Its Associated EDG Inoperable With Both Units Entering An Action Statement)

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown.

a. Inspection Scope

The inspectors conducted a detailed walkdown/review of the alignment and condition of the Unit 2 auxiliary feedwater (CA) system. The inspectors utilized licensee procedures, as well as licensing and design documents, to verify proper system (i.e., pump, valve, and electrical) alignment. During the walkdown, the inspectors also verified that: valves and pumps did not exhibit leakage that would impact their function; major system and components were correctly labeled; hangers and supports were correctly installed and functional; and essential support systems were operational. In addition, pending design and equipment issues were reviewed to determine if the identified deficiencies significantly impacted the system's functions. Items included in this review were: the operator workaround list; the temporary modification list; CA system Health Reports; and outstanding maintenance work requests/work orders (WOs). A review of open Problem Investigation Problem reports (PIPs) was also performed to verify that the licensee had appropriately characterized and prioritized CA related equipment problems for resolution in their corrective action program. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection Walkdowns

a. Inspection Scope

The inspectors walked down accessible portions of the plant to assess the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The inspectors observed the fire protection suppression and detection equipment to determine whether any conditions or deficiencies existed which could impair the operability of that equipment. The inspectors selected the areas based on a review of the licensee's safe shutdown analysis, probabilistic risk assessment (PRA) based on sensitivity studies for fire related core damage accident sequences, and summary statements related to the licensee's 1992 Initial Plant Examination for External Events submittal to the NRC. The inspectors toured the following areas important to reactor safety:

- 1A Emergency Diesel Generator Room
- 1B Emergency Diesel Generator Room
- 1A Sequencer Area
- 1B Sequencer Area
- 1ETB Essential 4160V Switchgear Area
- Unit 1 Nuclear Service Water Pumphouse Structure
- Unit 2 Nuclear Service Water Pumphouse Structure
- Unit 1 Auxiliary Feedwater Pump Room

b. Findings

No findings of significance were identified.

.2 Fire Drill Observations

a. Inspection Scope

On August 13, 2003, the inspectors observed an announced shift fire drill simulating a fire in safety-related motor control center 2EMXB, which is located on the 560 foot elevation of the Auxiliary Building. The purpose of this annual inspection was to: monitor the fire brigade's use of protective gear and fire fighting equipment; to verify that fire fighting pre-plan procedures and appropriate fire fighting techniques were used; and to verify that the directions of the fire brigade leader were thorough, clear, and effective. The inspectors also attended the subsequent drill critique to assess whether it was appropriately critical, including discussions of drill observations and identifying any areas requiring corrective action. Documents reviewed in conjunction with this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed a simulator training scenario conducted on September 10, 2003, to assess the performance of licensed operators. The scenario, Active Simulator Exam ASE-09, involved a power range nuclear instrument failure, a diesel generator inoperability, and a loss of all alternating current power. The inspection focused on high-risk operator actions performed during implementation of the emergency operating procedures, emergency plan implementation and classification, and the incorporation of lessons learned from previous plant events. Through observations of the critique conducted by training instructors following the training session, the inspectors assessed whether appropriate feedback was provided to the licensed operators regarding identified weaknesses.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

.1 Routine Maintenance Effectiveness Inspection

a. Inspection Scope

The inspectors reviewed the licensee's effectiveness in performing routine maintenance activities. This review included an assessment of the licensee's practices pertaining to the identification, scope, and handling of degraded equipment conditions, as well as common cause failure evaluations and the resolution of historical equipment problems. For those systems, structures, and components (SSC) scoped in the maintenance rule per 10 CFR 50.65, the inspectors verified that reliability and unavailability were properly monitored, and that 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. Documents reviewed during these inspections are listed in the Attachment to this report. The inspectors conducted this inspection for the degraded equipment conditions associated with the three items listed below:

- Unit 2 C Reactor Coolant (NC) Flow Channel Indicating Higher Than Normal
- Failure of Unit 1, Loop A, Hot leg Temperature Detectors, A1 and A3
- 1A Residual Heat Removal (ND) Train Inoperable Due to Snubber Acid Exposure

b. Findings

No findings of significance were identified.

.2 Biennial Periodic Evaluation Inspection

a. Inspection Scope

The inspectors reviewed the licensee’s Maintenance Rule periodic assessment, “Maintenance Rule Periodic Assessment [Report] for Maintenance Rule Implementation - Catawba Nuclear Station, October 1, 2000 - April 1, 2002,” dated September 17, 2002, while on-site the week of July 21, 2003. The report was issued to satisfy paragraph (a)(3) of 10 CFR 50.65, and covered the period as indicated for two units. The inspection was to determine the effectiveness of the assessment and that it was issued in accordance with the time requirement of the Maintenance Rule (MR) and included evaluation of: balancing reliability and unavailability, (a)(1) activities, (a)(2) activities, and use of industry operating experience. To verify compliance with 10 CFR 50.65, the inspectors reviewed selected MR activities covered by the assessment period for the following maintenance rule systems: Control Complex Chiller, Component Cooling, Service Water, Containment Personnel Air Lock, Fuel Transfer Canal, Ice Condenser Refrigeration System. Specific procedures and documents reviewed are listed in the attachment to this report.

During the inspection, the inspectors reviewed selected plant work order data, the site guidance implementing procedure, discussed and reviewed relevant corrective action issues, reviewed generic operations event data, probabilistic risk reports, and discussed issues with system engineers. Operational event information was evaluated by the inspectors in its use in MR functions. The inspectors selected work orders, a MR assessment, and other corrective action documents of systems recently removed from 10 CFR 50.65 a(1) status and those in a(2) status for some period to assess the justification for their status. The documents were compared to the site’s MR program criteria, and the MR a(1) evaluations and rule related data bases. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the licensee’s assessments concerning the risk impact of removing from service those components associated with the six emergent and planned work items listed below. This review primarily focused on activities determined to be risk significant within the maintenance rule. The inspectors also assessed the adequacy of the licensee’s identification and resolution of problems associated with maintenance risk assessments and emergent work activities. The inspectors reviewed Nuclear System Directive 415, Operational Risk Management (Modes 1-3) per 10 CFR 50.65 (a)(4).

- Unit 1, B Train RN pump for strainer retubing
- 2A Charging (NV) Pump Maintenance
- Unit 1, Nuclear Instrument, N42 cross channel calibration while channel 1 temperature instruments were in a bypass condition

- 1B Residual Heat Removal Pump Replacement
- Unit 2, ECC Battery Charger Voltage Swings
- 2B Emergency Diesel Generator Battery Cell 39 and 40 Replacement

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope

On August 26, 2003, the inspectors observed maintenance and operator performance during the implementation of a procedure for bypassing Unit 1, Channel 1, Loop A hot leg temperature instrument to allow performance of surveillance testing on nuclear instrumentation channels (IP/1/A/3222/076 E, Procedure for Bypassing Channel 1 Trip Signals Related to Delta-T Outputs). This procedure was a first time use procedure developed to comply with Technical Specifications (TS) to place the channel in a bypass condition. The inspectors observed licensed operators' and maintenance technicians' use of procedures, control room briefings, and plant equipment manipulations throughout the implementation and restoration from the procedure.

On September 8, 2003, the inspectors observed operator performance during a reactor startup and approach to criticality. The inspectors observed the conduct of control room activities, procedure use and adherence, and plant equipment manipulations.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability determinations (or justifications for continued operation) to verify that the operability of systems important to safety were properly established, that the affected components or systems remained capable of performing their intended safety function, and that no unrecognized increase in plant or public risk occurred. Documents reviewed are listed in the Attachment to this report. Operability evaluations were reviewed for the issues listed below:

- 1A, 1B and 2A Emergency Diesel Generators operability prior to delivery valve inspection

- 1A and 1B NV Pumps with identified gas emitted at 1NV-860

- Refueling Water Storage Tank (FWST) Operational Temperature Limit

- 1B Residual Heat Removal Pump Head Curve for Recirculation to FWST

- 1B Containment Spray Heat (NS) Exchanger (HX) Heat Capacity Test

- Reactor Coolant Pressure Boundary Valve Leak Rate Testing

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed permanent plant modification CNCE-62130, Installation of ECCS Vent Valves and Lines On Unit 2, to verify the adequacy of the modification package, and to verify that the design change and subsequent post-modification testing ensured continued reliability and satisfactory performance of the affected system. In addition, the inspectors reviewed PT/2/A/4200/006B, ECCS Valve Lineup Verification, and performed field walkdowns of accessible portions of the completed modification to verify that the installed vent valves were installed at high points to support system venting evolutions.

b. Findings

No findings of significance were identified

1R20 Unit 1 Forced Outage Activities

a. Inspection Scope

The inspectors evaluated Unit 1 forced outage activities to ensure that the licensee considered risk in developing outage schedules; adhered to administrative risk reduction methodologies developed to control plant configuration; developed mitigation strategies for losses of key safety functions; and adhered to operating license and TS requirements that ensure defense-in-depth. The following specific areas were reviewed:

- Monitoring of Shutdown Activities - The inspectors reviewed OP/1/A/6100/002, Controlling Procedure For Unit Shutdown, during unit shutdown, and also reviewed PT/1/A/4600/017, Surveillance Requirements For Unit 1 Shutdown, to ensure cooldown rates while cooling down the reactor coolant system were in accordance with TS 3.4.3, NC System Pressure and Temperature (P/T) Limits.

- Outage Configuration Management - The inspectors assessed the licensee's management of configuration control and the risk associated with outage activities by reviewing the licensee's implementation of Site Directive 3.1.30, Unit Shutdown Configuration Control (Mode 4, 5, 6 or No Mode). This assessment included verification that the licensee maintained defense-in-depth commensurate with the key safety functions and applicable TS when risk significant equipment was removed from service. The inspectors also assessed whether configuration changes due to emergent work and unexpected conditions were controlled properly and if control room operators were cognizant of plant configuration.

- Reactivity Control - The inspectors reviewed reactivity control to verify that proper control was maintained in accordance with the TS and Site Directive 3.1.30, Unit Shutdown Configuration Control (Mode 4,5,6 or No Mode).

- Monitoring of Heatup and Startup Activities - The inspectors reviewed TS, license conditions, commitments, and administrative procedure prerequisites for mode changes to verify they were met for changing plant configurations. The inspectors performed a walkdown of primary containment prior to reactor startup to verify that debris had not been left which could affect performance of the containment sumps. The inspectors observed reactor startup, the approach to criticality, and portions of the power ascension. The inspectors reviewed PT/0/A/4150/19, 1/M, Approach to Criticality and PT/1/A/4600/016, Surveillance Requirements for Unit 1 Startup, Enclosure 13.3, NC System Heatup Surveillance Items.

b. Findings

Introduction: During the containment closeout walkdown conducted on August 31, the inspectors identified a substantial amount of debris in the Unit 1 containment sump. The identified debris in the containment sump is an unresolved item (URI) pending review of the licensee's past operability review.

Description: The Unit 1 forced outage duration was approximately 3 days when the inspectors discovered the debris in the sump. The licensee had completed hot leg temperature RTD replacements and had also completed their containment closeout inspection prior to going to Mode 4 and initiating a plant heatup. The debris in the containment sump was removed by the licensee and determined to be a cork material. The cork material was a remnant of what had been removed from between the interface of the cold leg accumulator floor and the containment lining during the last Unit 1 refueling outage. Apparently, some of that material became deposited in the sump and remained there during plant operations for approximately the previous 15 months. No work had been performed near the containment sump during the forced outage.

Analysis: The significance of finding debris in the containment sump was that interfacing mitigating systems (i.e., residual heat removal, high head injection, safety injection, and containment spray systems) could be adversely affected. In addition, flow blockage through the core due to fuel bundle debris screen blockage could occur. Debris in the containment sump could effect the reactor safety mitigating system cornerstone objective to ensure the availability and capability of systems. Because the licensee removed the debris prior to entry into mode 4 there was no immediate safety concern for plant operations. Pending review of the licensee's past operability and reportability evaluation, and determination of the safety significance of the debris that was located in the containment sump, this issue is identified as URI 05000413/2003004-01: Containment Walkdown Identified Debris In Containment Sump. The licensee has entered this issue into their corrective action program as PIP C-03-4815.

Enforcement: The enforcement aspects of this issue will be addressed during the follow up inspection associated with the URI review.

1RST Post-Maintenance and Surveillance Testing (pilot)

a. Inspection Scope

The inspectors observed and/or reviewed the surveillance tests and post-maintenance tests listed below to verify that TS Surveillance Requirements and/or Selected Licensee Commitment (SLC) requirements were properly complied with, and that test acceptance criteria were properly specified. The inspectors verified that proper test conditions were established as specified in the procedures, that no equipment preconditioning activities occurred, and that acceptance criteria had been met. Additionally, the inspectors also verified that equipment was properly returned to service and that proper testing was specified and conducted to ensure that the equipment could perform its intended safety function following maintenance or as part of surveillance testing. The documents reviewed are listed in the Attachment to this report.

Surveillance Tests:

- PT/2/A/4350/002B, Diesel Generator 2B Operability Test
- IP/1/A/3200/008 A, Train A Reactor Trip Breaker Trip Actuating Device Functional and Operational Test
- IP/ 1/A/3200/002 A, Solid State Protection System (SSPS) Train A Periodic Testing
- IP/1/A/3240/0141, Power Range N42 Incore-Excore Cross Calibration

In-Service Test:

- PT/0/A/4400/022A, Nuclear Service Water Pump Train A Performance Test, 2A RN Pump

Post-Maintenance Tests (associated with):

- Unit 2 C NC Flow Channel 2 Reading Higher Than Normal
- 2A NV Pump Breaker PM
- 1B ND Pump Replacement
- Unit 1, Loop A, Hot Leg RTD Replacements

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a Inspection Scope

The inspectors observed and evaluated the licensee's performance during an exercise conducted on August 6 and a drill conducted on September 10. The inspectors reviewed the exercise scenario on August 6 to determine if elements of the licensee's Radiological Emergency Plan would be sufficiently challenged. Licensee activities inspected during the exercise included those occurring in the Control Room Simulator, and in the Technical Support Center. The NRC's assessment focused on the timeliness and location of classification, the notification and protective action recommendations (PAR) developmental activities, and the licensee's expectations of response. The performance of the emergency response organization was evaluated against applicable licensee procedures and regulatory requirements. The inspectors attended the post-exercise critique to evaluate the licensee's self-assessment process for identifying deficiencies relating to failures in classification and notification, as well as PAR development activities. The inspectors observed the active simulator exam drill on September 10 to determine whether the drill was of the appropriate scope to be included in the emergency preparedness performance indicator statistics. A review of the checklist and form (Drill/Evaluated ASE Classification, PAR and Notification Grading Form) for emergency planning performance indicator data was completed. The inspectors assessed the drill for weaknesses and deficiencies in performance of classification and notification requirements.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period from July 2002 through June 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2, were used to verify the basis in reporting for each data element.

- Safety System Unavailability - Auxiliary Feedwater System, Unit 1
- Safety System Unavailability - Auxiliary Feedwater System, Unit 2

The inspectors reviewed a selection of Licensee Event Reports (LERs), portions of Unit 1 and Unit 2 operator log entries, PIP descriptions, monthly operating reports, and PI data sheets to verify that the licensee had adequately identified the number of unavailability hours and safety system functional failures. These numbers were compared to the numbers reported for the PIs.

b. Findings

No findings of significance were identified.

.2 Barrier Integrity Cornerstone

a. Inspection Scope

The inspectors sampled licensee submittals for the PI listed below for the period from June 2002 through July 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, Revision 2, were used to verify the basis in reporting for each data element.

- Reactor Coolant System Identified Leakage, Unit 1
- Reactor Coolant System Identified Leakage, Unit 2

The inspectors reviewed a selection of Unit 1 and Unit 2 operator log entries compiled from the reactor coolant leakage calculation performed on a daily basis. The inspectors verified that the reported reactor coolant leakage performance indicator data was conservative and accurate.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors selected PIP C-03-00541 for detailed review. The PIP was associated with the high-high steam generator level caused by maintenance activities on the Digital Feedwater Control System, which resulted in a main turbine and subsequent reactor trip. The PIP was reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the PIP against the requirements of the licensee's corrective action program, Nuclear System Directive (NSD) 208, Problem Investigation Process, and 10 CFR 50 Appendix B. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings and Observations

No findings of significance were identified.

4OA3 Event Followup

.1 Unit 1 Over-Temperature Delta-Temperature Reactor Trip

a. Inspection Scope

On August 29, 2003, the inspectors responded to the Unit 1 control room following a reactor trip that was initiated by an over-temperature delta-temperature trip. Prior to the reactor trip, the channel 1 hot leg temperature instrument was in a trip condition due to failed temperature detectors in that channel. A pressurizer pressure instrument power supply randomly failed, which provided a second channel trip logic to be met in the reactor protection system, and an appropriate reactor trip occurred. The inspectors assessed plant status and parameters, including mitigating system performance during and following the unit trip. Plant process computer traces, operator statements, and the licensee's trip investigation report were also reviewed.

b. Findings

No findings of significance were identified.

.2 Notice of Enforcement Discretion (NOED) to Repair Refueling Water Storage Tank Level Instruments

On August 16, 2003, the NRC granted a Unit 2 NOED related to enforcing compliance with the requirements of TS 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation. The details of the failure and the NOED written request are documented in a Letter dated August 18, 2003, from the licensee to the NRC. An apparent lightning strike caused channels 1 and 3 of the Unit 2 Refueling Water Storage Tank (FWST) level instrumentation to be declared inoperable as a result of their failing to a high indication. Both of the channels were placed in a bypass condition and Unit 2 entered TS 3.0.3. An additional 48 hours was requested to the Limiting Conditions for Operations (LCO) time to repair one of the two failed channels. The inspectors reviewed the applicable TS requirements, assessed the impact of the inoperable FWST level instruments, and monitored compliance for granting of the NOED. The licensee repaired one channel of FWST level instrumentation prior to expiration of the required time for Unit 2 to be in Mode 3, as required by TS 3.0.3; therefore, the NOED was not actually utilized.

.3 (Closed) LER 50-413/03-003-01: Failure Of 4160 Volt Breaker Renders 1B1 Component Cooling Water (KC) Pump Inoperable For Longer Than Technical Specifications Allow

a. Inspection Scope

The inspectors reviewed the LER and associated root cause investigation to verify that the cause was identified, that both short-term and long-term corrective actions were reasonable, and whether a licensee performance deficiency was associated with the cause of the breaker failure. The LER documented that the failure of the 4160 volt breaker for the 1B1 KC pump occurred on April 4, 2003, when the 1B1 KC pump failed to start upon demand. The licensee replaced the breaker subsequent to the failure and declared the pump operable on April 4, 2003, following the satisfactory completion of functional testing. The licensee determined that the breaker failure occurred on April 1, 2003, when the pump was secured following system testing. The licensee determined the cause of the breaker failure to have been an intermittent interference problem internal to the 1B1 KC pump breaker control device.

b. Findings

Introduction: A violation of TS was identified for equipment being inoperable in excess of the allowed times. Enforcement discretion was exercised for this violation. This issue was determined not to be a finding because a performance deficiency was not identified.

Description: The licensee determined the cause of the breaker failure was an intermittent interference problem internal to the control device located in the 1B1 KC pump breaker. Technical Specification 3.7.7, Component Cooling Water System, requires that two trains of KC remain operable in Mode 1, and in the event that one train becomes inoperable, the inoperable train be restored to operable within 72 hours or the plant be placed in Mode 3. The licensee determined that the 1B1 KC pump was inoperable for 88 hours with the plant operating in Mode 1.

Analysis: The inspectors determined that a violation of TS 3.7.7 occurred since the pump was inoperable in excess of the TS allowed time for inoperability of 72 hours. The inspectors determined that this violation was greater than minor because the failure of the 1B1 KC pump to start on demand was associated with the reactor safety cornerstone objective for mitigating systems to ensure availability and reliability of mitigating equipment.

The inspectors determined that the breaker failure was not a performance deficiency because the cause of the breaker failure was not reasonably within the licensee’s ability to foresee and correct to prevent the failure. Because a performance deficiency was not associated with this issue, it was not subject to evaluation under the Significance Determination Process (SDP). However, to understand the significance of the TS violation, a risk assessment was performed by the Region II Senior Reactor Analyst. This assessment concluded that the violation had very low safety significance because the opposite and redundant train of KC remained available to perform its safety function during the period of time that the 1B1 KC pump was inoperable. This issue was captured in the licensee’s corrective action program as PIP C–03-2273.

Enforcement: The NRC concluded that a violation of TS occurred; however, the violation was not attributable to an equipment failure that was avoidable by reasonable licensee quality assurance measures or management controls. Because the applicable criteria specified in the NRC’s Enforcement Policy was satisfied, the NRC is exercising enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and is refraining from issuing enforcement action for this violation.

.4 (Closed) LER 50-413/03-001-00: High Steam Generator Level Turbine Trip Causes Reactor Trip and Automatic Start of Motor Driven Auxiliary Feedwater System Pumps

On February 4, 2003, the Unit 1 reactor tripped from 100 percent RTP due a main turbine trip caused by a high-high level in the B steam generator resulting from a hydraulic transient in the common pressure sensing line on the feedwater header during maintenance activities. The high-high steam generator level also resulted in the trip of the main feedwater pumps, which caused the automatic start of the motor-driven auxiliary feedwater pumps. The event was captured in the licensee's corrective action program as PIP C-03-0541. No findings of significance were identified during the inspectors review of the LER and associated root cause failure analysis report.

4OA5 Other Activities

.1 (Closed) URI 50-413,414/02-08-02: Effect of RCP Thermal Barrier Rupture on Motor Operated Valve (MOV) Closure and Containment Isolation

Introduction: A Green non-cited violation (NCV) was identified for failure to comply with 10 CFR 50 Appendix B, Criterion 3, Design Control, due to inadequate design measures, specifically, the licensee failed to assure adequate relief valve sizing to prevent exceeding the design pressure of the component cooling water (KC) piping in the event of a reactor coolant pump (RCP) thermal barrier rupture.

Description: During the safety system design and performance capability inspection completed on January 31, 2003, the inspectors identified a finding involving the capability of the KC RCP thermal barrier outlet relief valve to provide over pressure protection for the KC piping. The finding was unresolved pending further technical evaluation of the condition by the licensee and a significance determination by the NRC. The finding was unresolved to address three issues: adequacy of the sizing of the relief valves, capability of the KC containment isolation valve to meet their isolation function, and the basis for not considering the RCP thermal barrier rupture as a design basis event.

Analysis: As discussed in NRC Inspection Report No. 50-413,414/02-08, the KC relief valves were inadequately sized to prevent exceeding the American Society of Mechanical Engineers (ASME) Code pressure and temperature requirements for the KC piping in the event of a RCP thermal barrier rupture. This was due to a design control deficiency and was documented in the licensee's technical evaluation in Problem Investigation Report (PIP) C-03-01028. This condition was evaluated by the licensee in Operability Assessment: Component Cooling Water System, RCP Thermal Barrier Rupture.

The closure capability of the KC containment isolation valves was adversely impacted by the potential high pressure condition resulting from inadequately sized relief valves. Due to the high pressure condition of the KC line anticipated from a RCP thermal barrier rupture, the containment isolation valves would experience a differential pressure condition that exceeded their established motor operator set point. The licensee's operability assessment determined that although the containment isolation valves would not close automatically against the higher differential pressure, there were actions that could be taken to achieve containment isolation. For example, the RCP thermal barrier isolation valve could be manually closed from the control room. The inspector noted this valve motor operator is designed and set to close against the differential pressure resulting from a RCP thermal barrier rupture. This would terminate the high pressure condition and allow the containment isolation valve to close at the motor operator setpoint. The inspector noted that there was sufficient control room indication to alert the operators to a RCP thermal barrier rupture condition should the thermal barrier isolation valve not automatically close on a sensed high flow condition. Additionally, operators could be dispatched to manually close the containment isolation valves locally. The inspector concluded there was reasonable expectation of containment isolation based on operator training, and available plant indication and controls. Procedure AP/1(2)/A/5500/010, Reactor Coolant System Leak, Rev. 41, provides guidance for operator actions to address the RCP thermal barrier rupture.

The RCP thermal barrier event was not considered a design basis event because the expected leak rate was within the capacity of the high head safety injection pump. The inspector noted that the thermal barrier rupture event calculated leak rate at Catawba was 156 gpm. The capacity of one high head safety injection pump is 180 gpm. The licensee's technical analysis additionally evaluated the off site dose assuming no containment isolation on the thermal barrier rupture event and determined that the off site dose was bounded by the Steam Generator Tube Rupture event which is a design base event.

This finding represented a performance deficiency because it involved the licensee's failure to assure the design adequacy of the KC relief valve to protect the piping from exceeding design limits in the event of a reactor coolant pump thermal barrier leak. This finding is more than minor because it affects the Reactor Safety Cornerstone, Barrier Integrity attribute of design control and affects the associated objective. The inadequately sized relief valve represents a potential open path way in the physical integrity of the reactor containment. The NRC performed a phase three significance determination screening analysis and concluded the finding is of very low safety significance (Green). The phase three screening analysis was performed to determine the large early release frequency contribution of a thermal barrier heat exchanger failure using the core damage frequency as an upper limit. A conservative estimate of frequency of heat exchanger failure was made based on industry operating experience. The results were multiplied by the conditional core damage probability for a reactor trip, since the event would require a shutdown, to establish a conservative upper bound for the change in core damage probability. The result was less than 1E-7; therefore, the finding is Green.

Enforcement: 10 CFR 50, Appendix B, Criterion III, "Design Control" states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions." The failure of the licensee's design control program to assure the adequacy of relief valve sizing to prevent exceeding the design ASME Code pressure and temperature limits of the KC piping in the event of a RCP thermal barrier rupture is being treated as an NCV, consistent with Section VI.A.1 of the enforcement policy and is identified as NCV 05000413,414/2003004-02: Effect of RCP Thermal Barrier Rupture on MOV Closure and Containment Isolation. As noted earlier, this issue was documented in the licensee's corrective action program as PIP C-03-01028.

.2 (Closed) URI 50-413,414/01-08-01: Changes to the Approved Fire Protection Program for Areas Designated as Alternative Shutdown.

The triennial fire protection inspection of 2001 observed that for a number of fire areas there was a difference between what was described in the July 1983 Fire Protection

Review sent to the NRC and the current design basis specification for post-fire safe shutdown. (NRC Safety Evaluation Reports issued at the time of plant licensing were based on the July 1983 Fire Protection Review.) These differences involved the credited method for post-fire safe shutdown. The areas in question were:

- Fire Area 20, a Unit 1 electrical penetration room
- Fire Areas 23 and 24, Units 1 and 2 fuel storage rooms
- Fire Areas 25, 26, 27 and 28, the four diesel generator rooms (have fixed suppression)
- Fire Areas 41, 42, 43 and 44, tunnel shaped rooms containing only the emergency diesel generator sequencer panels
- Fire Areas 29 and 30, Train A and Train B, respectively, nuclear service water pump rooms, which are in the service water pump building
- Fire Areas 38 and 47, heating and ventilating equipment spaces for the fuel storage rooms
- Fire Areas 50 and 51, the Unit 1 and Unit 2 outer doghouses, respectively, which contained feedwater and main steam piping and valves

For the areas in question, the current program specified that the safe shutdown system was the only credited method for post-fire safe shutdown. However, the July 1983 document stated that either control room shutdown or control room and safe shutdown would be available. The issue also involved the fact that, with four exceptions, these areas did not have fixed suppression.

To resolve this issue, the inspector reviewed the Catawba Fire Protection Review, dated July 1983, transmitted to the NRC by letter dated November 4, 1983. In addition, Section 9.5.1, Fire Protection Program, in the NRC Safety Evaluation Report and Supplements 1 through 5, was reviewed. The inspectors also reviewed NRC Inspection Report 50-414/85-66 for an inspection conducted December 9 - 13, 1985, which was shortly before Unit 2 was issued an operating license.

The conclusion from these reviews was that the licensee's fire protection program for the above mentioned fire areas was acceptable. The program was acceptable because the licensing basis for Catawba is NUREG 0800, Section 9.5.1, which did not require fixed suppression when dedicated shutdown is utilized. The licensee effectively re-designated a number of areas from control room shutdown to dedicated shutdown after the Safety Evaluation Reports were issued but before the startup of Unit 2. The changes were not submitted to the NRC for review and approval, but since the changes stayed within the licensing basis they would not be considered changes adversely affecting the ability to achieve and maintain safe shutdown in the event of a fire. Also, NRC Inspection Report 50-414/85-66 shows that the NRC was aware of the changes, and found them acceptable.

Although not part of the Unresolved Item, NRC Inspection Report 50-413,414/01-08 mentioned that the team noted that the licensee had not performed evaluations (in accordance with the guidance in NRC Generic Letter (GL) 86-10, Implementation of Fire Protection Requirements, dated April 24, 1986) for the 19 fire areas not approved in SSER 4 which credited alternative shutdown. A Region II based inspector addressed this issue during a site visit July 21 through 24, 2003. The inspector found that the licensee's analysis of electric cables was contained in the Design Basis Document for

Post-Fire Safe Shutdown, CNS-1435.00-00-0002. Appendix E was a listing of all the required components. Appendix F was a table titled "Analysis Results by Fire Area." This table was essentially a listing of potential problem cables in each fire area together with the resolution for the potential problem. If there were no potential problem cables in a particular fire area, no cables would be listed in Appendix F. The inspector reviewed the Chemical and Volume Control System and the Reactor Coolant System in terms of analysis of cables in the fire areas that were the subject of this Unresolved Item. The inspector did not identify any potential problem cables. During this review, it became apparent that the areas in question were not in the routing paths for cables running between the control room, the motor control centers and the end devices. Accordingly this URI is closed.

.3 Institute of Nuclear Power Operations (INPO) Report Review

The inspectors reviewed the final report issued by INPO for the evaluation that was conducted at the Catawba facility during October 2002. The inspectors did not note any safety issues in the INPO report that either warranted further NRC followup or that had not already been addressed by the NRC.

4OA6 Meetings

Exit Meeting Summary

On October 9, 2003, the resident inspectors presented the inspection results to Mr. D. Jamil, Site Vice President, and other members of licensee management, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

E. Beadle, Emergency Planning Manager
S. Brown, Operations Superintendent
W. Byers, Security Manager
M. Carwile, Rotating Equipment Engineering Manager
T. Daniels, Emergency Planning/Fire Protection
J. Foster, Radiation Protection Manager
G. Gilbert, Regulatory Compliance Manager
M. Glover, Station Manager
P. Grobusky, Human Resources Manager
P. Herran, Engineering Manager
D. Jamil, Catawba Site Vice President
J. Kammer, Mechanical, Civil Engineering Manager
P. Kowalewski, Nuclear General Office Maintenance Rule Engineer
A. Lindsay, Work Control Superintendent
P. McIntyre, Safety Review Group Manager
F. Smith, Chemistry Manager
G. Strickland, Regulatory Compliance Specialist
R. Sweigart, Safety Assurance Manager
C. Trezise, Maintenance Superintendent
E. Wagner, CA System Engineer

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed	Item Type	Description
05000413,414/2003004-02	NCV	Effect of RCP Thermal Barrier Rupture on MOV Closure and Containment Isolation (Section 4OA5.1)

Opened		
05000413/2003004-01	URI	Containment Walkdown Identified Debris In Containment Sump (Section 1R20)

Closed

50-413/03-003-01	LER	Failure Of 4160 Volt Breaker Renders 1B1 Water (KC) Pump Inoperable For Longer Than Technical Specifications Allow (Section 4AO3.3)
50-413/03-001-00	LER	High Steam Generator Level Turbine Trip Causes Reactor Trip and Automatic Start of Motor Driven Auxiliary Feedwater System Pumps (Section 4OA3.4)
50-413,414/02-08-02	URI	Effect of RCP Thermal Barrier Rupture on MOV Closure and Containment Isolation (Section 4OA5.1)
50-413,414/01-08-01	URI	Changes to the Approved Fire Protection Program for Areas Designated as Alternative Shutdown (Section 4OA5.2)

DOCUMENTS REVIEWED

(Section 1RO4)

OP/2/A/6250/002, Auxiliary Feedwater System, Revision 113
Drawing CN-2592-(1.0, 1.1, and 1.2), Flow Diagram of Auxiliary Feedwater System
Auxiliary Feedwater Health Reports (2002Q4 and 2003T1) PT/2/A/4250/006E, CA Valve Verification, Revision 9
TS SR 3.7.5.1, Auxiliary Feedwater System 31 Day Flow Path Valve Alignment Verification
Design Basis Specification CNS-1592.CA-00-0001, Auxiliary Feedwater System, Revision 28 CNS Operator Workaround List
UFSAR Section 10.4.9, Auxiliary Feedwater System
UFSAR Change Summary Form 02-041, Inclusion of Auxiliary Feedwater Pump Turbine Steam Supply Piping Heat Trace
Selected Licensee Commitment Change Request 02-41, Addition of Section 16.7-13 for Auxiliary Feedwater Pump Turbine Steam Supply Piping Temperature Monitoring Instrumentation and Heat Trace System
PIP C-01-05586, Low SA Pipe Temperatures
PIP C-01-00967, Applicability of CA-2, CA-4, and CA-6 for Meeting TS SR 3.7.5.1
PIP C-02-05978, 2A CFPT Failed to Trip
PIP C-02-06119, SA Pipe Design Temperature Exceeded
PIP C-03-04473, Unit 2 Train B RN to CA Suction Piping Flow Measurement Test Not Performed Inside Established Frequency
PIP C-99-03979, 2CA-33 Stuck Close Preventing Tempering Flow From Being Established to the 2A Steam Generator

PIP C-99-03621, Single Failure of Auctioneered Vital I&C Panelboard May Result in Inability to Stop Flow From CATDP to Two Steam Generators

(Section 1RO5.2)

Fire Drill Scenario No. 03-1, dated 8/13/03
Nuclear System Directive NSD-112, Appendix B, Fire Drill Critique, dated 8/13/03
Response Procedure RP/0/B/5000/029, Fire Brigade Response, Revision 3
Operations Management Procedure 2-22, Attachment 12, Shift Assignments - E Shift, dated 8/13/03
Pre-Fire Plan, Section 1.5 (Fire Area 11, Auxiliary Building 560 Level)

(Section 1R12.1)

Unit 2 C Reactor Coolant (NC) Flow Channel Indicating Higher Than Normal

IP/2/A/3222/015G, Reactor Coolant Flow, Loop C, Protection Ch. 2, 2NCFT5070 (FT-435) Calibration
IP/0/A/3817/013A, Calibration and Maintenance Procedure for Rosemount Model 1153 and 1154 Differential Pressure Transmitters
PIPs C-03-03141, C-03-04124
Work Order 98600247, C NC Loop Channel Higher Than Normal Indication
Work Order 98610503, Channel 2 NC Flow On C Loop Is Inaccurate

Failure of Unit 1, Loop A, Hot leg Temperature Detectors, A1 and A3

Maintenance Rule: SSC Summary Sheets - Reactor Coolant and NSSS Process Instrumentation & Control
PIP C-03-04446, CHAR testing of loop A T-hot RTDs appears to have caused the failure of RTD A1 & A3

1A Residual Heat Removal (ND) Train Inoperable Due to Snubber Acid Exposure

PIP C-03-3056, 1A ND train inoperable and unplanned entry into TS (C1-03-01311) due to damaged snubbers.
Maintenance Rule: SSC Summary Sheets - Residual Heat Removal and Low Head Safety Injection

(Section 1R12.2)

Problem Investigation Process reports (PIP)

C-00-04489, Revision to CNS PRA Maintenance Rule Calculation is needed
C-99-03172, NF system active Air Operated Valves NCI
C-02-02295, Personnel on “on station” as Required by Procedure
C-00-01079, Unplanned Entry into Tech Spec 3.6.15 for FW Drains (C-01-01124 relates)
C-02-05473, PIP documentation of a Regular Maintenance Rule Assessment
C-02-02568, Tube Support Damage on 115 NS HX 1B Tubes
C-01-00083, 2KC-C40 (B train miniflow) did not Open as Expected
C-01-02802, Several Valves have been identified as High Cycle Components

C-00-04194, Maintenance Rule Application continues to input incorrect Dates
C-98-04282, RN System for Unit 1 Classified Maintenance [MNT] Rule "A1"
C-00-03084, EDG is being Classified as MNT Rule "A1"
C-01-01426, CF System is now Classified MNT Rule "A1" (Unit 1)
C-01-01701, Unit 2 KC System is being Classified MNT Rule "A1"
C-01-05890, Power supplies are MNT Rule "A1"
C-02-01172, Starters are MNT Rule "A1"
C-02-06128, Containment Closure Supersystem is Classified as "A1"

Administrative Procedures

Nuclear System Directive 301, Requirements for the Maintenance Rule, Rev. 7
Engineering Directives Manual (EDM) 210, Engineering Responsibilities for the Maintenance Rule, Rev. 15

Miscellaneous

CNC-1535.00-00-0008, PRA Risk Significant SSCs for the Maintenance Rule, Rev. 3
CNM 2201.06-0009.001, NS HX 2B Sleeving Documentation by FramatoneANP, Rev. 0
PIP C-01-02703, Period Assessment of Maintenance Rule (April 1, 1999 through October 1, 2000)
PIP C-01-00280, Periodic Assessment of Maintenance Rule (September 24, 2001 - October 4, 2001)

(Section 1R15)

1A, 1B and 2A Emergency Diesel Generators operability prior to delivery valve inspection

PIP C-03-3589
Work Order 98606096, Inspection of 1A EDG Delivery Valves
Work Order 98605352, Inspection of 1B EDG Delivery Valves
Work Order 98606156, Inspection of 2A EDG Delivery Valves

1A and 1B NV Pumps with identified gas emitted at 1NV-860

PIP C-03-4174
PT/1/A/4200/006B, Emergency Core Cooling System (ECCS) Valve Lineup Verification

Refueling Water Storage Tank (FWST) Operational Temperature Limit

PIP C-03-04711, During a review of the FWST temperature uncertainty calculation, it was noticed that the calculated error could be non-conservative with respect to safety analysis calculations

1B Residual Heat Removal Pump Head Curve for Recirculation to FWST

ND Pump 1B Head Curve, TT/1/A/9300/044
PIP C-03-4916

<u>1B Containment Spray Heat (NS) Exchanger (HX) Heat Capacity Test</u>

PIP C-03-4658, 1B NSHX Heat Capacity Test Invalid

<u>Reactor Coolant Pressure Boundary Valve Leak Rate Testing</u>

PIP C-03-4965
PT/1/A/4200/001 N, Reactor Coolant Pressure Boundary Valve Leak Rate Test

(Section 1RST)

<u>Unit 2 C NC Flow Channel 2 Reading Higher Than Normal</u>

IP/2/A/3222/015G, Reactor Coolant Flow, Loop C, Protection Ch.2, 2NCFT5070 (FT- 435) Calibration
IP/0/A/3817/013A, Calibration and Maintenance Procedure for Rosemount Model1153 and 1154 Differential Pressure Transmitters
PIPs C-03-03141, C-03-04124
Work Order 98610503, Channel 2 NC Flow On C Loop Is Inaccurate

<u>2A NV Pump Breaker PM</u>

PT/2/A/4200/007A, Centrifugal Charging Pump 2A Test, Breaker Current Transformer Tests
Work Order 98605697, 2A NV Pump Breaker Testing

<u>1B ND Pump Replacement</u>

PT/1/A/4200/010 B, Residual Heat Removal Bump 1B performance Test
TT/1/A/9300/044, ND Pump 1 B Head Curve

<u>Unit 1, Loop A, Hot Leg RTD Replacements</u>

Work Order - CE72661- Replace U1 Loop A T-Hot Resistance Temperature Detectors (RTDs)
IP/1/A/3222/076 E, Procedure for Bypassing Channel I, Trip Signals Related to Delta-TOutputs

(Section 4OA2)

Root Cause Failure Analysis Report, “Unanticipated Feedwater Transient and Unit 1 Reactor Trip”
LER 50-413/03-001; “High Steam Generator Level Turbine Trip Causes Reactor Trip and Automatic Start of Motor Driven Auxiliary Feedwater System Pumps”
Annunciator Response Procedures OP/1/B/6100/010E and OP/2/B/6100/010E , tile D/5 NSD 705, “Instructions for the Verification and Validation of Technical Procedures”
Operations Communications dated 02/12/03 covering the expected plant response to the DFCS transferring from automatic to manual following a loss of the feed header pressure input

LIST OF ACRONYMS USED

ASME	-	American Society of Mechanical Engineers
CA	-	Auxiliary Feedwater
CFR	-	Code of Federal Regulations
CNS	-	Catawba Nuclear Station
DFCS	-	Digital Feedwater Control System
ECCS	-	Emergency Core Cooling System
EDG	-	Emergency Diesel Generator
FWST	-	Refueling Water Storage Tank
IMC	-	Inspection Manual Chapter
INPO	-	Institute of Nuclear Power Operations
KC	-	Component Cooling Water
LCO	-	Limiting Condition for Operations
LER	-	Licensee Event Report
MOV	-	Motor Operated Valve
MR	-	Maintenance Rule
NC	-	Reactor Coolant
NCV	-	Non-Cited Violation
ND	-	Residual Heat Removal
NEI	-	Nuclear Energy Institute
NOED	-	Notice of Enforcement Discretion
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
NSD	-	Nuclear System Directive
NSHX	-	Containment Spray Heat Exchanger
NSSS	-	Nuclear Steam Supply System
NV	-	Charging/Volume Control
PI	-	Performance Indicator
PIP	-	Problem Investigation Process (report)
PM	-	Preventive Maintenance
PRA	-	Probabilistic Risk Assessment
PT	-	Periodic Test
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RN	-	Nuclear Service Water
RTD	-	Resistance Temperature Detector
RTP	-	Rated Thermal Power
SDP	-	Significance Determination Process
SLC	-	Selected Licensee Commitment
SSC	-	Systems, Structures, and Components
TS	-	Technical Specification
UFSAR	-	Updated Final Safety Evaluation Report
URI	-	Unresolved Item
WO	-	Work Order