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

October 11, 2000

Duke Energy Corporation
ATTN: Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION - NRC INSPECTION REPORT 50-413/00-04, 50-414/00-04

Dear Mr. Peterson:

On September 23, 2000, the NRC completed an inspection at your Catawba Units 1 and 2. The enclosed report documents the inspection findings which were discussed on September 27, 2000, with you and other 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.

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

In accordance with 10 CFR 2.790 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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Charles R. Ogle, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket No.: 50-413, 50-414
License No.: NPF-35, NPF-52

Enclosure: Inspection Report 50-413/00-04, 50-414/00-04
w/Attached NRC's Revised Reactor Oversight Process

cc w/encl:
Regulatory Compliance Manager
Duke Energy Corporation
Electronic Mail Distribution

Lisa Vaughn
Legal Department (PB05E)
Duke Energy Corporation
422 South Church Street
Charlotte, NC 28242

Anne Cottingham
Winston and Strawn
Electronic Mail Distribution

North Carolina MPA-1
Electronic Mail Distribution

Henry J. Porter, Assistant Director
Div. of Radioactive Waste Mgmt.
S. C. Department of Health
and Environmental Control
Electronic Mail Distribution

cc w/encl: Continued see page 3

cc w/encl: Continued
R. Mike Gandy
Division of Radioactive Waste Mgmt.
S. C. Department of Health and
Environmental Control
Electronic Mail Distribution

Richard P. Wilson, Esq.
Assistant Attorney General
S. C. Attorney General's Office
Electronic Mail Distribution

Vanessa Quinn
Federal Emergency Management Agency
Electronic Mail Distribution

North Carolina Electric
Membership Corporation
Electronic Mail Distribution

Peggy Force
Assistant Attorney General
N. C. Department of Justice
Electronic Mail Distribution

County Manager of York County, SC
Electronic Mail Distribution

Piedmont Municipal Power Agency
Electronic Mail Distribution

L. A. Keller, Manager
Nuclear Regulatory Licensing
Duke Energy Corporation
526 S. Church Street
Charlotte, NC 28201-0006

Distribution w/encl:
P. Tam, NRR
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REGION II

Docket No:	50-413, 50-414
License No:	NPF-35, NPF-52
Report No:	50-413/00-04, 50-414/00-04
Licensee:	Duke Energy Corporation
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	422 South Church Street Charlotte, NC 28242
Dates:	June 25 - September 23, 2000
Inspectors:	D. Roberts, Senior Resident Inspector M. Giles, Resident Inspector G. Hopper, Senior Operations Engineer (Section 1R11.2) J. Kreh, Emergency Preparedness Inspector (Sections 1EP1, 4OA1.3 - .5) W. Sartor, Senior Emergency Preparedness Inspector (Sections 1EP1, 4OA1.3 - .5) M. Shannon, Senior Resident Inspector, Oconee (Section 1R04) F. Wright, Senior Health Physicist (Sections 2OS1, 2OS3, 4OA1.6)
Approved by:	C. Ogle, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000413-00-04, IR 05000414-00-04, on 06/25 - 09/23/2000, Duke Energy Corporation, Catawba Nuclear Station, Units 1 & 2 - Initiating Events and Occupational Radiation Safety

The inspection was conducted by resident inspectors, regional emergency preparedness inspectors, a regional health physics inspector, and a regional operations engineer. The inspection identified two Green findings, one of which involved two non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using the Significance Determination Process (SDP) found in Inspection Manual Chapter 0609. Findings to which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

Cornerstone: Initiating Events

- Green. Poor workmanship and inadequate oversight of turbine building roof repairs, coupled with inadequately constructed roof drainage systems, resulted in a June 5, 2000, Unit 2 reactor trip. Water from heavy rains that day could not be properly drained from the turbine building roof, partially due to debris and other roofing material that had collected in the drainage system. Water overflowed from the roof and into the turbine building, and leaked into the 2B main feedwater pump turbine speed control cabinet. A secondary plant transient resulted, which ultimately led to a turbine trip/reactor trip. This issue was determined to be of very low safety significance because it did not affect the ability of mitigating systems to perform their safety functions (Section 4OA3.1).

Cornerstone: Occupational Radiation Safety

- Green. A single event, resulting in two non-cited violations, involved: (1) a failure to implement radiation control procedures for posting an extra high radiation area as required by TS 5.4.1.a.; and (2) failure to lock or control entrance to an extra high radiation area as required by Technical Specification 5.7.2 and Title 10 CFR Part 20.1601. This event was determined to be of very low safety significance because minimal radiation exposure was received by the workers and inadvertent entry into the area of concern (i.e., containment building in the area near the personnel air lock) would not immediately result in workers being in radiation fields greater than 1000 milliroentgen equivalent man per hour (Section 2OS1).

Report Details

Summary of Plant Status: Unit 1 was at 100 percent power throughout the inspection period, except for a brief period from July 7 to July 8, 2000, when reactor power was reduced to 97 percent to facilitate performance of end-of-cycle moderator temperature coefficient testing. On July 8, after moderator temperature coefficient testing was completed, the unit was further reduced to 88 percent power to facilitate main turbine control valve movement testing. The unit was returned to 100 percent power on July 9, 2000, following successful completion of the test, where it remained for the duration of the inspection period.

Unit 2 began the period at 100 percent power where it remained until an automatic power reduction to 97 percent occurred on September 10, 2000, as a result of the C and D Moisture Separator Reheater drain valves opening when the feeder breaker supplying their associated motor control center inadvertently tripped open. The unit was returned to 100 percent reactor power on September 11, 2000, and remained there until power was reduced to 87 percent to support turbine control valve testing on September 23, 2000. The unit returned to 100 percent power after successful valve testing and remained there through the end of the inspection period.

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors performed a review of the licensee's procedures and interviewed personnel to assess the plant's readiness for potential severe weather-related conditions during the current hurricane season. The licensee's procedures were verified to contain appropriate entry conditions and precautionary measures for protecting equipment important to safety from the threat of tornados, hurricanes, and other severe weather conditions associated with the season. The inspectors verified that operators were cognizant of the adverse weather procedures and were tracking severe weather systems that threatened the area.

b. Issues and Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed partial walkdowns of the shared B train nuclear service water (RN) system, the 2A emergency diesel generator (EDG), and the Unit 1 A and B train motor-driven auxiliary feedwater (CA) systems to verify their availability while redundant system equipment was inoperable. In addition, the inspectors conducted a full system walkdown of the 2B emergency onsite alternating current (AC) power system to verify that components were properly operating, labeled, and in good working condition. Included in the scope of this full system walkdown were the major EDG components and support systems, such as lubricating oil, fuel oil, jacket cooling water, and starting air.

This inspection included a review of outstanding work requests and corrective action program documents to verify that the licensee was properly identifying and correcting system problems.

b. Issues and Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured six areas important to reactor safety to verify that combustibles and ignition sources were properly controlled, and that fire detection and suppression capabilities were intact. For areas where fire detection equipment was out of service, the inspectors verified that compensatory measures (i.e., fire watch tours) were properly implemented. For dry-pipe suppression systems, the inspectors verified that pre-fire plans specified proper steps for fire brigade personnel to activate the systems when needed. The inspectors selected the areas based on a review of the licensee's safe shutdown analysis, probabilistic risk assessment (PRA)-based 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. Areas toured this quarter included the Unit 1 A and B auxiliary shutdown panel rooms, the common RN pump intake structure, the auxiliary building in the vicinity of the component cooling water (KC) pumps where ongoing fire barrier repair work necessitated compensatory measures, the 4160 volt essential bus ETB switchgear room, and the A and B train EDG rooms for both units.

b. Issues and Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Quarterly Review

a. Inspection Scope

The inspectors observed a control room simulator training scenario on August 9, 2000, to assess licensed reactor operator and senior reactor operator performance. The training scenario involved a faulted steam generator (steamline break inside containment). The inspectors focused on the performance of the operators in implementing the emergency plan, plant procedures, and Technical Specifications (TS). The inspectors also observed the post-simulator critique to assess the licensee's ability to identify operator or simulator performance issues.

b. Issues and Findings

No findings of significance were identified.

.2 Biennial Review

a. Inspection Scope

The inspectors reviewed a segment of the licensee's biennial written examination and evaluated its effectiveness in providing a basis for evaluating operator knowledge of subjects covered in the requalification program. Examination quality, licensee effectiveness in incorporating plant and industry feedback into the training program, and examination development methodology were evaluated for compliance with guidelines contained in the Operations Training Management Procedures (OTMP) Sections 1 through 8. The inspectors observed the annual dynamic simulator examination for one shift of operators to evaluate the adequacy of licensee training on high risk operator actions. During these observations, the inspectors assessed licensee evaluator effectiveness in identifying operator performance deficiencies requiring supplemental training. The inspectors also evaluated and observed a portion of the walkthrough examination administered during the requalification segment.

The inspectors reviewed and evaluated the licensee's remedial training program for selected operator deficiencies identified during the previous requalification cycle. The inspectors also reviewed a sample of on-shift licensed operator qualification records to ensure compliance with 10 CFR 55.59, Requalification, and 10CFR 55.53, Conditions of License.

b. Issues and Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule (10 CFR 50.65) to determine whether responsible personnel were properly evaluating the effectiveness of maintenance on equipment important to safety. To this end, the inspectors verified that the licensee was properly classifying maintenance preventable functional failures (MPFFs). Systems, structures, and components (SSCs) were also reviewed for proper scoping and risk categorization within the licensee's tracking system. The inspectors conducted this inspection with respect to the six equipment issues/SSCs identified in the following Problem Investigation Process reports (PIPs) and/or other program documents:

PIP or program document	Equipment Problem
PIP C-00-01692	Valve 2CA-15 (RN suction transfer) failure during simulation of 2/3 loss of normal CA pump suction

PIP or program document	Equipment Problem
PIP C-00-02488	2A EDG load sequencer timer intermittent failures from Spring 2000
Various PIPs (listed at end of this report)	Evaluation of various MPFFs for SSCs under the “Containment Isolation At Power” supersystem against 10 CFR 50.65 (a)(1) performance monitoring criteria
Maintenance Rule Summary Sheet	Verification of accurate scoping for the containment chilled water SSCs
PIP C-00-03853	Valve 1CA-36 (D steam generator flow control from turbine-driven CA pump) manual loader power failure in auxiliary shutdown panel room
PIP C-99-04675	1B EDG failures from Fall 1999 - (This was re-evaluated after a new root cause determination was completed)

b. Issues and 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 of the risk impact of removing from service those components associated with the six emergent and planned work items listed below, focusing primarily on activities determined to be risk-significant within the maintenance rule. The inspectors also verified that the licensee adequately identified and resolved problems associated with maintenance risk assessment and emergent work. In addition to those items below, the inspectors reviewed the licensee’s assessment and corrective actions for a work implementation error that resulted in the unplanned unavailability of the 2A KC heat exchanger for four hours on August 9, 2000.

Component or System	Reason for Removal from Service
1RN-3A	Valve binding and not opening fully
Unit 2 Refueling Water Storage Tank (FWST) level channel 3	Channel failure; failed circuit boards
EDGs (all four)	Monthly air roll activities with petcock valves open
2B KC heat exchanger	Planned maintenance
PCB-13 switchyard breaker	Planned maintenance while EDG 1A was inoperable for testing

Component or System	Reason for Removal from Service
Unit 1 RN to CA Pipe Flushing	Planned maintenance

b. Issues and Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability determinations (or justifications for continued operation) to verify that safety system operability was properly established, that the affected component or system remained available to perform its intended safety function, and that no unrecognized increase in plant or public risk occurred. Operability evaluations were reviewed for the four issues described in the following PIPs:

PIP Number	Issue
C-00-04025	Potential failure mechanism identified for main feedwater isolation valves
C-00-04032	Inspector-identified external corrosion on RN piping and threaded fittings
C-00-03508	Effects of potential degraded voltage conditions on hydrogen ignitor system glow plug temperatures
C-00-03366	1RN-3A valve binding in which valve would not fully open remotely

b. Issues and Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed the list of operator workarounds in place during the week of July 16-22, 2000, to assess individual workarounds and determine their cumulative impact on plant risk. A specific item reviewed this quarter involved a problem adding water to the containment isolation valve injection water (NW) system surge tank from its normal demineralized water (YM) supply. Because of the problem, operators were required to depressurize the NW system before adding water from the YM system. The inspectors verified that the same problem would not prevent the dedicated RN emergency water supply from making up to the NW system's surge tank when called upon during an event.

b. Issues and Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors observed or reviewed post-maintenance tests associated with the following six work activities to verify 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.

Procedure/ Work Order (WO) Number	Maintenance/Test Activity
WO 98290692	Relay replacement in 1ATC-CA-EATC09
PT/2/A/4200/013C, Rev. 39	2RN-144A (inlet to A containment spray heat exchanger) valve stroke testing following actuator spring-pack replacement
WO 98202684	Damper testing following motor assembly replacement on damper 1DSF-D-1
IP/1/A/3170/003B, Rev. 010, and IP/2/A/3170/003B, Rev. 013	Hydrogen Mitigation System Train B Quarterly Check following fuse replacement (Unit 1 and 2)
WO 98296836	2B EDG lube-oil check valve failure
PT/1/A/4700/020, Rev. 4	Unit 1 turbine-driven CA pump sump level switch maintenance

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the six surveillance test procedures listed below to verify that TS requirements were properly incorporated and that test acceptance criteria were properly specified. The inspectors observed actual performance of some of the tests and reviewed completed procedures to verify that acceptance criteria had been met. The inspectors also verified that proper test conditions were established in the procedures and that no equipment preconditioning activities were being conducted.

Procedure Number	Title
PT/1/A/4450/003C, Rev. 39	Annulus Ventilation System Performance Test

Procedure Number	Title
PT/1/A/4450/005A, Rev. 41	Containment Air Return Fan 1A And Hydrogen Skimmer Fan 1A Performance Test
PT/2/A/4350/002B, Rev. 70	Diesel Generator 2B Operability Test
PT/2/A/4200/004C, Rev. 26	Containment Spray Pump 2B Performance Test
PT/2/A/4400/009, Rev. 28	Cooling Water Flow Monitoring for Asiatic Clams and Mussels Quarterly Test
IP/2/A/3200/017A, Rev. 1	Response Time Testing of Reactor Protection System (RPS) and Engineered Safety Feature (ESF) Loops, Channels 1 and 4 (Test Group 1 of 3)

b. Issues and Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed two temporary modifications this quarter to verify that the functions of important safety systems were not affected. In both cases, the modifications were developed to restore functions for systems that had been degraded by earlier events. The following modifications were reviewed:

Temporary Modification	Title or Description
CNTM-0046	Provided local control of breaker 1FTA-3 (blackout bus feeder breaker)
CNCE-61633	A train RN pit low water level setpoint change

b. Issues and Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP1 Exercise Evaluation

a. Inspection Scope

The inspectors reviewed the objectives and scenario for the Catawba Nuclear Station biennial, full-participation emergency preparedness exercise on July 25, 2000, to determine whether they were designed to suitably test major elements of the licensee's emergency plan. The criteria against which the exercise scenario, the licensee's performance, and the licensee's critique were evaluated were contained in Attachment 01 of Inspection Procedure 71114.

During the period of July 24-27, 2000, the inspectors observed and evaluated the licensee's performance in the exercise, as well as selected activities related to the licensee's conduct and self-assessment of the exercise. The exercise was conducted on July 25, 2000 from 6:30 a.m. to 12:30 p.m. Licensee activities inspected during the exercise included those occurring in the Control Room Simulator (CRS), Technical Support Center (TSC), Operational Support Center (OSC), and Emergency Operations Facility (EOF). The NRC's evaluation focused on the risk-significant activities of event classification, notification of governmental authorities, onsite protective actions, offsite protective action recommendations (PARs), and accident mitigation. The inspectors also evaluated command and control, the transfer of emergency responsibilities between facilities, communications, adherence to procedures, and the overall implementation of the emergency plan. The inspectors attended the post-exercise critique to evaluate the licensee's self-assessment process, as well as the presentation of critique results to plant management.

b. Observations and Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed a control room simulator training scenario on August 9, 2000, to assess licensed operators' performance in the area of emergency preparedness. The inspectors observed operators make an emergency declaration and verified that the correct declaration and associated followup actions had been performed in accordance with the licensee's procedures and regulatory requirements. The observed scenario (a main steamline break inside containment) was performed in conjunction with the licensed operator requalification program.

b. Issues and Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

This review was performed to verify licensee requirements and controls for high radiation areas (HRAs), extra high radiation areas (EHRAs) and very high radiation areas (VHRAs) were established and properly implemented. The inspector reviewed TS, applicable procedures, records, and conducted interviews with licensee personnel.

b. Issues and Findings

Section TS 5.7.2, described radiological protection controls and entrance requirements for areas having high radiation levels greater than 1000 milliroentgen equivalent man (mrem) per hour. The TS required that the licensee lock or continuously guard doors to prevent unauthorized entry.

Licensee procedure SH/0/B/2000/0012, “Access Controls for High, Extra High, and Very High Radiation Areas,“ Revision 001, established and defined the proper radiological protection controls and procedures for high radiation area classifications.

Section 4.1 defined an EHRA as any area accessible to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent in excess of 1000 mrem in one hour, but less than or equal to 500 rads in one hour, at 1 meter from the radiation source or from any surface that the radiation penetrates.

The inspector reviewed a recent event in which the above TS and licensee procedure requirements were not met. On the morning of May 10, 2000, mechanical maintenance (MM) personnel requested radiation protection (RP) permission to enter a posted and controlled EHRA. The workers were to install and remove a strong-back on the Unit 1 inner personnel air lock (PAL) door to lower containment (LC).

Licensee procedure HP/0/B/1000/061, “Containment Work,” Revision 2, provided radiation protection technicians (RPT) guidance for entering containment and performing various tasks. The procedure utilized enclosures as checklists to ensure adequate radiological controls were met. Enclosure 5.2, of SH/0/B/2000/0012, “Power Entry Into Annulus,” was used by the staff for controlling work in the LC/PAL. Enclosure 5.2 step 6 required, “If working in lower containment airlock, post on the inner airlock door as an EHRA with a flashing yellow light.”

While the reactor is at power, the area outside the PAL is posted as an EHRA and locked. The licensee did not have a means to lock the LC/PAL doors that would prevent personnel from entering LC.

The RP shift technician responsible for specifying the radiological controls on Enclosure 5.2 marked step 6 as not applicable (NA). The RPT dispatched to unlock the iron bar door leading to the PAL did not move the EHRA posting from the bar door to the inner airlock door and did not place a flashing yellow light at the inner door. The technician left the area. The MM personnel were left to work in a posted EHRA without RPT surveillance and controls. During the periods before and after the strong back was installed and removed, workers only had to open the inner PAL door to gain uncontrolled access to the EHRA in LC. The unattended workers reported that they had not opened the inner PAL door and the workers received minimal external dose during the job.

The failure to post an EHRA in accordance with the requirements of licensee procedure HP/0/B/1000/061, “Containment Work,” Revision 2, was considered to be a violation of TS 5.4.1.a (Regulatory Guide 1.33, Appendix A, Section 7.e, Radiation Protection Procedures). This issue is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the enforcement policy and is identified as NCV 50-413/00-04-01: Failure to Implement Radiation Control Procedures for Posting Extra High Radiation Areas as Required by TS 5.4.1.a. This violation is in the licensee’s corrective action program as PIP C-00-02492.

The failure to lock or control entrance to an EHRA was considered a violation of TS 5.7.2 and Title 10 CFR Part 20.1601 requirements. This issue is being treated as a NCV, consistent with Section VI.A of the enforcement policy and is identified as NCV 50-413/00-04-02: Failure to Control Access to High Radiation Areas as Required by 10 CFR Part 20.1601 and TS 5.7.2. This violation is in the licensee’s corrective action program as PIP C-00-02492.

The risk significance of the single event resulting in these NCVs has been evaluated to be Green using the Occupational Safety Significance Determination Process. This event was determined to be of very low safety significance because minimal radiation exposure was received by the workers and inadvertent entry into the containment building in the area near the personnel air lock would not immediately result in workers being in radiation fields greater than 1000 mrem per hour.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector evaluated the accuracy and operability of plant radiation monitoring instruments used for the protection of occupational radiation workers. The reviews included the operability of plant area radiation monitors (ARMs) identified in the Updated Final Safety Analysis Report. The inspector observed selected ARM equipment material conditions and verified local, remote and control room radiation monitor readouts were in agreement with the radiation levels at the detectors measured by the inspector.

The availability and operability of portable radiation monitoring instruments for use in high radiation areas were evaluated.

The inspector evaluated the adequacy of the licensee’s self-contained breathing apparatus (SCBA) respiratory protection program for providing SCBAs to radiation workers entering areas of unknown radiological conditions and areas where the atmosphere could be immediately dangerous to life and health.

The inspector verified the licensee had reviewed and evaluated, the personnel safety issues documented in NRC Information Notices (IN) 98-20, “Respiratory protection Program and IN 99-05, “Inadvertent Discharge of Fire Protection System and Gas Migration.”

b. Issues and Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Initiating Events and Mitigating Systems PIs

a. Inspection Scope

The inspectors conducted annual reviews of the following three Reactor Safety PIs, as submitted to the NRC by the licensee, for accuracy:

Cornerstone	PI
Initiating Events	Scrams with a Loss of Normal Heat Removal
Initiating Events	Unplanned Scrams per 7,000 Critical Hours
Mitigating Systems	Safety System Unavailability - Emergency AC Power System

To verify the PI data, the inspectors reviewed control room logs, Technical Specification Action Item Log (TSAIL) entries, work management system data, and maintenance rule data.

b. Issues and Findings

There were no findings of significance identified for the “Safety System Unavailability - Emergency AC Power System” and “Unplanned Scrams per 7,000 Critical Hours” performance indicators. However, for the “Scrams with Loss of Normal Heat Removal” indicator, the inspectors identified that the licensee did not count the June 5, 2000, Unit 1 reactor trip (also discussed in Section 4OA3.1 of this report), which was caused by a degraded 2B main feedwater pump speed control system as a result of an electrical cabinet which was exposed to rainwater. Speed control circuit card failures required operators to take the feedwater pump master controller to “manual” operation when both the 2A and 2B pumps’ speeds began experiencing large fluctuations. With both main feedwater pumps in manual and their flow regulating valves wide open, steam generator water levels increased to their hi-hi actuation setpoints before operators could reduce flow, which caused a feedwater isolation and turbine trip signal. The CA pumps automatically started (with both main feedwater pumps tripped) and continued to feed the steam generators until troubleshooting and repairs to the main feedwater system were completed approximately a day and a half later. The licensee continued to use the CA pumps during this time because the extent of damage to the main feedwater pumps and their speed control circuitry was not immediately apparent. It was determined hours after the transient that the 2A feedwater pump controller had not been affected, but the licensee cautiously chose to continue using auxiliary feedwater until any potential adverse main feedwater pump interactions and speed control problems were conclusively ruled out.

This PI monitors those reactor trip events that necessitate the use of mitigating systems (i.e., CA) and are therefore more risk-significant than uncomplicated trips. Those situations in which normal heat removal systems are isolated by design features or operator actions in order to control the reactor cooldown rate are excluded from this PI. The steam generators continued to be fed by CA pumps until after troubleshooting determined that the 2A main feedwater pump was reliable and the 2B main feedwater pump control circuitry needed repairs. Because the main feedwater system was degraded, which ultimately caused the reactor trip, the inspectors preliminarily concluded that the June 5, 2000, occurrence did not meet the exception criteria and should have been counted against the PI. However, a formal determination of whether the June 5, 2000, reactor trip should be counted had not been obtained from the NRC Office of Nuclear Reactor Regulation (NRR) by the close of the inspection period. Per guidance in Temporary Instruction (TI) 2515/144, “Performance Indicator Data Collecting and Reporting Process Review,” the inspectors are opening an Unresolved Item (URI) pending formal resolution by NRR. It is identified as URI 50-414/00-04-03: Minor PI Discrepancy Associated with the “Scram with Loss of Normal Heat Removal” indicator (June 5, 2000, Unit 2 reactor trip).

.2 Potential Discrepancy with Safety System Unavailability - CA System PI

a. Inspection Scope

The inspectors also reviewed Safety System Unavailability for the Unit 2 CA system to verify whether fault exposure hours associated with an A-train motor-driven CA pump suction valve (2CA-15A) failure had been properly counted. The inspectors became aware of this valve failure from a discussion with the system engineer and a review of PIP C-00-01692, which documented the failure.

b. Issues and Findings

Valve 2CA-15A is a normally closed motor-operated valve that automatically opens to provide an emergency supply of raw water to the suction of the 2A motor-driven CA pump upon loss of suction pressure (usually associated with a loss or depletion of its normal condensate-grade water supply). The valve failed to open on March 30, 2000, during a test in which a low suction pressure signal was simulated. It was later determined by the licensee that faulty electrical contacts on a circuit relay for the valve were not contacting properly, and the valve did not receive a signal to open. The relay was replaced and the valve subsequently passed its surveillance test.

After reviewing the failure and taking credit for a control room switch that would have provided operators with remote manual operation of the valve if it had failed to automatically open during an event, the licensee decided not to report the fault exposure hours for this test failure.

The inspectors inquired of NRR as to whether or not the licensee’s approach was valid.

The inspectors were informed that operator action was not to be credited for equipment test failures when determining fault exposure hour applicability. However, because this conflicted with guidance in NEI 99-02, Rev. 0 (page 29, lines 5, 6, and 7), the inspectors submitted a feedback request to NRR for formal interpretation and clarification of this issue. Per guidance in TI 2515/144, the inspectors are opening a URI to track this issue. It is identified as URI 50-414/00-04-04: Potential Discrepancy with the “Safety System Unavailability - Auxiliary Feedwater System” PI (2CA-15A valve failure).

.3 Emergency Response Organization (ERO) Drill/Exercise Performance PI

a. Inspection Scope

The inspectors assessed the accuracy of the PI for ERO drill and exercise performance (DEP) over the past eight quarters through review of a sample of drill records. Detailed documentation for drills conducted in June 1999, January 2000, and March 2000 was reviewed to verify the licensee’s reported data regarding successes in emergency classifications, notifications, and PARs. The inspectors concurred in the licensee’s determination of 8 successes out of 10 opportunities for the DEP PI during the exercise on July 25, 2000.

b. Issues and Findings

No findings of significance were identified.

.4 ERO Drill Participation PI

a. Inspection Scope

The inspectors assessed the accuracy of the PI for ERO drill participation during the previous eight quarters by selective review of the training records for the 191 personnel assigned to key positions in the ERO. Drill participation was verified by reviewing training attendance records for approximately 10 percent of key ERO personnel against the drill/event participation matrix for specific drill dates.

b. Issues and Findings

No findings of significance were identified.

.5 Alert and Notification System Reliability PI

a. Inspection Scope

The inspectors assessed the accuracy of the PI for the alert and notification system reliability through review of the licensee’s records of the siren tests for the previous 12 months. A sample of records for the weekly silent tests, weekly low-growl tests, and quarterly full-cycle tests was reviewed.

b. Issues and Findings

No findings of significance were identified.

.6 Occupational Radiation Safety and Public Radiation Safety Cornerstone PIs

a. Inspection Scope

The inspectors reviewed Duke Power Nuclear Policy Manual, Nuclear System Directive, 225 “NRC Performance Indicators,” revision 0. The inspectors independently reviewed data supporting the Occupational Radiation Safety and Public Radiation Safety Cornerstone PIs listed in the licensee’s PI data during the period of April 1 to August 4, 2000

b. Issues and Findings

No findings of significance were identified.

.7 (Closed) URI 50-413,414/00-03-04: Minor Discrepancy Involving the Calculation of Reactor Coolant System Specific Activity Performance Indicator. This item was opened to track a minor discrepancy involving the licensee’s use of a non-conservative reactor coolant system dose equivalent iodine TS limit (1.0 micro-Curies per gram) as the denominator in the specific activity PI calculation instead of the more appropriate administrative limit (0.064 micro-Curies per gram) that had been imposed by the licensee due to design basis constraints. The licensee has since revised the first quarter and submitted the second quarter PI data, both of which used the more appropriate administrative limits in the denominator. This item has been resolved.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 50-414/00-003-(00,01): Reactor Trip Caused by Moisture Intrusion into Main Feedwater Pump 2B Speed Control Circuitry, and Operation Prohibited by Technical Specification 3.3.2.

Unit 2 Reactor Trip

This event was discussed in NRC Inspection Report 50-413/414-00-03. The licensee determined this reactor trip to be caused by inadequate oversight of turbine building roof work conducted under modification CN-61511. During this activity, workers did not contain or properly dispose of roofing material and debris. This allowed the turbine building roof gutters and drainage system to become clogged, leading to an overflow of rainwater into the turbine building during a heavy rainstorm on June 5, 2000. The same rainwater entered the 2B main feedwater pump turbine (CFPT) speed control cabinet causing a steam generator water level transient, which resulted in a Unit 2 turbine trip and subsequent reactor trip. PIP C-00-02872 was generated to document this event. After discussions with licensee management, the inspectors learned that the roof was considered to be poorly constructed, in that water that had collected on the roof was able to leak down the interior walls of the turbine building and onto the 2B CFPT speed control cabinet one elevation below. The inspectors concluded that the inadequate roof drainage system construction and/or poor workmanship associated with the roofing modification represented a performance issue that ultimately resulted in an initiating event (reactor trip). However, this issue did not result in the inability of safety systems to perform their intended functions. Therefore, this issue was screened as one of very low safety significance (Green) in Phase 1 of the Significance Determination Process.

Operation Prohibited By TS 3.3.2

After the unit was restarted following the reactor trip, the licensee began troubleshooting and repair activities on the 2B CFPT speed control circuitry. After repairs, the licensee began calibrating the 2B CFPT speed control circuitry in accordance with instrumentation procedure IP/2/B/3226/001B, Revision 010, Calibration Procedure For CFPT 2B Turbine Governor. During this calibration, the technicians incorrectly placed the 2B CFPT trip logic circuitry in the “reset” condition eight times. This occurred because a jumper was not installed to maintain a continuous trip signal as required in Mode 1 while the pump was in the shutdown condition. A note preceding step 10.18.4 of the procedure stated that placement of the jumper would simulate a CFPT 2B trip. However, step 10.18.4 itself read, “If Control Room SRO wants actions listed on Enclosure 11.3 *prevented*, install jumpers between the following links....” The enclosure listed the CA auto start and main turbine trip associated with the loss of both feedwater pumps as being among those actions affected by the CFPT trip signal. Having the CFPT trip signal reset while the pump was still shut down placed the unit in a configuration in which all three trip channels for the 2B CFPT were simultaneously inoperable (i.e., not capable of satisfying the CFPT trip logic necessary to provide a CA auto start or main turbine trip and feedwater isolation upon loss of the remaining 2A CFPT). The TS Limiting Condition for Operation (LCO) associated with the CFPT trip circuitry (TS LCO 3.3.2, Table 3.3.2-1, items 5.f and 6.e) did not have an associated action for more than one inoperable channel, which meant that Unit 2 was in TS 3.0.3 unbeknownst to control room operators. The eight reset occurrences all happened over brief periods not exceeding the time required by TS 3.0.3 to have the unit, which was operating in Mode 1, placed in Hot Standby; therefore, no TS LCO violation occurred.

The licensee determined the root cause of the CA pump auto start logic issue to be inadequate communication between station groups. The inspectors, after reviewing the calibration procedure, determined that the procedure was inadequate because it had misleading or conflicting information regarding the impact of the jumper installation on the TS-required CFPT trip logic circuitry. Failure to provide an adequate procedure for maintaining engineered safety feature actuating system functions 5.f and 6.e operable while calibrating the 2B main feedwater pump circuitry was contrary to TS 5.4.1 and Regulatory Guide 1.33 (item 9.a). The inspectors concluded that this non-compliance had no actual or credible impact on plant safety due to the availability of other trip features (i.e., steam generator low-low level and subsequent reactor/turbine trip signals) for which credit is taken in the plant’s design basis accident analysis. Thus, this noncompliance constitutes a violation of minor significance and is not subject to formal enforcement action. The LER is closed.

.2 (Closed) LER 50-413/00-004-00: Operation Prohibited by Technical Specification 3.7.12

Due to Inadequate Control of the Auxiliary Building Filtered Ventilation Exhaust (VA) System Pressure Boundary. The inspectors determined that this issue had minor significance because an evaluation completed by the licensee concluded that the Unit 1 VA system would still have filtered any potential emergency core cooling system leakage, which in itself was minimal. In addition, no violation of TS 3.0.3 occurred. This LER is closed.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. Gary Peterson, Site Vice President, and other members of licensee management at the conclusion of the inspection on September 27, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

E. Beadle, Emergency Preparedness Manager
R. Beagles, Safety Review Group Manager
M. Boyle, Radiation Protection Manager
G. Gilbert, Regulatory Compliance Manager
R. Glover, Operations Superintendent
P. Grobusky, Human Resources Manager
P. Herran, Engineering Manager
R. Jones, Station Manager
R. Parker, Maintenance Superintendent
G. Peterson, Catawba Site Vice President
F. Smith, Chemistry Manager
R. Sweigart, Safety Assurance Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-414/00-04-03	URI	Minor PI Discrepancy Associated with the “Scram with Loss of Normal Heat Removal” Indicator (June 5, 2000, Unit 2 reactor trip) (Section 4OA1.1)

50-414/00-04-04	URI	Potential Discrepancy with the “Safety System Unavailability - Auxiliary Feedwater System” PI (2CA-15A valve failure) (Section 4OA1.2)

Opened and Closed During this Inspection

50-413/00-04-01	NCV	Failure to Implement Radiation Control Procedures for Posting Extra High Radiation Areas as Required by TS 5.4.1.a (Section 2OS1)
50-413/00-04-02	NCV	Failure to Control Access to High Radiation Areas as Required by 10 CFR Part 20.1601 and TS 5.7.2 (Section 2OS1)

Previous Items Closed

50-413,414/00-03-04	URI	Minor Discrepancy Involving the Calculation of Reactor Coolant System Specific Activity Performance Indicator (Section 4OA1.7)
50-414/00-003-(00,01)	LER	Reactor Trip Caused by Moisture Intrusion into Main Feedwater Pump 2B Speed Control Circuitry, and Operation Prohibited by Technical Specification 3.3.2. (Section 4OA3.1)
50-413/00-004-00	LER	Operation Prohibited by Technical Specification 3.7.12 Due to Inadequate Control of the Auxiliary Building Filtered Ventilation Exhaust System Pressure Boundary (Section 4OA3.2)

Discussed

None

DOCUMENTS REVIEWED

PIPS required for evaluation of various MPFFs for SSCs under the “Containment Isolation at Power” supersystem against 10 CFR 50.65 (a)(1) Performance Monitoring Criteria (Section 1R12)

PIP C-98-03149
PIP C-98-03191
PIP C-98-03252
PIP C-98-03835

PIP C-98-03974
PIP C-98-04569
PIP C-99-00199
PIP C-99-00597
PIP C-99-01497
PIP C-99-01675
PIP C-99-01747
PIP C-99-01993
PIP C-99-02405
PIP C-99-02795
PIP C-99-03033
PIP C-99-03132
PIP C-99-03738
PIP C-99-04897
PIP C-00-02068

LIST OF ACRONYMS USED

AC	-	Alternating Current
ARM	-	Area Radiation Monitor
CA	-	Auxiliary Feedwater
CFPT	-	Main Feedwater Pump Turbine
CFR	-	Code of Federal Regulations
CRS	-	Control Room Simulator
DEP	-	Drill and Exercise Performance
EDG	-	Emergency Diesel Generator
EHRA	-	Extra High Radiation Area
EOF	-	Emergency Operations Facility
ERO	-	Emergency Response Organization
ESF	-	Engineered Safety Feature
FWST	-	Refueling Water Storage Tank
HRA		High Radiation Area
IN	-	Information Notice
IP	-	Instrumentation Procedure
KC	-	Component Cooling Water
LC	-	Lower Containment
LCO	-	Limiting Condition for Operation
LER	-	Licensee Event Report
MM	-	Mechanical Maintenance
MPFF	-	Maintenance Preventable Functional Failure
MREM	-	Milliroentgen Equivalent Man
NA	-	Not Applicable
NCV	-	Non-Cited Violation
NEI	-	Nuclear Energy Institute
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
NW	-	Containment Isolation Valve Injection Water
OSC	-	Operational Support Center
OTMP	-	Operations Training Management Procedures

PAL	-	Personnel Air Lock
PAR	-	Protective Action Recommendations
PI	-	Performance Indicator
PIP	-	Problem Investigation Process
PRA	-	Probabilistic Risk Assessment
RN	-	Nuclear Service Water
RP	-	Radiation Protection
RPS	-	Reactor Protection System
RPT	-	Radiation Protection Technicians
SCBA	-	Self-Contained Breathing Apparatus
SDP	-	Significance Determination Process
SRO	-	Senior Reactor Operator
SSC	-	Structures, Systems and Components
TI	-	Temporary Instruction
TS	-	Technical Specification
TSAIL	-	Technical Specification Action Item Log
TSC	-	Technical Support Center
URI	-	Unresolved Item
VA	-	Auxiliary Building Ventilation
VHRA	-	Very High Radiation Area
WO	-	Work Order
YM	-	Demineralized Water

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.