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

October 26, 2006

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

SUBJECT: CATAWBA NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000413/2006004 AND 05000414/2006004

Dear Mr. Morris:

On September 30, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Catawba Nuclear Station Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on October 5, 2006, with Mr. Regis Repko and other members of your staff.

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

This report documents two NRC-identified findings of very low safety significance (Green) which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they have been entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV in this report, you should provide a written 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 Senior 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 *www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael E. 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 05000413/2006004 and 05000414/2006004 w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to J. R. Morris from Michael Ernstes dated October 26, 2006

SUBJECT: CATAWBA NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000413/2006004 AND 05000414/2006004

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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.:	05000413/2006004 and 05000414/2006004
Licensee:	Duke Power Company, LLC
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	York, SC 29745
Dates:	July 1, 2006 through September 30, 2006
Inspectors:	E. Guthrie, Senior Resident Inspector E. Riggs, Resident Inspector A. Sabisch, Senior Resident Inspector G. Williams, Project Engineer E. Lea, Senior Operations Engineer (Section 1R11.1) G. Laska, Senior Operations Examiner (Section 1R11.1) R. Chou, Reactor Inspector (Section 4OA5.2)
Approved by:	Michael E. Ernstes, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000413/2006-004, 05000414/2006-004; 7/1/2006 - 9/30/2006; Catawba Nuclear Station, Units 1 and 2; Event Followup, Mitigating Systems.

The report covered a three-month period of inspection by three resident inspectors, a project engineer, two operations engineers, and a reactor inspector. Two Green findings which were non-cited violations (NCVs) were 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, (ROP) Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. An NRC-identified NCV of 10 CFR 55.59 was identified for failure to adequately examine Senior Reactor Operators (SROs). Job Performance Measures (JPMs) that contained immediate operator actions was excluded from the sample of JPMs used to examine SROs.

  The finding is more than minor because if left uncorrected it would lead to a more significant safety concern and affected the Mitigating Systems cornerstone. This finding affected an individual operating examination, was related to examination quality, and affected more than 20% of the SRO operating tests. Using MC 0609 Appendix I, License Operator Requalification Significance Determination Process (SDP), the inspectors determined the finding was of very low safety significance. (Section 1R11.1)

- Green. A self-revealing NCV of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings, was identified when the licensee failed to maintain appropriate design control in ensuring below-grade electrical conduits were properly sealed to prevent water intrusion into areas of the plant containing safety-related equipment.

  The finding is more than minor in that it affected the flood hazard objective of the Protection Against External Factors attribute under the Mitigating Systems cornerstone. Based on the results of the Significance Determination Process Phase 1 screening and the Phase 2 evaluation using the Catawba Plant Notebook, it was determined that a Phase 3 evaluation was required. A regional Senior Risk Analyst performed a Phase 3 SDP evaluation and determined the performance deficiency was of very low safety significance. The dominant factor in the analysis was that a tornado-induced Loss of Offsite Power (LOOP) would have to coincide with a Predicted Maximum Precipitation flooding event. Such an initiating event frequency was sufficiently low enough to determine that, when also considering the possible recovery actions such as cross tying power from Unit 2 or the recovery of the 1A DG, that the performance deficiency was Green. Although the failure to seal the electrical conduits occurred during initial construction, this finding was not considered to be an old design issue because it was identified through a self-revealing event. (Section 4OA3.1)

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) and remained at 100 percent RTP through the end of the inspection period.

Unit 2 began the inspection period operating at 100 percent RTP and remained at or near 100 percent RTP through the end of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Severe Weather Condition

a. Inspection Scope

On August 30, severe thunderstorms with heavy rainfall were experienced on site. As a result, pools of standing water formed and some water entered the Unit 1 and Unit 2 turbine building basement elevation. The inspectors reviewed the plant response to the water intrusion including drainage of water from inside the flood walls surrounding the 6.9kV transformers and associated terminal cabinets. The inspectors toured the plant site to check for possible blockage of flow paths leading to the yard drain system and other points where water could enter the power block structures. The inspectors interviewed Operations, Maintenance, and Engineering personnel regarding preparations and inspections performed prior to the arrival of the severe weather, and reviewed the adequacy of the operations procedures associated with internal flooding, and additional actions to be taken subsequent to the August 30 conditions. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed a walkdowns of the following three system alignments to verify that critical portions of equipment alignments remained operable while the redundant trains for that system 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 reviewed equipment alignment problems which could cause initiating events or impact mitigating system availability to verify that they had been properly identified and resolved. Documents reviewed are listed in the Attachment.

- The 2A DG with the 2B DG out of service for maintenance
- The 1B Charging and Volume Controls System (NV) pump and support equipment with the 1A NV pump out of service for preventative maintenance work
- Protection of safety-related equipment potentially impacted by the Nuclear Service Water piping excavation

b. Findings

No findings of significance were identified.

.2 Complete Walkdowns

a. Inspection Scope

The inspectors conducted one complete walkdown/review of the Unit 2 auxiliary feedwater system. The inspectors utilized licensee procedures, as well as licensing and design documents to verify that the system (i.e., pump, valve, and electrical) alignment was correct. During the walkdown, the inspectors also checked that valves and pumps did not exhibit leakage that would impact their function; major portions of the 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, system Health Reports, and outstanding maintenance work requests/work orders. A review of open Problem Investigation Process reports (PIPs) was also performed to verify that the licensee had appropriately characterized and prioritized auxiliary feedwater related equipment problems for resolution in the corrective action program. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors walked down accessible portions of the following eight plant areas 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 are listed in the Attachment.

- Unit 2 main transformer area
- Unit 1 A & B NV pump rooms
- Unit 1 Mechanical Penetration Room, 560 foot elevation
- Unit 1 A & B Containment Spray pump rooms
- Unit 2 Cable Tray Access Room
- Unit 2 refueling water storage tank area
- Unit 1 Auxiliary Shutdown Panels
- Unit 1 Electrical Penetration Room, 577 foot elevation

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

.1 External Areas

a. Inspection Scope

The inspectors reviewed the licensee's external flood protection features. The inspectors performed a walkdowns of external site areas including inside several nuclear service water conduit manhole bunkers to assess flood protection measures. Through observation and design review, the inspectors reviewed sealing of doors, cables and splices subject to submergence, sump pump and level circuit operations, and potential flooding sources. The inspectors reviewed the corrective action program documents to verify that the licensee was identifying and resolving flooding protection issues. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Biennial Program Review

a. Inspection Scope

The inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of operating tests and written examinations associated with the licensee's operator requalification program to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR Part 55, Operators'

Licenses. The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG 1021, Operator Licensing Examination Standards for Power Reactors, and Inspection Procedure 71111.11, Licensed Operator Requalification Program. The inspectors also evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations using ANSI/ANS-3.5-1998, American National Standard for Nuclear Power Plant Simulators for use in Operator Training and Examination. The inspectors observed three crews during the performance of operating tests. Documents reviewed are listed in the Attachment.

Following the completion of the annual operating tests, the inspectors reviewed the overall pass/fail results of the biennial written examination, the individual JPM operating tests, and the simulator operating tests administered by the licensee. These results were compared to the thresholds established in Manual Chapter 609, Appendix I, Operator Requalification Significance Determination Process.

b. Findings

Introduction. An NRC-identified Green NCV of 10 CFR 55.59 was identified for failure to adequately examine Senior Reactor Operators (SROs). JPMs that contained immediate operator actions was excluded from the sample of JPMs used to examine SROs.

Description. While observing annual operating tests, inspectors observed that SROs were not administered the same JPMs as the ROs. The inspectors found that JPMs containing immediate operator actions had been removed from the sample for SRO examinations. The inspectors determined that, in 2004, Catawba's training staff determined that SROs would not be examined with JPMs that contained immediate operator actions. The basis for this decision was that SROs typically do not operate the console controls.

Analysis. The finding is more than minor because if left uncorrected it would lead to a more significant safety concern and affected the Mitigating Systems cornerstone. This finding affected an individual operating examination, was related to examination quality, and affected more than 20% of the SRO operating tests. Using MC 0609 Appendix I, License Operator Requalification Significance Determination Process (SDP), the inspectors determined the finding is of very low safety significance (Green).

Enforcement. 10 CFR 55.59(a)(2)(ii) states, in part that, "…the operating test will require the operator or senior operator to demonstrate an understanding of and the ability to perform the actions necessary to accomplish a comprehensive sample of items specified in 10 CFR 55.45 (a) (2) through (13) inclusive to the extent applicable to the facility." 10 CFR 55.45 (a)(2) states "Manipulate the console controls as required to operate the facility between shutdown and designated power levels;" and 10 CFR 55.45 (a)(3) states "Identify annunciators and condition-indicating signals and perform appropriate remedial actions where appropriate." Contrary to the above, excluding JPMs that contained immediate operator actions from the operating tests for SROs did not meet the requirements of 10 CFR 55.45 (a)(2) or (a)(3). Because this finding is of low safety significance and has been entered into the corrective action program as PIP C-06-05501, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000413,414/2006004-01, Inadequate Senior Reactor Operator Operating Examinations.

.2 Resident Inspector Quarterly Review

a. Inspection Scope

The inspectors observed OP-CN-ASE-25, Active Simulator Exam Scenario 25, Rev. 17 to assess the performance of licensed operators. The exercise included a pressurizer power operated relief valve failure, the failure of the main turbine to trip automatically when required and a loss of the secondary heat sink. The inspection focused on high-risk operator actions performed during implementation of the abnormal and emergency operating procedures, 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 if appropriate feedback was provided to the licensed operators regarding identified weaknesses.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee’s effectiveness in performing the following two 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 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 are listed in the Attachment.

- Manual operation of steam generator power operated relief valve 2SV-13 exceeded the acceptance criteria and required rework (PIP C-06-4961)
- Replacement of hydrogen igniter circuit breakers on both Unit 1 and Unit 2

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 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), and Nuclear System Directive 403, Shutdown Risk Management (Modes 4,5,6, and No Mode), for appropriate guidance to comply with 10 CFR 50.65 (a)(4).

- Work on Power Circuit Breaker (PCB) 29 with 2A DG emergent work in-progress
- Reviewed of planned work coded as T2 with Oconee Unit 2 off-line
- Reassessment of scheduled work and troubleshooting activities during period of heavy thunderstorm activity at the site on 8/30/06
- Planned and emergent work with the 1B DG out of service for modification work
- Review of planned and emergent work activities following identification of the failed fan motor on the 2B Charging and Volume Controls System
- Planned and emergent work with the 2B DG out of service for modification work
- Review of planned and in-progress activities prior to the excavation of the nuclear service water piping in conjunction with the construction of the independent spent fuel facility installation crawler roadbed

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the six operability evaluations listed below, the inspectors evaluated the technical adequacy of the evaluations to ensure that Technical Specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors reviewed the Updated Final Safety Analysis Report to verify that the system or component remained available to perform its intended function. In addition, the inspectors reviewed compensatory measures implemented to verify that the compensatory measures worked as stated and the measures were adequately controlled. The inspectors also reviewed a sampling of PIPs to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment.

- Ground detected on the 2A DG (PIP C-06-5397)
- Unexpected entry into Tech Specs due to 2B2 Emergency Ventilation fan trip during 2B Diesel Generator operability test (PIP C-06-05104)
- During performance of 1A2 Component Cooling pump Inservice Pump Test (IWP), component cooling pump was operated above run-out flow rate (PIP C-06-05694)
- During Non-Destructive Examination inspections of the 2A2 diesel fuel oil storage tank, one reading was found to be below the minimum wall thickness value (PIP C-06-6082)
- Internal insulation was found dislodged and partially blocking airflow in the Unit 1 Auxiliary Building Ventilation supply ducting (PIP C-06-6017)
- Unplanned entry into Technical Specification Action Item Log due to hydrogen igniter breaker tripping (PIP C-06-5085)

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post-maintenance test listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee’s test procedure to verify that the procedure adequately tested the safety function(s) that may have been affected by the maintenance activity, that the acceptance criteria in the procedure were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed the test data to verify that test results adequately demonstrated restoration of the affected safety function(s). Documents reviewed are listed in the Attachment.

- Replacement of the existing 30A Hydrogen Igniter circuit breakers with new 40A circuit breakers

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed the eight surveillance tests listed below to verify that Technical Specification surveillance requirements and/or Select 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 pre-conditioning 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. Documents reviewed are listed in the Attachment.

Surveillance Tests
- PT/1/A/4200/007C, Standby Makeup Pump #1 Performance Test, Rev. 37
- PT/1/A/4200/009A, Auxiliary Safeguards Test Cabinet Periodic Test, Rev. 225, Enclosures 13.12, 13.20, 13.75 and 13.77
- IP/2/A/3200/001A, Solid State Protection System Train A Periodic Testing, Rev. 001
- PT/2/A/4350/002A, Diesel Generator 2A Operability Test, Rev. 085
- PT/0/A/4150/012B, Moderator Temperature Coefficient of Reactivity Measurement (End of Life), Rev. 24

In-Service Tests
- PT/2/A/4200/007A, 2A NV pump IWP, Rev. 34
- PT/2/A/4200/010A, 2A Residual Heat Removal pump IWP, Rev. 46
- PT/1/A/4400/003A, Component Cooling Train 1A Performance Test, Rev. 71

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed and evaluated the licensee's performance during emergency drills conducted on August 31 and September 7. 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, offsite agency notification, 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 the drill to evaluate the licensee's self-assessment process for identifying potential deficiencies relating to failures in classification and notification, as well protective action recommendation process activities. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (PI&R)

.1 Daily Review

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 screening of items entered into the licensee's corrective action program. This was accomplished by reviewing copies of PIPs, attending some daily screening meetings, and accessing the licensee's computerized database.

4OA3 Event Follow-up

.1 Flooding of the Unit 1 A Diesel Generator Room on May 22, 2006

a. Inspection Scope

The inspectors responded to the 1A diesel generator room flooding that occurred on May 22. The inspectors discussed the event with the operators, engineering, and licensee management personnel to gain an understanding of the event and assess followup actions. The inspectors reviewed the licensee's initial investigation reports and cause determinations. An NRC augmented inspection was initiated on May 23 in response to this event and the LOOP that occurred two days prior to the flooding event. The NRC Augmented Inspection Report can be reviewed via Inspection Report 05000413/2006-009 and 05000414/2006-009.

b. Findings

(Closed) Unresolved Item (URI) 05000413/2006009-03; Review of failure to seal conduits into manholes and the 1A DG room as required by design and construction documents

Introduction. A self-revealing Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings was identified when the licensee failed to maintain appropriate design control in ensuring below-grade electrical conduits were properly sealed to prevent water intrusion into areas of the plant containing safety-related equipment.

Description. On May 22, 2006, the control room received notification of flooding occurring in the 1A diesel generator room. Operations personnel were dispatched and water was found to be entering the room through electrical conduits penetrating the outer wall onto diesel generator support equipment and into the room itself. The flow rate exceeded the capacity of the two installed sump pumps and as the pumps became submerged, one of the pumps failed. The water level in the sump area continued to increase.

The licensee brought in temporary sump pumps to remove the water that was collecting in the diesel generator sump area. The determination was made that the source of the water was overflow from the Unit 2 cooling towers which had found its way through trenches and unsealed electrical cable conduits into the diesel generator room. The cooling towers producing the overflow conditions were secured and the influence into the diesel generator room subsided.

The 1A DG was declared inoperable based on the components that had been wetted during the flooding event which required inspection and testing to ensure that overall operability of the diesel generator could be assured. These components included the 1) the diesel pre-lube oil pump, motor and filter; 2) the diesel generator room sump pumps and motors; 3) the diesel starting air compressors; 4) the diesel generator vital battery enclosure, and 5) the load sequencer cabinets located in the corridor outside the diesel room. Work Orders were initiated to perform inspections and necessary repairs on the aforementioned equipment. All inspections and repairs were completed prior to declaring the 1A diesel generator operable. The total time the diesel generator was logged as being inoperable was approximately 60 hours.

As part of the licensee’s immediate corrective actions, engineering and maintenance personnel conducted an extent-of-condition review by inspecting all other electrical conduits entering areas of the plant containing safety-related equipment. During these inspections, additional seals were identified to either be missing or degraded. The conditions were corrected prior to restarting either unit following the dual-unit loss of offsite power.

Electrical Equipment Layout Drawing CN-1938-01, Rev. 66, shows that the electrical conduits communicating between the cooling tower cable trench and safety-related manhole CMH-4A and between the safety-related manholes and the DG rooms were to have been sealed to prevent drainage into the safety-related manholes or buildings. The lack of these seals resulted in the unobstructed flowpath that allowed the cooling tower overflow to reach the 1A DG room.

Analysis. The inspectors determined that the failure to maintain proper design control during initial construction and install the required seals associated with safety-related electrical conduits as specified on the construction drawings was a performance deficiency. The finding is more than minor in that it affected the flood hazard objective of the Protection Against External Factors attribute under the Mitigating Systems cornerstone. Based on the results of the Significance Determination Process Phase 1 screening and the Phase 2 evaluation using the Catawba Plant Notebook, it was determined that a Phase 3 evaluation was required. A regional Senior Risk Analyst performed a Phase 3 SDP evaluation and determined the performance deficiency was of very low safety significance. The dominant factor in the analysis was that a tornado-induced Loss of Offsite Power (LOOP) would have to coincide with a Predicted Maximum Precipitation flooding event. Such an initiating event frequency was sufficiently low enough to determine that, when also considering the possible recovery actions such as cross tying power from Unit 2 or the recovery of the 1A DG, that the performance deficiency was Green. Although the failure to seal the electrical conduits occurred during initial construction, this finding was not considered to be an old design issue because it was identified through a self-revealing event.

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part that activities affecting quality shall be prescribed by documented drawings and shall be accomplished in accordance with these drawings. Drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, on May 22, 2006, while investigating the flooding of the 1A diesel generator room, it was identified that the licensee had failed to include appropriate quantitative or qualitative acceptance criteria for ensuring that below-grade electrical conduits had been sealed in accordance with initial construction drawings to prevent external flooding from impacting areas of the plant containing safety-related equipment.

Because this finding is of very low safety significance and was placed in the corrective action program as PIP C-06-3902, this violation is being treated as a non-cited violation in accordance with Section VI.A.1 of the Enforcement Policy, and is identified as NCV 05000413/2006004-02, Failure to Maintain Design Control Over Installation of Seals in Below-Grade Electrical Conduits.

.2 Improper relay settings in the 230kV switchyard resulting in a total loss of offsite power to both units

a. Inspection Scope

The inspectors responded to the dual-unit loss of offsite power (LOOP) event that occurred on May 20, 2006. The inspectors monitored the licensee's response to the even and interviewed operators, engineering, and licensee management personnel to gain an understanding of the event and assess followup actions. The inspectors reviewed the licensee's initial investigation reports and cause determinations. An NRC augmented inspection was initiated on May 23 in response to this event and the subsequent flooding of the 1A diesel generator room that occurred two days following the LOOP event. The NRC Augmented Inspection Report can be reviewed via Inspection Report 05000413/2006-009 and 05000414/2006-009.

b. (Closed) URI 05000413, 414/2006009-02; Improper relay settings in the Catawba 230kV switchyard resulting in a total loss of offsite power following failure of a PCB current transformer

Background. On May 20, 2006, both Catawba units tripped automatically from 100% power following a LOOP event. The event began when a fault occurred internal to a current transformer (CT) associated with one of the switchyard power circuit breakers. A second current transformer failure along with the actuation of differential relaying associated with both switchyard busses, cleared both busses and separated the units from the grid. The root cause of the event was determined to be that certain switchyard relay tap settings were set at a value too low to handle the fault currents experienced during this transient.

The initial installation and setting of these relay occurred during the construction phase of the Catawba station. The switchyard relaying scheme was designed in the late 1970's by Duke Energy's Design Engineering Department, which had responsibility for the design of the entire station. The settings for the relays were initially calculated by Duke Energy's Power Delivery Relay Engineering group based on grid demand, projected growth and postulated fault currents. Power Delivery managed the field technicians that physically set the relays during construction of the switchyard prior to commercial operation in 1979. In 1981, Power Delivery revised the setpoints for the differential relays in the Catawba switchyard; however, no documentation has been located that indicates that this revised setpoint was ever communicated to the field technicians via updated relay setting cards for implementation at the next available opportunity.

In the late 1980's, the relay field technicians that maintained these relays were reassigned to the station's in-house maintenance organization. They brought the relay setting cards with them that still reflected the original 100-volt tap settings, which were incorporated into the station's maintenance documentation. The relay settings cards were converted to station procedures in 1996.

The relays were included in the station's maintenance program throughout the operational life of the station. However, they were tested and maintained in accordance with the lower values that were reflected on the original relay setting cards and consequently, the latent error of the lower-than-desired tap settings was not discovered until it was revealed by the dual unit LOOP event on May 20, 2006.

Analysis. The event and associated root cause was screened using Manual Chapter (MC) 0612, Power Reactor Inspection Reports; Appendix B, Issue Screening, to determine if a Performance Deficiency existed which led to the LOOP. Appendix B defines a performance deficiency as "…an issue that is the result of a licensee not meeting a requirement or standard where the cause was reasonably within the licensee's ability to foresee and correct, and that should have been prevented." The three aspects reviewed following the event contained in the definition of a Performance Deficiency include applicable requirements, applicable standards and opportunities for the licensee to foresee and correct the condition.

In terms of requirements, the switchyard and all components in the switchyard are non safety-related, and therefore do not fall under 10CFR50, index B, Criterion XVI requirements. Catawba's Quality Assurance (QA) Program requirements are not applicable to these components for the same reason. The Duke Power Delivery switchyard analysis is not a QA analysis or a QA document; therefore, it does not become part of the design basis for the plant. The licensee is not required to perform periodic assessments and reviews of design of switchyard per their QA program. With regard to the maintenance rule, the switchyard system is scoped as required and performance of routine monitoring and maintenance of the components that fall within the scope of the rule meets the requirements as stated in basis documents. The specific relays associated with the red and yellow busses were checked on a routine basis using the relay setting cards and, more recently, station procedures.

An assessment was conducted to determine if there were any applicable industry standards that were not met where the cause was reasonably within the licensee's ability to foresee and correct. The Institute of Electrical and Electronics Engineers (IEEE) 308 "IEEE Standard Criteria for Class 1E power Systems for Nuclear Power Generating Stations," in conjunction with other related IEEE standards, established specific design criteria for nuclear power plant electrical systems and equipment. The switchyard is a 'Preferred Power Supply' which has to meet design criteria to supply power to the class 1E systems and not abnormally effect those systems in the event of fault, etc. The design of the Catawba switchyard met those requirements. No 'standard' exists in the IEEE that the plant does not meet. Maintenance activities conducted in the switchyard meet all current industry standards and practices. The licensee is involved in the Institute of Nuclear Power Operations (INPO) and switchyard owners group and have implemented and / or evaluated all recommendations from these sources.

The Federal Energy Regulatory Commission boundary between the generation stations and transmission assets is the high-to-low voltage interface of the main step-up transformer. All equipment and systems from the low voltage side of the main step-up transformer into the station are considered generation assets. All equipment and systems from the high voltage side of the main step-up transformer into the switchyard and transmission grid are considered transmission assets.

Based on a review of current Duke Energy procedures and available documentation from the 1981 time-frame, there were no requirements; regulatory, industry standards or self-imposed, that were not met by the Power Delivery group (transmission) or the licensee (generation) in regards to the switchyard relay settings.

A review was performed to determine if the licensee had previous opportunities to identify the lower relay settings. Three specific previous anomalies occured between 1998 and 2004 at Catawba and McGuire related to the switchyard that initially directed station personnel to look at relay settings. In all cases, another cause for the event was determined to be the actual contributor before any review of relay tap settings was conducted by the Catawba engineering or maintenance staff. None of the reviews concluded that the relays needed to be changed, nor did they question the actual settings.

In mid-2005, Power Delivery revised the relay settings based on calculations performed due to increased load projections for the grid surrounding the Catawba station. These forward-looking projections are performed periodically system-wide as demands on the grid increase based on local growth. The requisite skill set needed for performing these calculations is centralized in the Power Delivery group who does them for all Duke Energy generation and distribution facilities. As a result, the station did not require an independent verification of the calculation once received from Power Delivery. The Power Delivery group believed that the relays had been set at 250 volts, and as a result, told Catawba that it was acceptable to wait until the next scheduled maintenance window for the relays to make the adjustment. The engineering group at Catawba developed revisions to the relay calibration procedures for use during the next maintenance window. This window was in November 2006 for the Yellow Bus and April

2007 for the Red Bus, approximately five months after the LOOP event occurred at the station.

Following the LOOP event, the licensee initiated a root cause investigation into the cause of the event using station personnel. In addition, a corporate Special Event Investigation Team was formed to conduct an independent evaluation of the station's response to the event, potential contributors to the event and determine if additional corrective actions were required to prevent recurrence. As a result of these assessments, several corrective actions were developed and assigned to station and off-site groups for implementation. The assessments and corrective actions were documented in PIP's C-06-3864 and C-06-3865, and the Special Event Investigation Team Final Investigation Report. Corrective actions implemented in response to the LOOP event included the following:

- The setpoints of all Power Delivery-controlled relays in the Catawba switchyard were verified to be set properly
- Enhancements were made to the methodology used to maintain the switchyard model including configuration management related to changes to setpoints and formal communication of those changes to all applicable groups
- Detailed inspections of all CT's in the Catawba switchyard were conducted to ensure no additional degradation existed on the CT's still in-service
- Station operating and maintenance procedures were reviewed and enhancements added as deemed appropriate based on the post-event review conducted by station and corporate personnel

Additional corrective actions associated with the LOOP event have been or are scheduled for implementation at Catawba and within the Power Delivery group.

Enforcement. Based on an in-depth review of all available information pertaining to the inappropriately set protective relays in the Catawba switchyard that resulted in the dual unit loss of offsite power on May 20, 2006, there is no performance deficiency associated with the licensee’s actions that could have prevented the event. The definition of a “performance deficiency” requires that a requirement or standard was not met and that the licensee had the opportunity to identify and correct the condition. As discussed in the previous section, there are no requirements or standards that were not met in this situation and it was not reasonable for the licensee to have identified the lower-than-desired relay tap setting prior to the planned switchyard maintenance outage and re-adjustment of the relays scheduled for the late-2006 / early-2007 time frame.

This will serve as the basis to close URI 05000413, 414/2006009-02; "Improper Relay Settings in the Catawba 230kV Switchyard Resulted in a Total Loss of Offsite Power Following Failure of a PCB Current Transformer"; with no enforcement action being taken against the licensee for the LOOP event.

4OA5 Other Activities

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

The inspectors reviewed the interim report issued by INPO for the evaluation that was conducted at the Catawba facility during June 2006. The inspectors did not note any safety issues in the INPO report that neither warranted further NRC followup nor that had not already been addressed by the NRC.

.2 (Closed) Unresolved Item (URI) 72-45/2005-001-01, ISFSI Concrete Storage Pad Calculated Seismic Acceleration Exceeds the General Licensee Limit

The inspectors reviewed NRC Safety Evaluation Report (SER) and Certificate of Compliance (CoC) No. 1015 for Docket No. 72-1015, NAC-UMS STORAGE SYSTEM Amendment No. 4. Section B3.4.1.3.(b) of Appendix B, Design Features of the CoC, requires, alternatively for the site seismic acceleration exceeding the general license limit, that the acceleration g-load resulting from the collision of two sliding casks remain bounded by the accident condition analyses presented in Chapter 11 of the FSAR (this was a part of the CoC) and site-specific analysis by the cask user shall demonstrate that a cask does not slide off the ISFSI pad. The licensee generated Calculation No. CNC-1140-04-00-0001, Rev. 0, Seismic Qualification of NAC Casks on CNS ISFSI Pads Using NUREG/CR-6865. The NUREG/CR-6865, Parametric Evaluation of Seismic Behavior of Freestanding Spent Fuel Dry Cask Storage Systems involved data from the peak ground acceleration ranging from 0.25 to 1.25 g which were collected from requirements of the seismic design from one Regulatory Guide and two NUREG/CRs. The licensee generated Calculation No. CNC-1140-04-00-0001 based on the analytical methods defined in NUREG/CR 6865 to verify that casks on the Catawba ISFSI concrete pads would not collide with each other and slide off during the design base earthquakes. The general license in its FSAR Rev. 5, dated October 2005, in Section 11.2.8.2.4 Vertical Concrete Cask Sliding provided a calculation to verify that the acceleration g-value (or g-load) of 36.0 g in the collision of the two casks during the seismic events would not exceed the 40.0 g bounded for the design basis tip-over acceleration. Therefore, the design of the Catawba ISFSI concrete pads meets the general license requirements.

4OA6 Meetings, Including Exit

On October 5, 2006, the resident inspectors presented the inspection results to Mr. R. Repko and other members of licensee management, who acknowledged the findings. The inspectors confirmed that all proprietary information provided or examined during the inspection period had been returned.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
K. Adams, Human Performance Manager
E. Beadle, Emergency Planning Manager
S. Beagles, Chemistry Manager
E. Brewer, Operations Training Manager
W. Byers, Security Manager
J. Ferguson, Safety Assurance Manager
J. Foster, Radiation Protection Manager
W. Green, Reactor and Electrical Systems Manager
G. Hamrick, Mechanical, Civil Engineering Manager
D. Jamil, Catawba Site Vice President
R. Hart, Regulatory Compliance Manager
A. Lindsay, Training Manager
J. McConnell, Shift Operations Manager
J. Morris, Catawba Site Vice President
J. Pitesa, Station Manager
L. Reed, Modifications Engineering Manager
R. Repko, Engineering Manager
G. Spurlin, LOR Training Supervisor
C. Trezise, Operations Superintendent

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000413,414/2006004-01	NCV	Failure to adequately examine Senior Reactor Operators in accordance with 10 CFR 55.59 (Section 1R11.1)
05000413/2006004-02	NCV	Failure to Maintain Design Control Over Installation of Seals in Below-Grade Electrical Conduits (Section 4OA3.1)

Closed

05000413/2006009-03	URI	Review of Failure to Seal Conduits into Manholes and the 1A DG Room as Required by Design and Construction Documents (Section 4OA3.1)
05000413, 414/2006009-02	URI	Improper Relay Settings in the Catawba 230kv Switchyard Resulted in a Total Loss of Offsite Power Following the Failure of a PCB Current Transformer (Section 4OA3.2)
72-45/2005-001-01	URI	ISFSI Concrete Storage Pad Calculated

Seismic Acceleration Exceeds the General
License Limit (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Preparations
PIPs C-06-6192, C-06-6193, C-06-6197, C-06-6201, C-06-6198

Section 1R04: Equipment Alignment
Critical maintenance plan for the 2B DG maintenance activities
OP/2/A/6250/002, Rev. 123; Auxiliary Feedwater System, Enclosure 4.7 Valve Checklist
CN-2592-1.0, 1.1, and 1.2; Flow Diagrams of Unit 2 Auxiliary Feedwater System (CA)
CNS-1562.CA-00-0001, Rev. 35; Design Basis Specification for the Auxiliary Feedwater System (CA)
Catawba Technical Specifications - Plant Systems: 3.7.5 and 3.7.6
CA - Auxiliary Feedwater Health Report, 2006T1
UFSAR, Section 10.4.9
Operations Management Procedure 2-33, Rev. 35 Valve and Breaker Position Verification, Valve Operation and Pipe Cap Removal and Installation
PIPs C-04-3251, PIP C05-05511, C-04-6942

Section 1R05: Fire Protection
Pre-Fire Plan for Fire Strategy Area AZ; Transformer Yard Unit 2
Pre-Fire Plan for Fire Strategy Area 4; Auxiliary Building 543 foot elevation
Pre-Fire Plan for Fire Strategy Area 1; Auxiliary Building 522 foot elevation
Pre-Fire Plan for Fire Strategy Area 11; Auxiliary Building 560 foot elevation
Pre-Fire Plan for Fire Strategy Area 13; Auxiliary Building 577 foot elevation
Pre-Fire Plan for Fire Strategy Area 34; Auxiliary Building 543 foot elevation, 1B Auxiliary Shutdown Panel
Pre-Fire Plan for Fire Strategy Area 32; Auxiliary Building 543 foot elevation, 1A Auxiliary Shutdown Panel

Section 1R06: Flood Protection
PIP C-06-4884
Electrical Equipment Layout Drawings - Outdoor Areas; CN-1938–01 through 07

Section 1R11: Licensed Operator Requalification
Written Examination Development, Administration, and Grading, Rev. 10
Licensed Operator Requal Active Simulator Exam Development Guide, Rev. 00
Operations Training Management Procedure 7.0, Rev. 9
JPM ADMIN Guide, Rev. 10
SIM ADMIN Simulator Exercise Guide, Rev. 33
Initial Licensed Operator Examination Security Procedure, Rev. 05
HLP Simulator Briefing Checklist, Rev. 08
2005 Annual LOR Exam Assessment
Operations Training Management Procedure 5.0, Program Evaluations, 5/2/06
Operations Training Management Procedure 6.0, Training Programs, 2/23/06

Nuclear System Directive 512, Maintenance of RO/SRO NRC Licenses
Reactivation Records: Appendix B.512
Badge Access Transaction Reports
Licensed Operator Medical Records
Remedial Training Records
Written Exams: LOR0614ERO, LOR0614ESO, LOR0614CR, LOR0614CSO
JPMs: OP-CN-CF-CF-001, OP-CN-MT-MT-006, OP-CN-PSS-RN-004, OP-CN-PS-NC-046
Simulator Scenarios: ASE-25 OP-CN-ASE-25, Rev. 17, ASE-40 OP-CN-ASE-40, Rev. 0, ASE-08 OP-CN-ASE-08, Rev. 25, ASE-16 OP-CN-ASE-16, Rev. 22, ASE-12 OP-CN-ASE-12, Rev. 23
Simulator Fidelity Documents:
Transient #1 S/G Tube Leak 2005 Annual testing, Rev. 09/08/04. Test performed 11/15/2005.
Transient #2 Large Break LOCA with Loss of Off site power. 2005 Annual testing, Rev. 09/08/04. Test performed 11/15/2005.
Transient # 10 PORV failure 2005 Annual testing, Rev. 09/09/04. Test performed 11/15/2005.
Transient # 11 Reactor Trip 2005 Annual testing, Rev. 09/09/04. Test performed 11/15/2005.
Transient #12 Steam Line Break 2005 Annual testing, Rev. 09/09/04. Test performed 11/15/2005.
Transient #13 Feed-water Line Break 2005 Annual testing, Rev. 09/09/04. Test performed 11/15/2005.
Simulator Core Reload and Normal Evolution Procedure, Rev. 12/01/04. Test performed 5/16/05.
Steady State Power Operation 2005, Rev. 13 08/07/04. Test performed 11/15/2005.
Reviewed all normal evolutions.
Simulator Work Requests: CAS-029, CVC-027, EHC-044, PCS-056, CFW-099, SGN-048, EHC-046, SYS-022
PIPs C-06-05105, PIP C-05-04104, PIP C-05-04471

Section 1R12: Maintenance Effectiveness
PIP C-06-5156
WOs 01703730 and WO 01703735
Modification CD 101024; Replace 30A Hydrogen Mitigation (EHM) breakers with 40A breakers
Modification CD 201025; Replace 30A EHM breakers with 40A breakers

Section 1R15: Operability Evaluations
Technical Specification Action Item Log entry C2-06-01808; 2B DG inoperable due to 2B2 Emergency Ventilation fan inoperable
PT/1/A/4400/003A; Component Cooling Train 1A Performance Test
TSAIL entry 06-1789; Unit 1 hydrogen igniter breaker tripping open

Section 1R19: Post-Maintenance Testing
PIP C-06-5156
WOs 01703730 and 01703735
Modification CD 101024; Replace 30A EHM breakers with 40A breakers
Modification CD 201025; Replace 30A EHM breakers with 40A breakers

Section 1R22: Surveillance Testing
Unit 1, Cycle 16 91-01 Pre-Job Briefing for End of Life Moderator Temperature Coefficient

Measurement

1EP6 Drill Evaluation
Catawba Emergency Response Organization Drill Scenario Guide 06-4
Catawba Emergency Response Organization Drill Scenario Guide 06-5

4OA5 Other
NRC Safety Evaluation Report and Certificate of Compliance No. 1015 for Docket No. 72-1015, NAC-UMS STORAGE SYSTEM Amendment Nos. 4 and 5
Calculation No. CNC-1140-04-00-0001, Rev. 0
NUREG/CR-6865
PIP C-04-06877