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

April 22, 2005

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

SUBJECT: CATAWBA NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000413/2005002 AND 05000414/2005002

Dear Mr. Jamil:

On March 31, 2005, 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 April 8 and 18, 2005, 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 three NRC-identified findings of very low safety significance (Green) which were determined to be violations of NRC requirements. However, because of their very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. In addition, the NRC identified a Severity Level IV violation of 10 CFR 50.59 for a failure to include a written evaluation which provided adequate bases for the determination that a change to the facility did not require a license amendment. It was determined that this violation should also be non-cited in accordance with Section VI.A of the NRC's Enforcement Policy. If you contest the NCVs 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 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system

(ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael Ernstes, 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 05000413/2005002 and 05000414/2005002
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:	05000413/2005002, 05000414/2005002
Licensee:	Duke Energy Corporation
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	4800 Concord Road York, SC 29745
Dates:	January 1, 2005 - March 31, 2005
Inspectors:	E. Guthrie, Senior Resident Inspector A. Sabisch, Resident Inspector Lee Miller, Senior Emergency Preparedness Inspector (Sections 1EP2, 1EP3, 1EP4, 1EP5, and 4OA1) Ramon Cortes, Reactor Inspector (Section 1R12.2)
Approved by:	Michael E. Ernstes, Chief Reactor Projects Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000413/2005002, IR 05000414/2005002; 1/1/2005 - 3/31/2005; Catawba Nuclear Station, Units 1 and 2; Fire Protection, Operability Evaluations, and Problem Identification and Resolution.

The report covered a three month period of inspection by two resident inspectors and announced regional-based inspections by a senior emergency preparedness inspector and a reactor inspector. Four non-cited violations (NCVs) were identified; three Green and one Severity Level (SL) IV. 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: Initiating Events

- Green. The inspectors identified a non-cited violation of Facility Operating Licenses NPF-35 (Unit 1) and NPF-52 (Unit 2), Condition 2.C.5, for the failure to implement the provisions of the approved fire protection program (Branch Technical Position CMEB 9.5-1) set forth in the Updated Final Safety Analysis Report (UFSAR) regarding fire brigade training and drills. Specifically, during the fire drill on February 10, 2005, the drill evaluator did not observe and assess the performance of the three teams attacking the simulated hydrogen fire on the Unit 2 main generator or operators in the main control room. As a result, some fire brigade member performance weaknesses were not noted during the drill, discussed during the post-drill critique or subsequently noted for development of appropriate corrective actions. The licensee recognized the drill team deficiency and implemented a change that required adequate team evaluators for future drills.

  This finding was determined to be greater than minor because it involved the degradation of a plant fire protection feature and has a credible impact on safety since fire brigade performance deficiencies may prevent a fire from being extinguished or allow a fire to propagate leading to a more significant event. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Fire Protection Significance Determination Process because the fire brigade is only a single element of the defense-in-depth fire protection strategy and the noted deficiencies produced a minimal impact on the fire fighting capabilities of the fire brigade. This finding involved the cross-cutting aspect of Human Performance, since the single evaluator did not identify all of the drill deficiencies that occurred during the drill. (Section 1R05.2)

- Green. The inspectors identified a non-cited violation of Technical Specification 5.4.1.a, Written Procedures, because the licensee failed to establish and maintain an adequate surveillance procedure for containment atmosphere radioactivity monitor surveillance requirement (SR) 3.4.15.2 and SR 3.4.15.4, in that the associated alarm function was not set or tested to alarm at a value equivalent to 1 gallon per minute in one hour for a realistic current reactor coolant activity level.

The finding was determined to be greater than minor because the containment gaseous and particulate channel radiation monitors were not capable of performing the design bases function for an extended period of time. Additionally, the operability of the reactor coolant system (RCS) leakage detection instrumentation alarming functions was not verified for an extended period of time. The inoperability resulted in potential impact on reactor safety and adversely affected the availability and reliability of the barrier integrity equipment performance attribute of the initiating events cornerstone. The finding was determined to be of very low safety significance because other methods of reactor coolant system leak detection were available to the licensee and no actual leakage above 1gpm was indicated through the RCS water balance surveillance. The unavailability of the gaseous and particulate channel leak detection functions and the RCS leakage detection instrumentation alarm indications did not contribute to an increase in core damage sequences when evaluated using the significance determination phase 1 worksheets. This finding involved the cross-cutting aspect of problem identification and resolution. The licensee had evaluated the operability of the radiation monitors via the corrective action program and incorrectly determined that the radiation monitors were operable. (Section 1R15b.(2))

Cornerstone: Mitigating Systems

- SL IV. The inspectors identified a non-cited violation for making a change to the facility (implemented as a change to the UFSAR in 1995) that involved an Unreviewed Safety Question (USQ), for which no written evaluation provided an adequate bases for the determination that the change did not require a license amendment pursuant to 10 CFR 50.90. Specifically, the UFSAR change reflected an increased length of time for incore instrumentation room sump instrumentation, as well as gaseous and particulate radiation monitors, to detect a 1 gpm leak. This increased the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety evaluation report for the reactor coolant system loss of coolant accident (LOCA) leak rate predictions, because the ability to detect a 1 gpm leak within one hour was relied on and credited in the leak-before-break design analysis. The significance of the violation was evaluated under the 10 CFR 50.59 Rule that was in effect at the time of the change, as well as the current 10 CFR 50.59 Rule. The current 10 CFR50.59 Rule requires, in part that "records must include a written evaluation which provides the bases for the determination that the change does not require a license amendment". This information (i.e., the ability to detect a 1gpm leak within one hour) was relied on in part, by NRC for approval of the leak-before-break analysis. Since, the NRC Enforcement Manual states that violations which existed under the old and new rule should be categorized using the current enforcement guidance, this finding was assessed as a SL IV violation.

  The significance of this violation was not formally evaluated under the Reactor Oversight Process per the Enforcement Policy, because the Agency views 10 CFR 50.59 issues as potentially impeding the regulatory process (i.e., it precluded NRC review of a change to the facility). The finding was not suitable for evaluation using the SDP. Given that the change to the incore instrumentation room sump instrumentation sensitivity capabilities and the gaseous and particulate radiation monitor sensitivities increased the length of time to detect a 1 gpm leak, and the fact that a diverse means of detecting a 1 gpm leak within one hour existed in accordance with Technical Specification (TS) requirements, the delta core damage frequency for the applicable core damage accident sequences stemming from LOCA initiating events were determined to be of very low safety significance. (Section 1R15b.(1))

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation for the failure to implement Operations Management Procedure (OMP) 1-8, “Authority and Responsibility of On-Shift Operations Personnel,” when licensed operators in the work control center and main control room did not identify that testing performed on a TS containment isolation valve rendered it inoperable and as a result, required actions were not reviewed for implementation.

  The inspectors determined that this violation was greater than minor because it affected an objective and attribute of the Reactor Safety Barrier Integrity cornerstone associated with the reactor containment integrity in that one of two in-series containment isolation valves was rendered inoperable during planned maintenance activities and not identified by operations personnel so the required TS action statement was not reviewed for implementation. The finding was assessed using the SDP for Reactor Inspection Findings for At-Power Situations. The finding was evaluated using the Phase 1 SDP analysis and determined to be of very low safety significance based on the short length of time the containment isolation valve was de-energized in the non-closed position. This finding involved a human performance cross cutting issue when the licensed operators did not adequately fulfill their duties and responsibilities to recognize and understand plant conditions to implement TS requirements properly. (Section 4OA2.2)

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 61 percent on January 12, 2005, in response to an automatic reactor power reduction initiated when the 1B main feedwater pump was removed from service due to the failure of the outboard high pressure pump seal. The unit returned to 100 percent RTP on January 19, 2005, and remained there for the remainder of the inspection period.

Unit 2 began the inspection period operating at 100 percent RTP and remained there for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Cold Weather Condition

e. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's cold weather protection program pertaining to the cold weather conditions experienced during the period of January 17 - 19, 2005. This included field walkdowns to assess the risk significant freeze protection equipment in the standby shutdown facility (SSF), refueling water storage tank pit area, and other selected outside locations. The inspectors discussed with licensed operators the specific measures to be taken during the period when low ambient temperatures were experienced. A walkdown of control room equipment related to cold weather protection was performed. The inspectors attended the monthly Catawba Action Register update meeting on Freeze Protection Readiness. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

Partial System Walkdowns

a. Inspection Scope

The inspectors verified the critical portions of equipment alignments for selected trains that remained operable while the redundant trains were inoperable. The inspectors reviewed plant documents to determine the correct system and power alignments, as well as the required positions of selected 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. Documents reviewed are listed in the Attachment to this report. The inspectors verified the following four partial system alignments:

- ‘A’ train of control room area ventilation/control room area ventilation chilled water (VC/YC) while maintenance was being performed on the ‘B’ train of VC/YC including switch/gauge calibrations, divider plate clamp replacement, and pump inservice testing (IWP) of the ‘B’ YC pump
- 1B chemical and volume control (NV) pump while preventive maintenance was performed on 1A NV pump
- 1A, 1B, and 2A trains of nuclear service water (RN) while maintenance was being performed on the 2B RN pump
- 2ETB and 1ETB emergency switchgear rooms and the 2B emergency diesel generator (EDG) while maintenance was being performed on the 2A EDG

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, 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. Documents reviewed/generated during this inspection are listed in the Attachment to this report. The inspectors toured the following eight areas important to reactor safety:

- Unit 2 auxiliary feedwater (CA) pump room, 543 foot elevation, Room 260/260A
- Unit 1 mechanical penetration room, 577 foot elevation, room 419/435
- Unit 1A EDG room
- Unit 1 switchgear room, 594 foot elevation, room 576
- Unit 2 electrical penetration room, 577 foot elevation, room 484
- Unit 1A train auxiliary shutdown panel, room 252
- Unit 1 CA pump panel, room 251
- Unit 2 spent fuel pool and new fuel receipt area, room 614/541

b. Findings

No findings of significance were identified.

.2 Fire Drill Observations

a. Inspection Scope

On February 10, 2005, the inspectors observed an unannounced shift fire drill simulating a hydrogen fire on the Unit 2 main generator, located on the 619 foot elevation of the turbine building. The purpose of this annual inspection was to: monitor the fire brigade's use of protective gear and fire fighting equipment; verify that fire fighting pre-plan procedures and appropriate fire fighting techniques were used; and 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, included discussions of drill observations, and identified any areas requiring corrective action. Documents reviewed in conjunction with this inspection are listed in the Attachment to this report.

b. Findings

Introduction: A Green non-cited violation (NCV) was identified by the inspectors for the failure of fire brigade performance deficiencies to be identified, discussed and corrected during an unannounced fire drill, as required by programs set forth in Condition 2.C.5 of the Facility Operating Licenses.

Description: On February 10, 2005, the inspectors observed an unannounced fire brigade drill involving a simulated hydrogen fire on the Unit 2 main generator at the west end of the turbine building. Once the fire brigade team members arrived at the command post location situated at the east end of the turbine building, the drill observer remained at this location to provide situational updates to the fire brigade leader and evaluate brigade member performance. Three teams, consisting of two fire fighters each, were dispatched to the location of the fire following arrival at the command post; however, their actions were not evaluated by a drill observer. As a result, individual fire brigade member performance deficiencies were not observed or discussed during the post drill critique. In addition, control room personnel actions taken in response to the simulated fire were not observed as part of the drill by the drill observer.

Analysis: This finding, associated with the Initiating Events and Mitigating Systems Cornerstones, was determined to be greater than minor because it involved the degradation of a plant fire protection feature and had a credible impact on safety since fire brigade performance deficiencies may prevent a fire from being extinguished or allow a fire to propagate, leading to a more significant event. The finding was determined to be of very low safety significance (Green) in accordance with Phase 1 of the Fire Protection Significance Determination Process because the fire brigade is only a single element of the defense-in-depth fire protection strategy and the noted deficiencies produced a minimal impact on the fire fighting capabilities of the fire brigade. This finding involved the cross-cutting aspect of human performance since, the single evaluator did not identify all of the drill deficiencies that occurred during the drill.

Enforcement: Facility Operating Licenses NPF-35 (Unit 1) and NPF-52 (Unit 2), Condition 2.C.5, requires that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in Section

9.5.1 of the UFSAR, as amended and approved in the Safety Evaluation Report (SER) through Supplement 6.

SER 9.5.1.4, “Fire Brigade and Fire Brigade Training,” states that Duke Energy Corporation will comply with Item C.3 of Branch Technical Position CMEB 9.5-1 in the establishment and training of the fire brigade.

Nuclear System Directive (NSD)-112, “Fire Brigade Organization, Training & Responsibilities,” implements these requirements related to the fire brigade manning, training and conduct of drills. Catawba Nuclear Station Fire Drill Evaluation Forms used to assess the fire brigades performance state that each fire brigade member’s role in fighting the fire shall be assessed in terms of conformance with established plant fire fighting procedures and the use of fire fighting equipment. NSD 112, section 112.6, Drills, states that, “A post drill critique shall be held for personnel participating in the drill. Performance deficiencies of a shift fire brigade or individual fire brigade members will be noted and appropriate action taken.”

Contrary to the above, during the fire drill on February 10, 2005, the licensee failed to implement the provisions of the approved fire protection program (Branch Technical Position CMEB 9.5-1) set forth in the UFSAR, regarding fire brigade training and drills when the drill evaluator did not observe and assess the performance of the three teams attacking the simulated hydrogen fire on the Unit 2 main generator or operators in the main control room. As a result, some fire brigade member performance weaknesses were not noted during the drill, discussed during the post-drill critique, or subsequently noted for development of appropriate corrective actions. Because the finding is of very low safety significance and because it has been entered into the corrective action program under Problem Investigation Process report (PIP) C-05-0807, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000413,414/2005002-01, Insufficient Fire Drill Oversight to Ensure Fire Brigade Performance Deficiencies are Identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors performed a heat sink performance inspection of the 1B containment spray (NS) heat exchanger. The inspectors observed PT/1/A/4400/006B, NS Heat Exchanger 1B Heat Capacity Test, and evaluated test data for acceptable performance. The inspectors also conducted discussions with engineering personnel concerning system configuration and heat load requirements, the methodology used in calculating heat exchanger performance, and the method for tracking the status of tube plugging used in the computer calculation program.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed a simulator exam conducted on February 23, 2005, to assess the performance of licensed operators. The scenario, Active Simulator Exam 27, involved a loss of power to an essential bus, a loss of feedwater, an anticipated transient without a scram, and a loss of heat sink with a feed and bleed scenario. 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 exam 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

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 scoped in the maintenance rule (MR) 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. The inspectors conducted this inspection for the degraded equipment conditions associated with the three items listed below. Documents reviewed are listed in the Attachment to this report.

- SSF diesel trip caused by a failed potential transformer
- Unit 2A solid state protection system (SSPS) power supply replacement during operations at power
- Unit 2 turbine driven CA pump inoperable due to a blown fuse associated with trip and throttle valve position indication

b. Findings

No findings of significance were identified.

.2 Review of MR Periodic Assessment

e. Inspection Scope

While on-site the week of March 21, 2005, the inspectors reviewed the licensee's most recent MR periodic assessment, "Duke Power Company Assessment Report Maintenance Rule Program Group Assessment, July 26 - 29, 2004", covering the period of October 1, 2003 through June 30, 2004. The report was issued to satisfy paragraph (a)(3) of 10 CFR 50.65, and covered the period indicated for the two units. The inspection was to determine the effectiveness of the assessment, the periodicity of issuance and to verify the evaluation for 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 MR systems and equipment: component cooling system (KC), instrument air (VI), RN, CA, and EDGs. Specific procedures and documents reviewed are listed in the Attachment to this report.

The inspectors reviewed the site MR implementing procedures, relevant PIPs, MR assessor unavailability hours log, (a)(4) assessments history for specific time frame, self-assessment procedure, and system health reports. The inspectors discussed issues with cognizant system engineers. Operational event information was evaluated by the inspectors in its use in MR functions. The inspectors selected system health reports and other corrective action documents of risk significant 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 MR (a)(1) evaluations and scoping documents. The inspectors also reviewed corrective actions and acceptance criteria for systems in (a)(1) such as the turbine driven CA pump and the control room ventilation system for both units to verify proper thresholds for entering systems into (a)(1) and timeliness commensurate with risk significance in resolving problems with the systems.

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 seven emergent and planned work items listed below. This review primarily focused on activities determined to be risk significant within the MR. 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 NSD 415, "Operational Risk Management (Modes 1-3)," for appropriate guidance to comply with 10 CFR 50.65 (a)(4). Documents reviewed are listed in the Attachment to this report.

- SSF unavailability for both Unit 1 and Unit 2 with turbine driven CA pump #1 unavailable

- Postponing planned maintenance on the “A” YC chiller when an abnormal noise was detected on the “B” YC chiller discharge check valve during operator rounds
- Spent fuel assembly reconstitution - Unit 2
- Replacement of the 2A SSPS power supply at power
- Replacement of the Unit 1 pressurizer level card
- Assessment of planned maintenance when the grid status became Yellow following the unexpected removal of McGuire Unit 2 from service
- Assessment of planned maintenance when the 1A EDG was unavailable due to maintenance in addition to expected inclement weather

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope

On January 12, 2005, the inspectors observed operator performance following a runback which occurred when the operators tripped the 1B main feedwater pump while at 100 percent RTP due to a reported steam leak located at the pump. The reported steam leak was a failure of the outboard high pressure seal on the feedwater pump. Unit 1 was reduced to 60 percent RTP via an automatic reactor power reduction. The plant responded as expected during the transient. The inspectors responded to the control room and observed plant stabilization and feed pump isolation activities. The inspectors observed licensed operators use of procedures and equipment manipulations during the transient. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability evaluations 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. Operability evaluations were reviewed for the seven issues listed below. Documents reviewed are listed in the Attachment to this report.

- Unexpected pressure fluctuations in the Unit 2 residual heat removal (ND) discharge header piping (PIP C-05-0218)
- Unplanned Technical Specification Action Item Log entry due to failing to meet the acceptance criteria for the CA pump turbine sump pump 2A pump down time (PIP C-04-7015)
- Unplanned entry into TSs due to pressurizer level channel 2 spiking up during Unit 1 main feedwater pump trip and runback (PIP C-05-0167)

- Active boron leak noted on the tube sheet to channel head joint on the 1A ND heat exchanger (PIP C -05-0797)
- 1A EDG day tank level was found to be not controlling in the proper range (PIP C-05-0841)
- Incore instrumentation room sump instrumentation did not meet TS requirements (PIP C-04-06541)
- Design temperature of the RN piping on the outlet of the KC heat exchangers may be exceeded during cooldown following a steam generator tube rupture (PIP C-05-0748)

b. Findings

(1) Introduction: A Severity Level (SL) IV NCV was identified by the inspectors for making a change to the facility (implemented as a change to the UFSAR in 1995) that involved an USQ, for which no written evaluation provided an adequate bases for the determination that the change did not require a license amendment pursuant to 10 CFR 50.90.

Description: The Catawba UFSAR Section 5.2.5.2.3.1, Containment Sumps, and Table 5-10, Leakage Detection Sensitivity, Containment Radiation Monitors Response Time were changed on January 12, 1995. The change was made prior to licensee implementation of the revised 10 CFR 50.59 rule, and therefore was evaluated as to whether or not the changes involved an unreviewed safety question. The inspectors determined that the change involved some Unresolved Safety Questions (USQs) and did not include a written evaluation which provided the bases for the determination that the change did not require a license amendment.

The UFSAR change, involving section 5.2.5.2.3.1, was associated with the licensee's containment sumps including the containment floor and equipment (CFE) sumps and the incore instrumentation room sump. The change involved removing the incore instrumentation room sump instrumentation from meeting the requirements committed in the UFSAR for Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems. The safety evaluation report (SER) stated that both the CFE and the incore instrumentation room sump instrumentation input to the process plant computer was designed to detect unidentified leakage inside containment in excess of 1 gallon per minute within one hour. The UFSAR change that was made changed the capability of the incore instrumentation room sump instrumentation detection capability to having an alarm that actuated whenever the sump pump started. The inspectors determined that changing the leak detection capabilities of the incore instrumentation room sump instrumentation increased the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety evaluation report. The change essentially altered an assumption of the leak-before-break design analysis, which was used in the SER assumptions to conclude acceptability of licensing basis approval, that in effect yielded a result that was considered to be in the non-conservative direction. The leak-before-break design analysis credited the leak detection system as having the capability to detect a 1 gpm leak within one hour. These changes, in effect, increased the time to detect an RCS leak and resulted in an increased probability of pressure boundary leakage due to the correlation between crack propagation and crack failure probability.

The UFSAR change, involving Table 5-10, changed both the gaseous and particulate containment radiation monitors response times from being able to detect leakage activity from corrosion products in the reactor coolant in 1 hour to being able to detect 1 percent failed fuel in the reactor coolant within one hour.

Analysis: This issue is more than minor because the change, may have affected the ability to detect a 1 gpm leak with in one hour, which was relied on in part by NRC in its approval of the original leak before break analysis and the SER. The significance of this violation was not formally evaluated under the Reactor Oversight Process per the Enforcement Policy because the Agency views 10 CFR 50.59 issues as potentially impeding the regulatory process (i.e., it precluded NRC review of a change to the facility). The finding is not suitable for evaluation using the SDP. Given that the change to the incore instrumentation room sump instrumentation sensitivity capabilities and the gaseous and particulate radiation monitor sensitivities increased the length of time to detect a 1 gpm leak and the fact that a diverse means of detecting a 1gpm leak existed in accordance with TS requirements, the delta core damage frequency for the applicable core damage accident sequences stemming from LOCA initiating events were determined to be of very low safety significance. The significance was categorized as a SL IV violation under the current Enforcement Policy, Supplement I.

Enforcement: The version of 10 CFR 50.59 that existed at the time of the change stated in part, that the licensee may make changes in the facility as described in the safety analysis report without prior Commission approval, provided the proposed change did not involve an USQ. A proposed change involved a USQ if the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SER may be increased, then the licensee shall submit an application for amendment of the license pursuant to 10 CFR 50.90. Contrary to the above, in 1995, the licensee failed to submit a license amendment pursuant to 10 CFR 50.90 when a change to the facility involving a USQ was implemented through a UFSAR change. The UFSAR change increased the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety evaluation report for the reactor coolant system loss of coolant accident leak rate predictions, because the ability to detect a 1 gpm leak in one hour was relied on for the leak-before-break design analysis.

The significance of the violation was evaluated under the 10 CFR 50.59 Rule that was in effect at the time of the change, as well as the current 10 CFR 50.59 Rule. The current 10 CFR 50.59 Rule requires, in part that "records must include a written evaluation which provides the bases for the determination that the change does not require a license amendment". Contrary to this requirement, the 10 CFR 50.59 documentation was inadequate, in that the written documentation failed to identify the relationship of the change to the one hour detection capabilities stated in the leak before break analysis. This information (i.e., the ability to detect a 1 gpm leak with in one hour) was relied on in part, by NRC for approval of the original leak before break analysis. Since, the NRC Enforcement Manual states that violations which existed under the old and new rule should be categorized using the current enforcement guidance, this finding is assessed as a SL IV violation as noted above in the Analysis section. The failure to include a written evaluation which provided adequate bases for the determination that the change did not require a license amendment as required by 10 CFR 50.59 is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV

05000413,414/2005002-02, Inadequate 10 CFR 50.59 Documentation. This issue is in the licensee's corrective action program under PIP C-05-01857.

(2) Introduction: A Green NCV was identified for the failure to establish and maintain adequate surveillance procedures for containment atmosphere radioactivity monitor TS SR 3.4.15.2 and SR 3.4.15.4, in accordance with TS 5.4.1.a.

Description: The inspectors found, on February 10, 2005, that the licensee had identified in PIP C-04-6546 that the containment particulate and gaseous monitors, used to implement TS 3.4.15, RCS Leakage Detection Instrumentation, had alarms set at values that would not indicate a 1 gpm leak in one hour. TS 3.4.15 was based on Regulatory Guide (RG) 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, which contained the 1 gpm in one hour value, and indicated that in analyzing the sensitivity of leak detection systems using airborne particulate or gaseous radioactivity, a realistic primary coolant radioactivity concentration assumption should be used. The licensee's PIP identified on December 12, 2004, that the alarm values were based on 0.1% failed fuel and would not detect a 1 gpm leak in one hour at current coolant activity concentrations. Consequently, the inspectors concluded that the alarm function was inoperable. The inspectors determined that the alarm function was required to be checked as part of the following surveillance procedures used to satisfy TS SR 3.4.15.2 and SR 3.4.15.4: IP/1/B/3314/038 Q and IP/2/B/3314/038 Q, 1EMF38 and 2 EMF38 Particulate Monitor Quarterly Channel Operational Tests, respectively; and IP/1/B/3314/039 R and IP/2/B/3314/038 Q, 1EMF39 and 2EMF39 (Low Range) Gas Channel Calibration, respectively. The inspectors found that the licensee had not established a surveillance acceptance criteria for the trip setpoints of the radiation monitors that adequately reflected the TS and UFSAR operability requirements to be able to detect a 1gpm leak in one hour. The inspectors informed the licensee that the containment radiation monitors were inoperable. The licensee implemented the TS action statements appropriately and began pursuing a licensing basis change.

On February 28, 2005, the inspectors found that the surveillance procedures used to satisfy the surveillance requirements of TS 3.4.15 did not test that the operator aid computer alarms associated with RCS leakage detection instrumentation, specified in TS 3.4.15, were received or actuated when the applicable alarm setpoints were reached. These alarms were credited for satisfying the RG 1.45 control room alarm requirements. The RG specified that RCS leakage detection instrumentation provide an alarm in the control room, which meant that they would have had to been verified via surveillance. The licensee declared all of the leakage detection instrumentation, with alarm indications in the control room, as inoperable and took the appropriate TS actions. The licensee revised the surveillance procedures and performed the necessary checks satisfactorily.

Analysis: The finding is greater than minor because the containment gaseous and particulate channel radiation monitors were not capable of performing the design bases function for an extended period of time. Additionally, operability of the RCS leakage detection instrumentation alarming functions was not verified for an extended period of time. The inoperability resulted in potential impact on reactor safety and adversely affected the availability and reliability of barrier integrity equipment performance attribute of the initiating events cornerstone. The finding was of very low safety significance because other methods of reactor coolant system leak detection were available to the licensee and no actual leakage above 1 gpm was indicated through the RCS water balance surveillance. The unavailability of the gaseous and particulate channel leak detection functions and the RCS leakage detection instrumentation alarm indications did not contribute to an increase in core damage sequences when evaluated using the significance determination phase 1 worksheets. The finding was determined to be of very low safety significance (Green). This finding involved the cross-cutting aspect of problem identification and resolution. The licensee evaluated the operability of the radiation monitors via the corrective action program and incorrectly evaluated that the radiation monitors were operable.

Enforcement: Technical Specification 5.4.1.a requires that written procedures be established, implemented and maintained covering applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 including Surveillance Procedures. Surveillance Requirement 3.4.15.2 requires the performance of a channel operational test of the required containment atmosphere radioactivity monitor every 92 days. Surveillance Requirement 3.4.15.4 requires the performance of a channel calibration of the required containment atmosphere activity monitor, every 18 months. Technical Specification Section 1.1, “Definitions,” indicates that channel calibration and channel operational tests set and test the alarms using simulated or actual signals. Regulatory Guide 1.45, position C.5 indicates that the leakage detectors should be set to detect a leakage rate, or its equivalent, of 1 gallon per minute within one hour. Contrary to the above, prior to February 11, 2005, the licensee failed to establish and maintain an adequate surveillance procedure for SR 3.4.15.2 and SR 3.4.15.4, in that the alarm function was not set or tested to alarm at a value equivalent to 1 gallon per minute in one hour for a realistic current reactor coolant activity level. Because this issue was of very low safety significance and was placed in the corrective action program under PIP C-05-1017, this violation is being treated as a non-cited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000413,414/2005002-03, Inadequate RCS Leakage Detection Instrumentation Surveillance Procedures.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors performed an in-depth review of the following operator workaround. This review assessed: (1) the impact on the reliability, availability, and potential for mis-operation of the identified system(s); (2) potential for increased initiating event frequency; and (3) impact on the ability of the operator to respond in a correct and timely manner to a plant transient and accident. Documents reviewed are listed in the Attachment to this report.

- Unit 1 pressurizer spray valves leaked past seat when they were closed, causing extra heater banks to be energized to maintain reactor coolant pressure. This condition required additional operator monitoring and oversight.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the following two permanent plant modifications to verify the adequacy of the modification packages, and to evaluate the modifications for adverse affects on system availability, reliability, and functional capability. Documents reviewed are listed in the Attachment to this report. The following plant modifications and associated attributes were reviewed:

- CNCE-73002, Addition of two vent valves to Unit 1 NV system suction piping to ensure hydrogen can be effectively removed from the system (Mitigating Systems)
  - Seismic Consideration
  - Functional Properties
  - Operational Procedures
  - Structural
  - Pressure Boundary Integrity
  - Process Medium (Fluid Pressures)

- CD-500123, Revision to the Nuclear Service Water minimum flow and Component Cooling Water temperature control setpoints for Unit 1 and Unit 2 (Mitigating Systems)

  - Operational Procedures
  - Licensing Basis
  - Testing Documents
  - Control Signals

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors witnessed and/or reviewed post-maintenance testing procedures and/or test activities, as appropriate, for selected risk significant systems to verify whether: (1) testing was adequate for the maintenance performed; (2) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (3) test instrumentation had current calibrations, range, and accuracy consistent with the application; (4) tests were performed as written with applicable prerequisites satisfied; and (5) equipment was returned to the status required to perform its safety function. Documents reviewed are listed in the Attachment to this report. The five tests reviewed are listed below:

- 2A safety injection (NI) pump performance test following preventative maintenance and oil lubrication pump check valve inspection and replacement
- Unit 2D steam generator power operated relief valve test following valve rebuild

- Replacement of 2B steam generator steam line pressure, protection channel 2 card
- Replacement of the air regulator associated with the 2A EDG
- Post-maintenance testing for motor-operated valves (MOVs) 2CA-38A, 2CA-66B, and 2CA-116B

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed the surveillance tests listed below to verify that TS surveillance requirements and/or Selected Licensee Commitment 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. Additional documents reviewed during this inspection are listed in the Attachment to this report. The following six activities were reviewed:

Surveillance Tests:

- Power range nuclear instrumentation detector N43, current versus voltage (IV) curves
- 1B NS heat exchanger flow verification
- Auxiliary Safeguards Testing initiation for the 1B NS pump
- Unit 1 End of Cycle Moderator Temperature Coefficient Reactivity Measurement

In-Service Tests:

- Auxiliary Safeguards Testing; Unit 2B CA Pump initiation and IWP
- 1B NS pump performance test

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspectors ascertained the licensee's commitments with respect to the testing and maintenance of the alert and notification system (ANS), which comprised 89 sirens in the ten-mile-radius emergency planning zone. The inspectors evaluated the design of the ANS, the licensee's methodology for testing the system, and the adequacy of the testing program design. Assessment of the program as actually implemented included review of siren test records (with an emphasis on identification of any repetitive individual siren failures), system changes during the past two years, procedures for periodic preventative maintenance (including post-maintenance testing), and a sample of corrective actions and their effectiveness for siren failures and issues. The review of this program area encompassed the period January 15, 2004, through March 14, 2005, Documents reviewed within this inspection area are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation

a. Inspection Scope

The inspectors identified the licensee's commitments with respect to timeliness and numbers of personnel for staffing emergency response facilities (ERFs) in the event of an emergency declaration at Alert or higher. The licensee's automated paging system and manual backup system for call-out of ERO personnel were reviewed to determine whether they would support staff augmentation in accordance with the criteria for ERF activation timeliness. Methodologies for testing the primary and backup systems for augmenting the ERO were reviewed and discussed with cognizant licensee personnel. The inspectors also reviewed and discussed the changes to the augmentation system and process during the past two years. The inspectors reviewed records of the last off-hour ERO augmentation drill conducted at 8:21 p.m., on June 13, 2002. Followup activities for a sample of problems identified through augmentation testing were evaluated to determine whether appropriate corrective actions were implemented. Documents reviewed within this inspection area are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes

a. Inspection Scope

The inspectors reviewed a selected sample of changes made to the Emergency Response Plan (ERP) since the last inspection in this program area conducted in

January 2003 against the requirements of 10 CFR 50.54(q) to determine whether any of the changes decreased ERP effectiveness. The subject changes, which were incorporated in ERP Revision 04-1, did not include modifications to the emergency action levels (EALs). Documents reviewed within this inspection area are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspectors evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in emergency preparedness. The procedure governing the plant corrective action program was reviewed for applicability to the emergency preparedness program. The last inspection of this program area conducted in June 2003. The inspectors reviewed event documentation to assess the adequacy of implementation of ERP requirements, as well as the licensee's self-assessment of ERO performance during the event. The inspectors evaluated selected drill scenarios and associated critiques to determine whether the licensee had properly identified failures to implement regulatory requirements and planning standards. A sample of weaknesses and deficiencies identified by means of these licensee processes was evaluated to determine whether corrective actions were effective and timely. Documents reviewed within this inspection area are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a Inspection Scope

The inspectors observed and evaluated the licensee's performance during two drills conducted on January 27, 2005, and March 23, 2005. The inspectors observed licensee activities 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 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 for both drills to evaluate the licensee's self-assessment process for identifying deficiencies relating to failures in classification and notification, as well protective action recommendations developmental activities. The inspectors assessed each of the drills 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 (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals relative to the PIs listed below for the period December 2003 through December 2004. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline”, Revision 2, were used to confirm the reporting basis for each data element.

Emergency Preparedness Cornerstone

- ERO Drill/Exercise Performance
- ERO Drill Participation
- ANS Reliability

For the specified review period, the inspectors examined data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify potential PI occurrences. The inspectors verified the accuracy of the PI for ERO drill and exercise performance through review of a sample of drill and event records. The inspectors reviewed selected training records to verify the accuracy of the PI for ERO drill participation for personnel assigned to key positions in the ERO. The inspectors verified the accuracy of the PI for alert and notification system reliability through review of a sample of the licensee’s records of periodic system tests. The inspectors also interviewed the licensee personnel who were responsible for collecting and evaluating the PI data. Documents reviewed within this inspection area are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (PI&R)

.1 Daily Screening of Items Entered Into the Corrective Action Program

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s corrective action program. This review was accomplished by reviewing copies of PIPs, attending some daily screening meetings, and accessing the licensee’s computerized database.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected two PIPs for detailed review. PIP C-05-0710, involved a periodic diagnostic test that was performed on the MOV associated with the 1D steam generator sample header outboard containment isolation valve. PIP C-04-05651, involved a flow transmitter with the incorrect range that was used in a portion of the emergency core cooling system flow balance. The two PIPs were reviewed to determine whether 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 PIPs against the requirements of the licensee's corrective action program document and 10 CFR 50, Appendix B.

b. Findings

Introduction: A Green NCV was identified for the failure to implement procedural requirements in accordance with TS 5.4.1.a, when licensed operators in the work control center and main control room did not identify that testing performed on a TS containment isolation valve rendered it inoperable and, as a result, required TS actions were not reviewed for implementation prior to or during the MOV testing.

Description: On February 7, 2005, periodic diagnostic testing was performed on the MOV associated with the 1D steam generator sample header outboard containment isolation valve in accordance with IP/0/A/3280/004F. Maintenance personnel signed on to the associated work order (WO) and performed the work using the maintenance procedure. On two separate occasions for a total of 38 minutes, the MOV was de-energized which would have prevented it from closing upon receipt of a containment isolation signal. This did not exceed the allowed Limiting Condition for Operations action time which requires that in the event one containment isolation valve is inoperable, the affected penetration must be isolated by the use of at least one component that can not be affected by a single failure within 4 hours. The inspectors determined that the scope of the work had been discussed with licensed operators in the main control room and work control center prior to the actual testing being performed. In addition, an out-of-service tag and pushbutton cover had been placed on the control panel switch associated with the containment isolation valve and an "Operation of MOVs from Outside the Control Room" form was discussed with the control room senior reactor operator (SRO) and placed in the control room log as required by the test procedure. Operations personnel did not identify the containment isolation valve inoperability or the associated TS implications until the completed work package was reviewed by operators on the following shift.

Analysis: The performance deficiency associated with this event was procedural noncompliance, in that the licensed operators failed to implement station procedure requirements of reviewing work prior to or during its execution for potential TS implications. The inspectors determined that this violation was greater than minor because it affected an objective and attribute of the Reactor Safety Barrier Integrity cornerstone associated with reactor containment integrity in that one of two in-series containment isolation valves was rendered inoperable during planned maintenance activities and not identified by operations personnel so the required TS action statement was not reviewed for implementation. The finding was assessed using the SDP for Reactor Inspection Findings for At-Power Situations. The finding was evaluated using the Phase 1 SDP analysis and determined to be of very low safety significance (Green) based on the short length of time the containment isolation valve was de-energized in the non-closed position. This finding involved a human performance cross-cutting issue, because the licensed operators did not adequately fulfill their duties and responsibilities to recognize and understand plant conditions to implement TS requirements properly.

Enforcement: TS 5.4.1.a, “Procedures,” states, in part, that, written procedures shall be established, maintained and implemented to cover the applicable procedures recommended in Regulatory Guide 1.33.

Regulatory Guide 1.33, Revision 2, dated February, 1978, Appendix A, Item 1.b states that administrative procedures covering the authorities and responsibilities for safe operation and shutdown shall be established and implemented.

OMP 1-8, “Authority and Responsibility of On-Shift Operations Personnel,” defines the roles and responsibilities of on-shift operations personnel including reactor operators and senior reactor operators at the station. These include conducting frequent control board walkdowns to verify system alignments and component status, complying with TSs and associated actions, reviewing the impact of the removal of TS related instruments and components from service and interfacing with work crews prior to the start of jobs to evaluate the work for TS impact.

Contrary to the above, on February 7, 2005, the licensee failed to implement OMP 1-8, “Authority and Responsibility of On-Shift Operations Personnel,” when licensed operators in the work control center and main control room did not identify that testing performed on a TS containment isolation valve rendered it inoperable and, as a result, required actions were not reviewed for implementation. Because this procedural noncompliance is of very low safety significance and has been entered into the corrective action program as PIP C-05-0710, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000413/2005002-04, Failure to Identify Containment Isolation Valve Inoperability During MOV Thrust Testing.

.3 Summary of PI&R Cross-Cutting Findings Documented Elsewhere

Section 1R15b.(2) describes Inadequate RCS Leakage Detection Instrumentation Surveillance Procedures. This finding involved the cross-cutting aspect of PI&R, since the licensee evaluated the operability of the radiation monitors via the corrective action program and incorrectly determined that the radiation monitors were operable.

4OA4 Summary of Human Performance Cross-Cutting Findings Documented Elsewhere

Section 1R05.2 describes the inspector identified problem concerning insufficient fire drill oversight by the licensee. It was identified as a human performance issue because the single drill evaluator did not identify all of the drill deficiencies that occurred during the drill.

Section 4OA2.2 involving a Failure to Identify Containment Isolation Valve Inoperability During MOV Thrust Testing was identified as a human performance cross cutting issue because the licensed operators did not adequately fulfill their duties and responsibilities to recognize and understand plant conditions to implement TS requirements properly.

4OA5 Other Activities

.1 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 July 2004. 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

.1 Exit Meeting Summary

On April 8, 2005, the resident inspectors presented the inspection results to Mr. D. Jamil, Site Vice President, and other members of licensee management, who acknowledged the findings. Additionally, on April 18, 2005, the resident inspectors presented inspection results to other members of licensee management, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

On March 22, 2005, the NRC's Chief of Reactor Projects Branch 1 and the Catawba Resident Inspectors met with Duke Energy to discuss the NRC's Reactor Oversight Process and the Catawba Nuclear Station annual assessment of safety performance for the period of January 1, 2004 - December 31, 2004. The major topics addressed were the NRC's assessment program and the results of the Catawba assessment. Attendees included Catawba site management, members of site staff, and local news media.

This meeting was open to the public. The presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML051090595. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Adams, Human Performance Manager
E. Beadle, Emergency Planning Manager
W. Byers, Security Manager
T. Daniels, Emergency Planning/Fire Protection
B. Dolan, Engineering Manager
J. Foster, Radiation Protection Manager
J. Fraedrich, Maintenance Rule Coordinator
T. Gaye, RN system engineer
R. Glover, Station Manager
W. Green, Reactor and Electrical Systems Manager
G. Hamrick, Mechanical, Civil Engineering Manager
D. Jamil, Catawba Site Vice President
R. Kayler, EDG system engineer
L. Keller, Regulatory Compliance Manager
A. Lindsay, Training Manager
S. Magee, Public Relations
J. McKeown, KC system engineer
G. Mitchel, Emergency Planning
M. Patrick, Work Control Superintendent
J. Pitesa, Operations Superintendent
T. Ray, Safety Assurance Manager
F. Smith, Chemistry Manager
R. Smith, Emergency Planning
G. Strickland, Regulatory Compliance Specialist
C. Trezise, Maintenance Superintendent
E. Wagner, CA and VI system engineer

NRC

M. Ernstes, Chief, Branch 1, Division of Reactor Projects, Region II (RII)
R. Hannah, Public Affairs Officer, RII

Other

J. Troutman, NCMPA-1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000413,414/2005002-01	NCV	Insufficient Fire Drill Oversight to Ensure Fire Brigade Performance Deficiencies are Identified (Section 1R05.2)
05000413,414/2005002-02	NCV	Inadequate 10 CFR 50.59 Documentation (Section 1R15b.(1))
05000413,414/2005002-03	NCV	Inadequate RCS Leakage Detection Instrumentation Surveillance Procedures (Section 1R15b.(2))
05000413/2005002-04	NCV	Failure to Identify Containment Isolation Valve Inoperability During MOV Thrust Testing (Section 4OA2.2)

Previous Items Closed

None

Items Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Alarm Response Procedures for Operator Aid Computer (OAC) points C1P0118 (Unit 1 Dry Bulb Ambient Temperature), C1P1821 (Unit 1 Wet Bulb Ambient Temperature), C2P0118 (Unit 2 Dry Bulb Ambient Temperature) and C2P1821 (Unit 1 Wet Bulb Ambient Temperature)
NSD-317; Freeze Protection Program
Freeze Protection Readiness Review update report dated 01/19/05

Section 1R04: Equipment Alignment

2005 Week 01 VC/YC Fragnet schedule
Work Order (WO) 98701670; Replace divider plate clamp
WO 98702903; IWP for YC pump B
WO 98709132; I/R low flow on 2RN-PS-7890
WOs 98708256, 98708257, 98708258; Verify RN level switch/pump operation
Safety Tagout-04-00508, Investigate and Repair Seat Leakage; 1NV pump PMs
WO 98699301; I/R 2 RN02013

WO 98691547; I/R 2 RN047A
WO 98706416; Routine RN System PMs

Section 1R05: Fire Protection

Pre-Fire Plan for fire area 2; Unit 2 CA pump room, 543 foot elevation
Pre-Fire Plan for fire area 18; Auxiliary Building, 577 foot elevation
Pre-Fire Plan for fire area 25; Diesel Generator Building Room 1A
Pre-Fire Plan for fire area 20; Unit 1 Switchgear Room, Auxiliary Building, 594 foot elevation
Pre-Fire Plan for fire area 12; Unit 1 Electrical Penetration Room, 577 foot elevation
Pre-Fire Plan for fire area 32; Unit 1A Train Auxiliary Shutdown Panel, Auxiliary building, 543 foot elevation
Pre-Fire Plan for fire area 37; Unit 1 CA Pump Panel, Auxiliary building, 543 foot elevation
Pre-Fire Plan for fire area 23; Unit 2 Spent Fuel Pool Area, 605 foot elevation
Catawba Nuclear Station Fire Drill Scenario 2005-1 and associated critique sheets
Pre-Fire Plan for Unit 2 turbine building, 619 foot elevation
Nuclear Site Directive 112; Fire Brigade Organization, Training and Responsibilities
CNS-1465.00-00-0006; Design Basis Specification for the Plant Fire Protection
UFSAR Section 9.5.1; Plant Fire Protection

<u>PIPs Generated as a result of this inspection</u>

PIP C-05-0807; Fire Brigade drill scenarios were conducted with insufficient number of evaluators/controllers to ensure adequacy of responders performance

Section 1R12.1: Routine Maintenance Effectiveness

IP/2/A/3200/002 A, SSPS Train A Periodic Testing
IP/0/A/3890/001, Troubleshooting
Critical Evolution Plan, Replace Unit 2 Train A SSPS Power Supply
WO 9871379, Inspect / Repair SSPS Train A General Warning Alarm
PIP C-04-06321, SSF D/G tripped
PIP C-01-01418, Risk Management Plans do not meet the requirement of NSDs and Site Directive, 03/28/2001
PIP G-01-00092, Areas for Improvement identified in Assessment SA-01-25(ALL)(PA), 03/28/2001
PIP G-01-00091, Procedure XSAA-113 is out of date and not being followed per NSD-415, 03/28/2001
PIP G-01-00090, SAAG review of plant modifications per XSAA-101 does not include impact review of ORAM/Sentinel models other than the PRA model, 03/28/2001
PIP G-01-00089, ORAM/Sentinel expert does not assure the latest approved O/S models are used as required by NSD-415, 03/28/2001
PIP G-01-00088, Significant differences exist between the sites regarding the implementation of risk management practices, 03/28/2001
PIP C-01-02258, IN 2001-06 discussion on centrifugal pump thrust bearing damage at Harris, 05/22/2001
PIP C-01-01701, Unit 2 KC system is being classified MR a(1) based on exceeding MPFFs limit, 04/16/2001
PIP C-01-05890, Power supplies are MR a(1) based on RMPFFs, 11/19/2001

PIP C-02-01172, Starters are MR a(1) based on adverse trend in auxiliary contacts failures, 03/07/2002
PIP C-04-00972, Limitorque Actuators SMB000 are being evaluated under MR a(1) on RMPFFs, 02/25/2004
PIP C-04-04655, TDAFW pump on both units is MR a(1) based on RMPFFs, 09/16/2004
PIP C-02-05982, Unanticipated TS entry due to SA heat tracing below required temperature, 11/09/2002
PIP C-03-02934, Unplanned TS entry due to heat trace alarms, 05/09/2003
PIP C-03-04799, Unplanned TS entry due to SA heat tracing not maintaining proper temperature, 08/29/2003
PIP C-04-03251, NLO rounds found burnt relay associated with SA heat tracing, 07/05/2004
PIP C-04-07015, Unplanned TS entry of AFW system due to down time of Sump pump 2A, 12/31/2004
PIP C-04-03682, Self-assessment and corrective actions of the CNS MR program, 07/29/2004
PIP C-03-04589, B-train YC chillers failed acceptance criteria, 08/18/2003
PIP C-01-00398, YC chillers are being declares operable but degraded, 01/24/2001
PIP C-04-01757, Divider plate in the channel head of the YC chiller became dislodged, 4/07/2004
PIP C-02-03685, 1B D/G breaker would not close on to the bus, 06/28/2002
PIP C-05-01011, EDM 410 Civil Inspection assessment, 02/24/2005
PIP C-03-07315, Unexpected TS entry due to chlorine detector failure, 12/29/2003
PIP C-97-03591, Possible inadequate RN flow to the VI compressors from the backup line, 11/10/1997
PIP C-03-01738, Part 21 on an inadequately staked capscrew rendering RHR pump inoperable, 3/17/2003
XSAA-106, Workplace Procedure for PRA Maintenance, Update, and Application, Rev. 10
XSAA-113, Guidelines for Providing PRA Support for ORAM-Sentinel, Rev. 1
XSAA-101, Risk Impact Review of Plant Changes, Modifications, and Emergency Procedure Changes, Rev. 10
EDM-410, Inspection Program for Civil Engineering Structures and Components, Rev. 10
EDM-210, Engineering Responsibilities for the Maintenance Rule, Rev. 17
EDM-310, Requirements of the Maintenance Rule, Rev. 8
NSD-403, Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10CFR50.65 a(4), Rev. 13
NSD-415, Operational Risk Management (Modes 1-3) per 10CFR50.65 a(4), Rev. 2
RN Health Report - Nuclear Service Water Health Report for 2004 2nd and 3rd trimester
VI Health Report - Instrument Air System Health Report for 2004 2nd and 3rd trimester
CA Health Report - Auxiliary Feedwater System Health Report for 2004 2nd and 3rd trimester
EDG Health Report - Emergency Diesel Generators Health Report for 2004 2nd and 3rd trimester
KC Health Report - Component Cooling Water Health Report for 2004 2nd and 3rd trimester
Catawba PRA Overview, Rev. 2b
MR Periodic Assessment for MR Implementation (April 1, 2002 - October 1, 2003), 03/30/2004
CNS Expert Panel Meeting Minutes for 11/27/2004, 02/02/2005, and 02/16/2005
Joint CNS/MNS Maintenance Rule Expert Panel Meeting Minutes, 11/08/2004
SA-01-25 (ALL)(PA), Maintenance Rule a(4) Assessment
OEDB-01-027594, Centrifugal Charging Pump Thrust Bearing Damage IN 2001-06, 05/16/2001
OEDB-04-035581, AFW pump recirculation line orifice fouling - Potential CCF, 01/22/2004
OEDB-03-034964, RHR Pump cap screws found inadequately staked, 10/29/2003
OEDB-04-036561, Plugging of safety injection pump lubricating oil coolers, 05/03/2004
OEDB-03-035456, AFW pump trip mechanism rod spring failure, 12/03/2003

OEDB-04-035571, Part 21 involving undersized EFW pump shaft, 01/20/2004
OEDB-02-031912, MSIV stem to stem disc separation resulting in Unit SCRAM, 11/20/2002
OEDB-03-032354, Target rock main steam safety/relief valve failure, 01/16/2003
OEDB-04-038097, Part 21 on GE auxiliary relays failure, 10/25/2004

Section 1R12.2: Review of MR Periodic Assessment

PIP C-01-01418, Risk Management Plans do not meet the requirement of NSDs, 3/28/2001
PIP G-01-00092, Areas for Improvement identified in Assessment SA-01-25(ALL)(PA), 03/28/2001
PIP G-01-00091, Procedure XSAA-113 is out of date and not being followed per NSD-415, 03/28/2001
PIP G-01-00090, SAAG review of plant modifications per XSAA-101 does not include impact review of ORAM/Sentinel models other than the PRA model, 03/28/2001
PIP G-01-00089, ORAM/Sentinel expert does not assure the latest approved O/S models are used as required by NSD-415, 03/28/2001
PIP G-01-00088, Significant differences exist between the sites regarding the implementation of risk management practices, 03/28/2001
PIP C-01-02258, IN 2001-06 discussion on centrifugal pump thrust bearing damage at Harris, 05/22/2001
PIP C-01-01701, Unit 2 KC system is being classified MR a(1) based on exceeding MPFFs limit, 04/16/2001
PIP C-01-05890, Power supplies are MR a(1) based on RMPFFs, 11/19/2001
PIP C-02-01172, Starters are MR a(1) based on adverse trend in auxiliary contacts failures, 03/07/2002
PIP C-04-00972, Limitorque Actuators SMB000 are being evaluated under MR a(1) on RMPFFs, 02/25/2004
PIP C-04-04655, TDAFW pump on both units is MR a(1) based on RMPFFs, 09/16/2004
PIP C-02-05982, Unanticipated TS entry due to SA heat tracing below required temperature, 11/09/2002
PIP C-03-02934, Unplanned TS entry due to heat trace alarms, 05/09/2003
PIP C-03-04799, Unplanned TS entry due to SA heat tracing not maintaining proper temperature, 08/29/2003
PIP C-04-03251, NLO rounds found burnt relay associated with SA heat tracing, 07/05/2004
PIP C-04-07015, Unplanned TS entry of AFW system due to down time of Sump pump 2A, 12/31/2004
PIP C-04-03682, Self-assessment and corrective actions of the CNS MR program, 07/29/2004
PIP C-03-04589, B-train YC chillers failed acceptance criteria, 08/18/2003
PIP C-01-00398, YC chillers are being declares operable but degraded, 01/24/2001
PIP C-04-01757, Divider plate in the channel head of the YC chiller became dislodged, 04/07/2004
PIP C-02-03685, 1B D/G breaker would not close on to the bus, 06/28/2002
PIP C-05-01011, EDM 410 Civil Inspection assessment, 02/24/2005
PIP C-03-07315, Unexpected TS entry due to chlorine detector failure, 12/29/2003
PIP C-97-03591, Possible inadequate RN flow to the VI compressors from the backup line, 11/10/1997
PIP C-03-01738, Part 21 on an inadequately staked capscrew rendering RHR pump inoperable, 03/17/2003
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OEDB-03-034964, ND Pump cap screws found inadequately staked, 10/29/2003
OEDB-04-036561, Plugging of safety injection pump lubricating oil coolers, 05/03/2004
OEDB-03-035456, CA pump trip mechanism rod spring failure, 12/03/2003
OEDB-04-035571, Part 21 involving undersized EFW pump shaft, 01/20/2004
OEDB-02-031912, MSIV stem to stem disc separation resulting in Unit SCRAM, 11/20/2002
OEDB-03-032354, Target rock main steam safety/relief valve failure, 01/16/2003
OEDB-04-038097, Part 21 on GE auxiliary relays failure, 10/25/2004

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

PIP C-05-0669; While performing routine Aux Building rounds, an unusual tapping noise was noted which seemed to come from the discharge piping on the “B” YC chiller pump in the area of 1-YC-108
WR 98337210; Inspect / repair 1 YC 108 due to abnormal noise coming from the valve
Critical Evolution Plan; Replace Unit 2 Train A SSPS Power Supply
WO 9871379; Inspect / Repair SSPS Train A General Warning Alarm
Complex Evolution Plan for the replacement of a Unit 1 pressurizer level control system card
Complex Evolution Plan for SSF unavailability for both Unit 1 and Unit 2 with turbine driven auxiliary feedwater pump #1 unavailable
Complex Evolution for Fuel Assembly Reconstitution

Section 1R14: Personnel Performance During Nonroutine Plant Evolutions

AP/1/A/5500/03, Load Rejection

Section 1R15: Operability Evaluations

Perform trend traces for ND pumps 2A and 2B discharge header pressure (points C2A1478 and C2A1484)
OP/2/A/6200/004; Enclosure 4.14; Venting the ND discharge header

PIP C-02-3469; Unit 2 NI header is pressurizing to CLA pressure
IP2B3314/039R; 2EMF39 (Low Range) Gas Monitor Channel Calibration
IP/1/B/3314/038Q; 1EMF38 Particulate Monitor Quarterly Channel Operations Test
PIP M-05-00813; Has McGuire been satisfying TS 3.4.15 surveillance requirements?
PIP M-05-5049; Incore Sump Room Hatch Leakage
EP/1/A/5000/ES-1.3; Transfer to Cold Leg Circulation
EP/1/A/5000/E-1; Loss of Reactor or Secondary Coolant
Reg. Guide 1.45; Reactor Coolant Pressure Boundary Leakage Detection Systems
PIP C-04-006546; MNS has identified a Potential Regulatory Issue concerning compliance with GDC 30 of Appendix A of 10 CFR 50
PIP C-04-006541; Evaluation of MCG PIP M04-5611; Containment floor and equipment sumps NCS leakage detection TS requirements may not be met for leakage into the incore room
SER of Westinghouse topical reports dealing with elimination of postulated pipe breaks in PWR primary main loops (Generic Letter 84-04)
UFSAR Table 11-20; Airborne process radiation monitoring equipment
10 CFR 50.59 USQ Evaluation; FSAR Section 5.2.5, Table 5-10
FSAR Table 5.2.2-1; Leakage Detection Sensitivity
UFSAR 5.2.5.2.3.1 Containment Sumps
UFSAR 5.2.5.2.3.2 Containment Airborne Monitor
FSAR 5.2.5.2.3.2 Containment Airborne Monitor
NUREG 0954; Safety Evaluation Report related to the operation of Catawba Nuclear Station
Duke Power Company Catawba Nuclear Station Environmental Report Volume 2
UFSAR 5.2.5; Detection of Leakage through Reactor Coolant Pressure Boundary
Drawing CN 1565-2.4; Flow Diagram of Liquid Radwaste System
PIP C-89-00014; Concerns about meeting the requirements of NRC Reg. 1.45
PIP C-94-00334; No NC leakage detection system operable per TS 3.4.6.1

PIPs generated as a result of this inspection

PIP C-05-01017; Document and Track actions resulting from 2/24/05 conference call between CNS, MNS, NRC Region 2, and NRR

Section 1R16: Operator Workarounds

PIP C-05-0242, Review of information and data for ODMI on leaking pressurizer spray valve
Nuclear Policy Manual, Volume 2, Appendix A.515, ODM Documentation Form, dated January 18, 2005, PIP C-05-00179.
Operations Troubleshooting Guideline, Enclosure 4.2, OP/0/A/6350/014, Unit 1 Pressurizer Spray Valves

Section 1R17:

Modification CNCE-73002
WO 98667365; Mod CE73002, Add vent valves 1NVA101 and 1NV102
Modification Package isometric drawings
Safe Shutdown Modification Screening Form
10CFR50.59 Screening Form
Modification CD-500123:
WO 98700009 & 98700010; Change setpoint on 1RN-PR-3510 and 1RN-PR-3511
PIP C-03-2910; 1A NS heat exchanger did not meet its acceptance criteria for the resistance

factor when the flow verification PT was run
Revised UFSAR (section 9.2.2.1) and TS Bases (3.7.7)

Section 1R19: Post-Maintenance Testing

IP/2/A/3222/079B; 2B S/G Steam Line Pressure, Protection Channel 2, Loop 2SMPT5120 (PT-25) Calibration
IP/0/A/3890/001; Controlling procedure for troubleshooting and maintenance
WO 98675985-01; Replacement of 2B steam generator steam line pressure, protection channel 2 card
PT/2/A/4350/002 A; Diesel Generator 2A Operability Test
PT/2/A/4200/005 A, Safety Injection Pump 2A Performance Test
WO 98690540 01, 2NI PU A- Perform PM Inspection
WO 98647131 01, 2NI PU A: Inspect Oil Pump Discharge Check Valve
WO 98960565 01, 2NI MR A: Perform Electrical Testing
PT/2/A/4200/031 A, S/G PORV Hot Stroke Test
IP/0/A/3820/040, AOV Diagnostic Testing Using The Universal Diagnostic System
PT/2/A/4200/013E; CA Valve Inservice Test

PIPs generated as a result of this inspection:

PIP C-05-0878; Several concerns were noted with PT/2/A/4350/002 A; Diesel Generator 2A Operability Test, performed on 02/08/05

Section 1R22: Surveillance Testing

PT/1/A/4400/009; Cooling Water Flow Monitoring for Asiatic Clams and Mussels Test, Enclosure 13.2, Containment Spray heat exchanger 1B flow verification
PT/1/A/4200/004C; Containment Spray pump 1B Performance Test
PT/1/A/4200/009C; Auxiliary Safeguards Test Cabinet Periodic Test; Enclosure 13.30, Containment Spray - Train B
PT/2/A/4200/009C; Auxiliary Safeguards Test Cabinet Periodic Test; Enclosure 13.10; Auxiliary Feedwater - Train B
Auxiliary Safeguards Testing; Unit 2 “B” CA Pump initiation and IWP
WO 98688315 01, Detector IV Curves on N43
IP/2/A/3240/043, N43 Analog Channel Operational Test
PIP C-05-0507; OTG Technician entered the 1B NS Pump Room with the pump on without donning the external alarming dosimetry as required by postings

Section 1EP2: Alert and Notification System Testing

Emergency Planning Functional Area Manual, 3.3 Alert and Notification System (Siren Program), 06/30/04
Low Growl Test Weekly results for 1st quarter 2005
Low Growl Test Weekly results for 1st - 4th quarters 2004

Section 1EP3: Emergency Response Organization (ERO) Augmentation

EP Group Manual Guideline 5.4.1, Emergency Response Organization Training Program
ERO Drill and Exercise Participation And supporting Data 1/1/2003 - 12/31/2004

Emergency Planning Functional Area Manual, Section 3.7 NRC Assessment Performance

Section 1EP4: Emergency Action Level (EAL) and Emergency Plan Changes

Catawba Nuclear Station Emergency Plan, Rev. 03-1
Catawba Nuclear Station Emergency Plan, Rev. 04-1
Catawba Nuclear Station Units 1 and 2 Technical Specifications and Bases
Emergency Plan Implementing Procedures change letters, December 15, 2004
Emergency Plan Implementing Procedures change letters, October 27, 2004
Emergency Plan Implementing Procedures change letters, May 27, 2004
Emergency Plan Implementing Procedures change letters, March 10, 2004
Emergency Plan Implementing Procedures change letters, February 4, 2004
RP/0/A/5000/006A, Notifications to States and Counties from the Control Room, Rev.019, 10CFR50.54(q) Evaluation Checklist Emergency Implementing Procedure
RP/0/A/5000/006B, Notifications to States and Counties from the Technical Support Center, Rev.019, 10CFR50.54(q) Evaluation Checklist Emergency Implementing Procedure
RP/0/A/5000/001, Classification of Emergency, Rev.016, 10CFR50.54(q) Evaluation Checklist Emergency Implementing Procedure
RP/0/B/5000/026, Site Response to Security Events, Rev.007, 10CFR50.54(q) Evaluation Checklist Emergency Implementing Procedure
RP/0/B/5000/026, Site Response to Security Events, Rev.006 10CFR50.54(q) Evaluation Checklist Emergency Implementing Procedure
Nuclear System Directive 228, Applicability Determination, Rev. 1
Nuclear System Directive 117, Emergency Response Organization Staffing, Training, and Responsibilities, Rev. 6
Emergency Planning Functional Area Manual, Section 3.1 Administration of the Emergency Plan and Emergency Plan Implementing Procedures, Rev. 4
Emergency Planning Functional Area Manual, Section 3.10 10CFR50.54(q) Evaluations, Rev. 4
Emergency Planning Functional Area Manual, Section 3.14 Forms for Emergency Plan Implementing Procedures, Rev. 0

Section 1EP5: Correction of Emergency Preparedness Weaknesses and Deficiencies

PIP C-05-00625, Emergency Planning Self-Assessment EMP-01-5, "4th Quarter 2004 CNS Emergency Planning (EP) Business Measures and EP Track & Trend Review.
EMP-01-05, Emergency Planning Group Assessment Plan, 1/01/05
EMP-03-04, Emergency Planning Group Assessment Plan, 7/1/04
EMP-02-04, Emergency Planning Group Assessment Plan, 4/12/04
EMP-06-04, Emergency Planning Group Assessment Plan, 10/01/04
EMP-01-04, Emergency Planning Group Assessment Plan, 1/19/04
EMP-04-04, Emergency Planning Group Assessment Plan, 8/16/04
GO-04-14(NPA)(EP)(ALL), Duke Power Company 2004 Emergency Planning Assessment, 09/07/04
PIP C-04-01367, ERO Drill 04-01 and ERO Drill 04-02
PIP C-04-05895, Duke Power Company 2004 Emergency Planning Assessment, GO-04-14(NPA)(EP)(ALL)
PIP C-04-02184, Emergency Planning Self-Assessment EMP-02-04
PIP C-04-02584, ERO Drill 04-03

Section 1EP6: Drill Evaluation

Catawba Nuclear Station Drill 05-1

Section 4OA1: Performance Indicator (PI) Verification

EP Group Manual Guideline 5.4.1, Emergency Response Organization Training Program
Catawba Nuclear Station Training Addenda, Addendum 7111.0
Emergency Planning Functional Area Manual, Section 3.7 NRC Assessment Performance Indicator Guideline - Emergency Preparedness Cornerstone, Rev. 6
Monthly Data for PI - Performance Indicator Validation Worksheet: for 1/1/2003 - 12/31/2004
    Alert and Notification System (Sirens)
    Drill/Exercise Performance
    ERO Drill and Exercise Participation And supporting Data

Section 4OA2.2: Problem Identification and Resolution Annual Sample Review

PIP C-05-00710; During testing of valve, found flow on valve. White tags were needed to meet TS requirements. This is a past operability concern due to work scope not being fully understood.
Performance OAC point trends for C1Q0088 Valve NM221A S/G D sample hdr containment isolation and C1Q0442 Valve NM220B S/G D blowdown simple containment isolation
WO 98687112091; 1NN221A, perform MOV thrust test
IP/0/A/3820/004F; Kerotest MOV Diagnostic Testing
IP/0/A/3820/004; Operating Checkout of Limitorque & Rotork Valve Actuators
Catawba Nuclear Station 1NM221A Mo Actuator Data Sheet
Site Directive 3.10.1, Att. 2; Operation of Motor Operated Valves from outside the Control Room
Root Cause Failure Analysis Report; Configuration control during MOV thrust testing of 1NM-221A

LIST OF ACRONYMS USED

ANS	-	Alert and Notification System
CA	-	Auxiliary Feedwater
CFE	-	Containment Floor and Equipment
CFR	-	Code of Federal Regulations
CNS	-	Catawba Nuclear Station
EAL	-	Emergency Action Level
EDG	-	Emergency Diesel Generator
EMF	-	Radiation Monitors
ERF	-	Emergency Response Facility
ERO	-	Emergency Response Organization
ERP	-	Emergency Response Plan
INPO	-	Institute of Nuclear Power Operations
IWP	-	Pump Inservice Testing
KC	-	Component Cooling Water
MOV	-	Motor Operated Valve
MR	-	Maintenance Rule

NCV	-	Non-Cited Violation
ND	-	Residual Heat Removal
NI	-	Safety Injection
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
NS	-	Containment Spray
NSD	-	Nuclear System Directive
NV	-	Chemical and Volume Control
OMP	-	Operations Management Procedures
OP	-	Operating Procedure
PI	-	Performance Indicator
PIP	-	Problem Investigation Process (report)
RCS	-	Reactor Coolant System
RG	-	Regulatory Guide
RN	-	Nuclear Service Water
RTP	-	Rated Thermal Power
SDP	-	Significance Determination Process
SER	-	Safety Evaluation Report
SL	-	Severity Level
SR	-	Surveillance Requirement
SRO	-	Senior Reactor Operator
SSF	-	Standby Shutdown Facility
SSPS	-	Solid State Protection System
TS	-	Technical Specification
UFSAR	-	Updated Final Safety Analysis
USQ	-	Unreviewed Safety Questions
VC	-	Control Room Area Ventilation
VI	-	Instrument Air
WO	-	Work Order
YC	-	Control Room Area Ventilation Chilled Water