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

April 22, 2002

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 INTEGRATED INSPECTION REPORT 50-413/01-07, 50-414/01-07

Dear Mr. Peterson:

On March 23, 2002, the NRC completed an inspection at your Catawba Nuclear Station. The enclosed report documents the inspection findings which were discussed on March 25, 2002, 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 did not identify any findings of significance.

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/reading-rm/adams.html* (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: NRC Integrated Inspection Report 50-413/01-07, 50-414/01-07
w/Attachment - Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-413, 50-414
License Nos:	NPF-35, NPF-52
Report No:	50-413/01-07, 50-414/01-07
Licensee:	Duke Energy Corporation
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	4800 Concord Road York, SC 29745
Dates:	December 23, 2001 - March 23, 2002
Inspectors:	D. Roberts, Senior Resident Inspector M. Giles, Resident Inspector Jerry Blake, Senior Project Manager (Sections 1R02 and 1R17) Rich Chou, Reactor Inspector (Sections 1R02 and 1R17) Norman Garrett, Resident Inspector - Hatch (Sections 1R02 and 1R17) Edwin Lea, Project Engineer (Sections 1R06, 1R22, and 4OA3.3) Curt Rapp, Senior Project Engineer (Sections 1R11, 1R13, 1R19, 1R22, and 4OA3.2)
Approved by:	R. Haag, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000413-01-07, IR 05000414-01-07, on 12/24–3/23/2002, Duke Energy Corporation, Catawba Nuclear Station, Units 1 and 2. Quarterly Integrated Resident Inspector report.

The inspection was conducted by three resident inspectors and four inspectors from the regional office. 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. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website.

A. Inspector Identified Findings

The inspectors did not identify any findings of significance.

B. Licensee Identified Violations

A violation of very low significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. The violation is listed in section 40A7 of this report.

Report Details

Summary of Plant Status: Unit 1 operated at 100 percent power throughout the inspection period, except for a brief period on January 10, 2002, when reactor power was reduced to 97 percent to support planned testing of the 1B steam generator (SG) power operated relief valve (PORV); and from January 17 to January 19, 2002, when reactor power was reduced to 87 percent to facilitate the end-of-cycle measurement of moderator temperature coefficient and main turbine control valve movement testing.

Unit 2 operated at 100 percent power throughout the inspection period, except for brief periods on February 7 and February 20, 2002, when reactor power was reduced to 97 percent to facilitate the planned testing of the 2B and 2C SG PORVs, respectively; and from March 22 to March 23, 2002, to allow main turbine control valve movement testing.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed selected samples of safety evaluations to verify that the licensee had appropriately considered the conditions under which changes to the facility or procedures may be made, or tests conducted, without prior NRC approval. The inspectors reviewed safety evaluations for ten design and procedure changes. The inspectors assessed, through review of additional information, such as calculations, supporting analyses and drawings, whether the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The ten safety evaluations reviewed are in the List of Documents Reviewed of the Attachment to this report.

The inspectors also reviewed samples of design/engineering packages and procedure changes for which the licensee had determined that evaluations were not required, and verified that the licensee's conclusions to "screen out" these changes were correct and consistent with 10 CFR 50.59. The twelve "screened out" changes reviewed are in the List of Documents Reviewed of the Attachment to this report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed partial walkdowns of the following equipment: Unit 1 SG PORVs 1A, 1C, and 1D, and associated nitrogen supply systems while the 1B SG PORV was out of service; key components of the high head safety injection and component cooling water systems on both units while the standby shutdown facility diesel generator was out of service; and the 1A emergency diesel generator (EDG) and support systems while the 1B EDG was out of service. These partial walkdowns were conducted to verify the availability of redundant or diverse systems and components during periods when safety equipment was inoperable due to planned maintenance. The walkdowns were performed to ensure that proper levels of defense-in-depth were maintained. As part of a periodic review of the licensee's identification and resolution of problems in this area, the inspectors reviewed Problem Identification Process report (PIP) C-02-01236, which documented a scheduling problem that nearly resulted in personnel taking the 2A EDG out of service while the 1A nuclear service water (RN) essential header was isolated for maintenance. Removing the 2A EDG from service would have required isolating other components that were in service to support the 1A RN work, which would have resulted in unanticipated RN train unavailability for both units.

b. 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 fire 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 patrols) were properly implemented. The inspectors verified that hot work permits and fire watches were established in accordance with the licensee's requirements at selected times when pipe-cutting, grinding, or welding occurred near safety related equipment. 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 included the Unit 2 turbine building near 6.9/4.16 kilovolt (kV) transformers; Unit 1, 4.16 kV essential bus ETA area; Unit 2, 4.16 kV essential bus ETA area; Unit 1, 125 Volts direct current (Vdc) vital bus area; the auxiliary building, elevation 577 (common to both units) while hot work was conducted on nuclear service water (RN) piping; and the auxiliary building at various elevations common to both units to verify key valve positions in the backup fire suppression header water supply (with valve RY-23 degraded). As part of a periodic review of the licensee's identification and resolution of problems in the fire protection area, the inspectors reviewed PIP C-01-01954, which documented the problem with valve RY-23 and the licensee's inability to complete a three-year flow test for the fire suppression water system.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors performed a walkdown of selected areas of the plant, interviewed licensee personnel and reviewed documentation to assess the effectiveness of the licensee's flood mitigation program. The systems, components or areas selected for review/inspection were chosen based on risk significance. Areas inspected included the turbine-driven auxiliary feedwater pump rooms, auxiliary shutdown panel rooms, selected areas of the turbine building and the EDG rooms. The inspectors evaluated the material condition of flood protection related components for the selected areas. The evaluation included physical inspections of the selected components, a review of plant procedures that controlled maintenance activities associated with the selected components, and the work request/surveillance under which required activities were performed. The inspectors also reviewed documentation to determine the adequacy of the licensee's expected response to potential flooding issues.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed a control room simulator training scenario on February 13, 2002. The training scenario involved the loss of the residual heat removal (ND) system under three different plant shutdown conditions: reactor coolant (NC) system loops not filled, NC loops filled with reactor vessel head off, and NC system vacuum refill in progress. The purpose of the training scenario was to demonstrate the quickest method to restore ND. Following the simulator scenario, the inspectors observed the critique conducted by training instructors to assess their ability in identifying operator or simulator performance deficiencies. As part of a periodic review of the licensee's identification and resolution of problems in this area, the inspectors reviewed PIP C-02-00611, which documented a potential simulator examination compromise situation.

b. 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. The inspectors verified that the licensee was properly classifying maintenance preventable functional failures. For those systems, structures, and components (SSCs) that were categorized as 10 CFR 50.65 (a)(1) due to previous performance problems, the inspectors reviewed corrective action documents to verify that the licensee had identified causal factors, recommended appropriate corrective actions, and established reasonable performance goals for the SSCs. Some SSCs were also reviewed for proper maintenance rule scoping and risk categorization within the licensee's Maintenance Rule Scoping Summary document. The inspectors reviewed the following Problem Investigation Process reports (PIPs) for this inspection:

PIP number	Equipment Problem
C-01-03993	Control Room Ventilation System low intake flow conditions
C-01-04455	1C SG PORV inoperable due to low nitrogen pressure
C-01-04981	Ice buildup on Unit 1 ice condenser intermediate deck door in bay 9
C-01-05616	Nitrogen supply valves found closed for the 2D SG PORV
C-01-06382	Unit 2 auxiliary feedwater flow control valve controllers found out of position on auxiliary shutdown panels
C-01-06462	120 Volts alternating current vital inverter EID failure

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 of the risk impact of removing from service those components associated with the 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.

Component or System	Reason for Removal from Service
1B KC heat exchanger	Planned heat exchanger cleaning
2A CA Pump	Failed sliding link associated with several key functions
Standby shutdown facility (SSF)/ valve 1NM-3A	Valve failure to close following sampling activities; (designed to auto-close on transfer to SSF)
2A EDG	Turbocharger failure during surveillance

Component or System	Reason for Removal from Service
1A RN header/ control room ventilation chilled water (YC) system chiller	Planned 1A YC chiller/ RN pipe replacement
1A EDG	Several planned work activities (including five-year preventive maintenance)

As part of a periodic review of the licensee's identification and resolution of problems in the maintenance risk assessment area, the inspectors also reviewed the licensee's disposition of PIP C-02-00326, which was generated after the inspectors questioned the apparent inconsistent probabilistic risk assessment (PRA) coding of work activities associated with auxiliary safeguards testing (ASG). The inspectors questioned the non-coding of activities associated with ASG testing of the 1A EDG in the online risk assessment management program. The inspectors noted that similar work had been assigned a PRA code a week earlier. After reviewing this matter further, the licensee determined that the associated work did not render the 1A EDG unavailable, and therefore, did not require PRA coding.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspectors observed and reviewed plant performance during the conduct of the Unit 1 end-of-cycle moderator temperature coefficient measurement, which included a planned reduction in average NC system temperature by six degrees while the unit was at full power. These reviews were conducted to determine if operator actions were appropriate and in accordance with plant procedures and training. The inspectors also reviewed the controlling procedure, PT/O/A/4150/012B, Rev. 12, Moderator Temperature Coefficient of Reactivity Measurement [End of Life], for adequacy.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

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

PIP Number	Issue
C-01-01954	Operability of the fire suppression water system after leaking valve RY-23 prevented successful performance of three-year test
C-01-05303, 5390, 5586, and 5923	Recurring heat trace issue associated with main steam/auxiliary steam supply piping to the Unit 1 turbine-driven CA pump
C-01-06297	Justification for operation with known leakage by main feedwater regulating bypass valves, potentially impacting offsite dose analysis calculations for SG tube rupture events
C-02-00786	Operability of the 2A EDG after a starting air solenoid valve failed to close in 3 seconds
C-02-00788	Evaluation of the impact of Westinghouse Nuclear Safety Advisory Letter 02-3, which described a previously-unaccounted-for pressure drop across the SG mid-deck plates, resulting in non-conservative reactor trip setpoints for Unit 2
Not Applicable	Operability of VA system (both units/trains) following significant painting in the auxiliary building at elevation 522 in January 2002

As part of the inspectors periodic review of the licensee's own identification and resolution of problems in this area, the inspectors reviewed PIP C-02-00265, which identified that the "Operability Notification Forms," (Appendix E to Nuclear Station Directive 203, Operability, Rev. 14) book used by the Operations Shift Managers was not up to date, and that certain operability evaluations that were in progress were not reflected in the book.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed the list of operator workarounds in place to assess individual workarounds and determine their cumulative impact on plant risk. The inspectors also reviewed the list of "operable but degraded" equipment to determine if any of the listed problems resulted in or constituted operator workarounds, and what their impact on overall plant risk was. For this inspection period, the inspectors expanded their review to include Operations Information Notices (located in the main control room). This information was reviewed to determine if any of the documented items involved significant workarounds, which may not have been considered as such or figured into the licensee's own cumulative assessment of the impact of workarounds. One of the items considered to be an operator workaround by the inspectors, but not logged as such by the licensee, involved the repeated draining of water from the Unit 2 turbine-driven CA pump alternate steam supply piping. This item was later addressed by a temporary modification, which minimized the draining required. The inspectors reviewed this item to verify that it would not have prevented the CA system from performing its intended function, and that it did not significantly detract from the operators' ability to safely operate the plant.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated design change packages for twelve modifications, in all three cornerstone areas, to verify that the modifications did not degrade system availability, reliability, or functional capability. The inspectors verified inspection procedure attributes such as: energy requirements can be supplied by supporting systems; materials and replacement components were compatible with physical interfaces; replacement components were seismically qualified for application; Code and safety classification of replacement system, structures, and components were consistent with design bases; modification design assumptions were appropriate; post-modification testing would established operability; failure modes introduced by the modification were bounded by existing analyses; and that appropriate procedures or procedure changes had been initiated. For selected modification packages, the inspectors verified that the as-built configuration accurately reflected the design documentation.

Documents reviewed included procedures, engineering calculations, modifications, work orders, site drawings, corrective action documents, applicable sections of the Updated Final Safety Evaluation Report, supporting analyses, Technical Specifications, and design basis documentation. The modification packages reviewed are in the List of Documents Reviewed of the Attachment to this report.

b. 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 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.

WO Number	Maintenance/Test Activity
WO 98376321-04	1B KC heat exchanger tube cleaning
WO 98468427	‘A’ fire suppression pump relay addition
WO 98466519-01	1A EDG testing following complex maintenance
WO 98458619-01	Unit 1 CA pump turbine steam supply piping electrical heat trace system breaker replacement
WO 98473113	2A EDG testing following turbocharger replacement

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the surveillance test procedures listed below to verify that Technical Specification surveillance requirements and/or Selected Licensee Commitment requirements were properly incorporated and that test acceptance criteria were properly specified. The inspectors observed actual performance tests and/ or reviewed completed procedures to verify that acceptance criteria had been met. The inspectors also verified that proper test conditions were established and that no equipment preconditioning activities were occurring.

Procedure Number	Title
IP/0/A/3710/011	125 VDC Vital I&C Power System Battery Charger Capacity Test
IP/1/A/3222/076D	Calibration Procedure for Delta T/T-AVG. Protection Channel IV
PT/0/A/4150/012B	Moderator Temperature Coefficient of Reactivity Measurement [End of Life]
PT/2/A/4250/003C	Turbine Driven Auxiliary Feedwater Pump Operability [Unit 2]
PT/2/A4350/002A	Diesel Generator 2A Operability Test [24-hr run]
PT/2/A/4350/002B	Diesel Generator 2B Operability Test [one-hr run]
PT/2/A/4600/002A	Mode 1 Periodic Surveillance Items
TT/0/A/9300/037	Controlling Procedure for Differential Pressure Testing of Motor Operated Valves 1(2)RN028A and 1(2)RN038B

As part of a periodic review of the licensee's own identification and resolution of problems in this area, the inspectors reviewed PIP C-02-00718, which documented the failure of the 2A EDG during a one-hour surveillance test on February 12, 2002. The failure was attributed to a failed turbocharger, which was subsequently replaced. The 2A EDG was retested and returned to operable status by February 15, 2002.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed two temporary modifications to verify that the functions of an important safety system were not compromised. The inspectors observed the modifications in the field, and reviewed post modification testing requirements to ascertain whether the modifications achieved the desired results. These modifications had been developed to ensure the operability of the CA system, after degraded conditions were identified by the licensee.

Temporary Modification	Title or Description
CNTM-0085	Auxiliary steam supply to Unit 2 turbine-driven CA pump drain line installation
CNTM-0089	Temporary jumper across sliding link E-3 in Unit 2 turbine-driven CA pump control panel

As part of a continuing review of the licensee's own identification and resolution of problems in this area, the inspectors reviewed PIP C-00-03682, which identified concerns related to how the temporary modification process was being implemented.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

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	Unplanned Scrams per 7,000 Critical Hours
Initiating Events	Unplanned Power Changes per 7,000 Critical Hours
Initiating Events	Scrams with a Loss of Normal Heat Removal

This review was conducted for fourth quarter 2001 PI data submitted to the NRC on or about January 21, 2002. To verify the PI data, the inspectors reviewed control room logs, Operator Aid Computer trends, Licensee Event Reports, and related licensee calculations. The inspectors verified samples of data for the entire period covered by the PI under review (e.g., for PIs covering four quarters, the inspectors reviewed samples of data for the three quarters immediately prior to fourth quarter 2001 in addition to that quarter's data).

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 50-414/01-002-00: Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks Found in Steam Generator Channel Head Bowl Drain Line on 2B Steam Generator.

On September 19, 2001, the licensee discovered boric acid residue located on the 2B SG bowl drain piping during the Unit 2 refueling outage. The licensee determined that NC system pressure boundary leakage existed at the bowl drain nozzle coupling-to-outer channel head weld. Several small flaw indications were identified through dye-penetrant testing and were repaired and properly tested prior to plant startup from the refueling outage. The flaws were attributed to primary water stress corrosion cracking (PWSCC) of Alloy 600 material, which was used in the construction of the bowl drain piping.

The inspectors concluded that the pressure boundary leakage had a credible impact on safety in that it represented degradation of the primary system pressure boundary. However, the finding was of very low safety significance because the pressure boundary leakage was considered to be minimal, as the volume of boron residue was reportedly only one cubic inch, and the leakage was not detectable during routine NC system leakage calculations conducted while the plant was operating.

The licensee determined that the leakage likely occurred while the plant was operating in Modes 1 through 4 during the previous operating cycle. Technical Specification 3.4.13.a allows no pressure boundary leakage in those operating modes and requires that the plant be in Mode 3 within six hours and in Mode 5 in the next 36 hours if such leakage exists. The licensee had not complied with this requirement since the leakage had gone undetected while the unit was operating. Operating with NC system pressure boundary leakage for longer than the limiting condition for operation allowed constituted a violation of TS 3.4.13.a. This violation is considered to be licensee-identified and is listed in Section 40A7 of this report.

.2 <u>(Closed) LER 50-413/01-003-00</u>: Control Room [Area] Ventilation System Inoperability due to Accuracy of Flow Measurements Resulting in Non-Compliance with Technical Specifications.

This condition was discovered by the licensee on October 11, 2001, during review of past performance tests for the control room area ventilation system (CRAVS). The licensee identified that CRAVS Train A air flow did not meet the 6,000 (plus or minus 10 percent) cubic feet per minute operability requirements of TS 3.7.10 and TS 5.5.11 a.,b., and d. on at least two occasions due to inaccurate flow measurements. The licensee also determined that CRAVS Train B was removed from service during the two periods that CRAVS Train A was inoperable. This resulted in both CRAVS trains being simultaneous inoperable, which was not permitted by TS 3.7.10. The inspectors reviewed the LER and previous test data and determined that these conditions represented two violations of TS 3.7.10. However, the CRAVS safety function to automatically initiate and provide adequate pressurization of the control room was not affected. The error introduced by the inaccurate air flow measurements did not result in flow rates which would have caused damage to the ventilation fans or prevented them from performing their functions, due to the additional margin provided by the TS flow requirements. Therefore, these two violations of TS 3.7.10 constitute violations of minor significance that are not subject to enforcement action, in accordance with Section IV of the Enforcement Policy. This condition was captured in the licensee’s corrective action program as PIP C-01-03993.

.3 <u>(Closed) LER 50-414/01-003-00</u>: Electrical Fault in Reactor Coolant Pump Motor Stator Causes Automatic Reactor Trip, Autostart of Emergency Diesel Generator, and Autostart of Auxiliary Feedwater

On December 7, 2001, the Unit 2 reactor tripped due to low flow in the D NC system loop when the 2D NC pump motor feeder breaker opened in response to protective relay actuation caused by an electrical fault internal to the motor. The inspector reviewed the causal factors and the corrective actions identified by the licensee. No findings were identified.

4OA5 <u>Other</u>

a. <u>Inspection Scope</u>

Using Temporary Instruction (TI) 2515/145, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” the inspectors reviewed the licensee’s activities in response to NRC Bulletin 2001-01 [same title] to verify licensee’s compliance with applicable regulatory requirements. The inspectors performed this TI for Unit 2, which had been shut down for a refueling outage. Note that this inspection was completed during the previous inspection period which ended on December 20, 2001, but is being documented in this inspection report.

b. Findings

No findings of significance were identified.

Because Catawba Unit 2 was categorized in NRC Bulletin 2001-01 as being a Bin 4 plant, or one having a low susceptibility to primary water stress corrosion cracking (PWSCC), the licensee did not perform enhanced ultrasonic or visual examination of its reactor pressure vessel (RPV) head penetrations during the Fall 2001 outage. However, the Agency has since issued NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," requiring all pressurized water reactor (PWR) licensees to submit plans for future inspections of the RPV head penetrations and/or outside surface of the RPV in light of recent discoveries of degradation at another PWR facility. The licensee submitted a response dated April 1, 2002, indicating that it would conduct a complete bare metal visual inspection of the RPV heads for Unit 1 (also a Bin 4 plant) and Unit 2 during their next refueling outages scheduled to begin in April 2002 and March 2003, respectively. The inspectors will review the licensee's upcoming activities for Unit 1 in accordance with the guidance contained in TI 2515/145.

4OA6 Meetings

.1 Exit Meeting

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 March 25, 2002. No proprietary information was identified.

.2 Annual Assessment Meeting

On March 20, 2002, the NRC Division of Reactor Projects Branch Chief and the Senior Resident Inspector assigned to Catawba met with Duke Energy Corporation, to discuss the NRC's Reactor Oversight Process (ROP) and the Catawba annual assessment of safety performance for the period of April 1, 2001 - December 31, 2001. The major topics addressed were: the NRC's assessment program, the results of the Catawba assessment, and the NRC's Agency Action Matrix. Attendees included Catawba site management, members of site staff, members of the public, members of the South Carolina Department of Health and Environmental Control, and news media personnel.

This meeting was open to the public. Information used for the discussions of the ROP is available from the NRC's document system (ADAMS) as accession number ML020600179. ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

4OA7 Licensee Identified Violations.

The following finding of very low safety significance was identified by the licensee and is a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations (NCVs).

If you deny this Non-cited Violation, 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.

NCV Tracking Number	Requirement Licensee Failed to Meet
50-414/01-07-01	Unit 2 Reactor Coolant System Pressure Boundary Leakage while Operating in Modes 1 through 4, resulting in a Violation of TS 3.4.13.a. This issue was captured in the licensee's corrective action program as PIP C-01-04283. This finding was of very low safety significance because the pressure boundary leakage was considered to be minimal, as the volume of boron residue was reportedly only one cubic inch, and the leakage was not detectable during routine NC system leakage calculations conducted while the plant was operating. (Green)

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

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W. Byers, Site Security Manager
J. Foster, Radiation Protection Manager
G. Gilbert, Regulatory Compliance Manager
M. Glover, Station Manager
W. Green, Work Control Superintendent
P. Grobusky, Human Resources Manager
P. Herran, Engineering Manager
R. Parker, Maintenance Superintendent
G. Peterson, Catawba Site Vice President
F. Smith, Chemistry Manager
R. Sweigart, Safety Assurance Manager

NRC

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B. Mallett, Region II
B. McCabe, Office of the Commission
R. Haag, Region II
E. Christnot, Region II

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-414/01-07-01	NCV	Violation of TS 3.4.13.a. due to Unit 2 Reactor Coolant System Pressure Boundary Leakage while Operating in Modes 1 through 4 (Section 4OA7)

Previous Items Closed

50-414/01-002-00	LER	Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks Found in Steam Generator Channel Head Bowl Drain Line on 2B Steam Generator (Section 4OA3.1)
50-413/01-003-00	LER	Control Room [Area] Ventilation System Inoperability due to Accuracy of Flow Measurements Resulting in Non-Compliance with Technical Specifications (Section 4OA3.2)

50-414/01-003-00	LER	Electrical Fault in Reactor Coolant Pump Motor Stator Causes Automatic Reactor Trip, Autostart of Emergency Diesel Generator, and Autostart of Auxiliary Feedwater (Section 4OA3.3)
TI 2515/145 (Unit 2)	TI	Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles - NRC Bulletin 2001-01 (Section 4OA5)

LIST OF ACRONYMS USED

ASG	-	Auxiliary Safeguards Test
CA	-	Auxiliary Feedwater
CFR	-	Code of Federal Regulations
CRAVS	-	Control Room Ventilation System
EDG	-	Emergency Diesel Generator
KC	-	Component Cooling Water
LER	-	Licensee Event Report
NCV	-	Non-Cited Violation
ND	-	Residual Heat Removal
NEI	-	Nuclear Energy Institute
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
PI	-	Performance Indicator Facility
PIP	-	Problem Investigation Process (report)
PORV	-	Power-Operated Relief Valve
PRA	-	Probabilistic Risk Assessment
PWR	-	Pressurized Water Reactor
PWSCC	-	Primary Water Stress Corrosion Cracking
QC	-	Quality Control
Rev	-	Revision
RN	-	Nuclear Service Water
ROP	-	Reactor Oversight Process
RPV	-	Reactor Pressure Vessel
SDP	-	Significance Determination Process
SG	-	Steam Generator
SSC	-	Systems, Structures, and Components
SSF	-	Standby Shutdown Facility
TS	-	Technical Specification
Vdc	-	Volts direct current
WO	-	Work Order
YC	-	Control Room Ventilation System Chilled Water System

LIST OF DOCUMENTS REVIEWED

Documents Reviewed in Section 1R02

Safety Evaluations

CN 11391	Modify CA Flow Control Valves to Prevent S/G Overfill During SGTR
CN 11396	Install New 8" MOV valves1RN250A and 1RN310B and New 8" Nuclear Service Water System to Auxiliary Feedwater System
CN 11389	Backup Cooling for NV System Centrifugal Charging Pump 1A Motor Coolers and Pump Oil Coolers
CN-11400	Additional Temperature Monitoring in the UST and Hotwell, and the Installation of Control Switches in the Control Room to Block CA Pumps Low Suction Pressure Trip
CN 11401	Reroute and Enlarge Upper Storage Tank to Auxiliary Feedwater Condensate Storage Tank Piping
CNCE 61291	Removal of Internals From Check Valves 1CA171 and 1CA172 in the RN to CA Supply Lines
CNCE-61531	Restore 1CA6 to Operable Status
CNCE-61595	Add Switches in Auxiliary Safeguards Cabinets to Bypass P-12 Interlock in Mode 4 for extended Cooldown on Condenser Steam Dumps
CNCE 61633	Nuclear Service Water System Pit A Swap Set Point
CNCE 70610	Replace Piping Upstream of 1RN865, and Add Valve 1RNF61

Screened Out Safety Evaluations

CE-61558	Modify Turbine Generator Manway for Test Equipment
CE-10518	Replace Motors on 1RN028A and 1RN038B with 15 ft-lb motors, Rev. 1
1P/0/A/3840/002	Calibration of Rochester Battery Ground Detector, Rev. 9
CNCE 11011	Restore the Load Recycle Temperature Switch for the Train ‘B’ Control Room Area Chillers
TN/1/A/1400/00/04E	Implementation Procedure for NSM CN-11400, work unit 04, Change C
TN/1/A/1400/00/05E	Implementation Procedure for NSM CN-11400, work unit 05, Change A

TN/1/A/1400/00/06E	Implementation Procedure for NSM CN-11400, work unit 06
CNCE 70110	Revise Pipe Supports Attaching to RCP 2B Motor
CNCE 10604	Update the 3 Foot NSW Pond Elevation Increase
CNCE 70856	Gasket Material for Valves 1(2)RN039 Does Not Match Type Shown on Design and Weld Isos
CNCE 11017	Change of Lubricant for Aux FW Pump Motors
CNCE 71243	Provide New Test Acceptance Criteria Associated With Maximum Fowling Factor, Tube Plugging Limits and RN System Flow Requirements for NS HX 2B

Documents Reviewed in Section 1R17

<u>Modifications</u>

CN 11389	Backup Cooling for NV System Centrifugal Charging Pump 1A Motor Coolers and Pump Oil Coolers
CN 11391	Modify CA Flow Control Valves to Prevent S/G Overfill During SGTR
CN 11396	Install New 8" MOV valves1RN250A and 1RN310B and New 8" Nuclear Service Water System to Auxiliary Feedwater System
CN-11400	Additional Temperature Monitoring in the UST and Hotwell, and the Installation of Control Switches in the Control Room to Block CA Pumps Low Suction Pressure Trip
CN 11401	Reroute and Enlarge Upper Storage Tank to Auxiliary Feedwater Condensate Storage Tank Piping
CNCE 10604	Update the 3 Foot NSW Pond Elevation Increase
CNCE 11017	Change of Lubricant for Aux FW Pump Motors
CNCE 61633	Nuclear Service Water System Pit A Swap Set Point
CNCE 70110	Revise Pipe Supports Attaching to RCP 2B Motor
CNCE-70261	Replacement of Ground Detector in 1EDC-F01B
CNCE 70610	Replace Piping Upstream of 1RN865, and Add Valve 1RNF61
CNCE 71243	Provide New Test Acceptance Criteria Associated With Maximum Fowling Factor, Tube Plugging Limits and RN System Flow Requirements for NS HX 2B