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

January 30, 2007

Duke Power Company LLC
d/b/a Duke Energy Carolinas, LLC
ATTN: Mr. J. R. Morris
Site Vice President
Catawba Site
4800 Concord Road
York, SC 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000413/2006005 AND 05000414/2006005

Dear Mr. Morris:

On December 31, 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 January 11, 2007, with Mr. Bill Pitesa 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 four NRC-identified findings of very low safety significance (Green) of which all were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is documented in this report. However, because of their 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/

James H. Moorman, III, 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 05000413/2006005 and 05000414/2006005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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/

James H. Moorman, III, 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 05000413/2006005 and 05000414/2006005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE

ADAMS: X Yes ACCESSION NUMBER:___ML070300874_

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRS	RII:DRS	RII:DRS
SIGNATURE	CWR /RA/	JHM /RA/	ATS via phone		JDF /RA/	JER via email	RMB via email
NAME	CRapp	JMoorman	ASabisch	GWilliams	JFuller	JRivera-Ortiz	RBerryman
DATE	01/25/2007	01/30/2007	01/30/2007		01/30/2007	01/25/2007	01/25/2007
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRS			
SIGNATURE	JHM for	RKH via email	JL /RA/	JHM for			
NAME	JDiaz-Velez	RHamilton	JLenahan	EMichel			
DATE	01/26/2007	01/25/2007	01/25/2007	01/29/2007			
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: C:\FileNet\ML070300874.wpd

cc w/encls:
Randy D. Hart
Regulatory Compliance Manager
Duke Energy Corporation
Electronic Mail Distribution

Lisa Vaughn
Associate General Counsel
Duke Energy Corporation
526 South Church Street
Mail Code EC 07H
Charlotte, NC 28202

Timika Shafeek-Horton
Assistant General Counsel
Duke Energy Corporation
526 South Church Street-EC07H
Charlotte, NC 28202

David A. Repka
Winston & Strawn LLP
Electronic Mail Distribution

North Carolina MPA-1
Electronic Mail Distribution

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

R. Mike Gandy
Division of Radioactive Waste Mgmt.
S. C. Department of Health and
Environmental Control
Electronic Mail Distribution

Elizabeth McMahon
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

County Manager of York County, SC
Electronic Mail Distribution

Piedmont Municipal Power Agency
Electronic Mail Distribution

R. L. Gill, Jr., Manager
Nuclear Regulatory Issues
and Industry Affairs
Duke Energy Corporation
526 S. Church Street
Charlotte, NC 28201-0006

Distribution w/encl: (See page 4)

Letter to J. R. Morris from James Moorman, III dated January 30, 2006

SUBJECT: CATAWBA NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000413/2006005 AND 05000414/2006005

Distribution w/encl:
J. Stang, NRR
C. Evans
L. Slack, RII EICS
OE Mail
RIDSNRRDIRS
PUBLIC

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-413, 50-414
License Nos.:	NPF-35, NPF-52
Report No.:	05000413/2006005 and 05000414/2006005
Licensee:	Duke Power Company, LLC
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	York, SC 29745
Dates:	October 1, 2006 through December 31, 2006
Inspectors:	R. Berryman, Senior Reactor Inspector (Section 1R06) J. Diaz-Velez, Health Physicist (Section 2OS2) J. Fuller, Acting Resident Inspector R. Hamilton, Health Physicist (Sections 2OS1, 2PS2, and 4OA1) J. Lenahan, Senior Reactor Inspector (Section 4OA5.3) E. Michel, Reactor Inspector (Section 1R08) J. Rivera-Ortiz, Acting Resident Inspector (Section 4OA5.2) A. Sabisch, Senior Resident Inspector G. Williams, Project Engineer
Approved by:	James Moorman, III, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000413/2006-005, 05000414/2006-005; 7/1/2006 - 9/30/2006; Catawba Nuclear Station, Units 1 and 2; Emergent Work Risk Management, Problem Identification and Resolution, Inservice Inspection Activities, and Design/Test Control.

The report covered a three-month period of inspection by three resident inspectors, a project engineer, three operations engineers, three reactor inspectors, and two health physicists. Four NRC-identified 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 Inspection Manual Chapter (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: Barrier Integrity

- Green. The inspectors identified a green NCV of 10 CFR 50.65(a)(4) for the licensee failing to adequately recognize, assess, and manage the increased risk resulting from the failure of the single operable spent fuel pool cooling pump with the opposite train's emergency diesel generator inoperable and the recently unloaded Unit 1 reactor core in the spent fuel pool.

  The finding was more than minor because the deficiency is consistent with IMC 0612, Appendix B, Section 3, Minor Screening Question (5)(i). Specifically, the licensee failed to expeditiously develop and implement risk management actions to address the elevated risk the unit was in based on the 1B KF pump failure and other equipment out of service or in an outage alignment; i.e., core in the spent fuel pool and the 1A DG disassembled. The finding was associated with the Systems, Structures and Components (SSC) Performance attribute of the Barrier Integrity cornerstone and affected the cornerstone objective of maintaining the functionality of the spent fuel pool cooling system. The inspectors completed a Phase 1 screening of the finding using Appendix K of Inspection Manual Chapter 0609, "Maintenance Risk Assessment and Risk Management Significance Determination Process," and determined that the performance deficiency represented a finding of very low risk significance (Green), based on the resulting magnitude of the calculated Incremental Core Damage Probability being below 1E-6. This was derived from discussions with the Region II Senior Reactor Analysts based on the time to boil in the Spent Fuel Pool being >24 hours which allows for operator actions to mitigate the effect of a postulated loss of cooling scenario. This finding has been entered into the licensee's Corrective Action Program as Problem Investigation Process reports (PIP) C-06-7829 and C-06-7840. The pump was returned to operable status approximately 48 hours after the failure occurred. This finding directly involved the cross-cutting aspect of Human Performance under the "Safety Significant / Risk Significant Decisions" aspect of the "Decision Making" component, in that the licensee failed to adequately recognize, assess and manage the increased risk resulting from the failure of the 1B Spent Fuel Pool Cooling (KF) pump during outage conditions on Unit 1 (Section 1R13).

- Green. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion III, Design Control, and Criterion XI, Test Control, for the licensee's failure to have design documentation to support the ice condenser lower inlet door surveillance procedure test acceptance limits. The licensee subsequently received the supporting information from the vendor and incorporated it into the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS) and surveillance procedures.

  The inspectors determined that the licensee's failure to have design documentation that supported the acceptance criteria contained in the TS surveillance procedures used to test the ice condenser's lower inlet doors at the 40-degree open position was a performance deficiency. The requirement to maintain design bases documentation for tests performed on safety-related SSC's is contained in 10 CFR 50, Appendix B, Criterion III. The requirement to implement a test program that incorporates the design basis for these components is contained in 10 CFR 50, Appendix B, Criterion XI. It was determined to be more than minor using the guidance contained in IMC 0612, Appendix B, Issue Screening, in that an excessively high closing torque could adversely impact the ability of the lower inlet door to modulate properly in the event of a small-break Loss of Coolant Accident (LOCA); however, with no lower limit defined in the surveillance test's acceptance criteria, this condition might not have been identified and corrected prior to returning the unit to power operation. The finding is associated with the Barrier Integrity cornerstone and affected the integrity of the reactor containment structure; i.e., the ice condenser's ability to control internal pressure following a LOCA event, and protect the public from radio-nuclide releases. The licensee contracted the vendor to reconstruct the design basis of the 40-degree torque test and has incorporated this analysis into the applicable surveillance procedure, Technical Specification and Design Basis Documents. This finding directly involved the cross-cutting area of Human Performance under the "Complete Documentation and Component Labeling" aspect of the "Resources" component, in that the licensee failed to maintain complete, accurate and up-to-date design documentation and procedures. (Section 1R22)

  Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding involving an NCV of 10 CFR Part 50.55a(g)(4)ii for failure to perform a volumetric examination of the 1A Residual Heat Removal (ND) heat exchanger as required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. The examinations were performed from the nozzle side of the weld only and the required examination coverage was not obtained as required by Section XI of the ASME Code. The limited ultrasonic (UT) examinations found no indications that the structural integrity of the supports was unacceptable for service. The licensee entered this issue into the Corrective Action Program as PIP C-06-5142 and has completed a 100 percent UT examination of the 1A ND heat exchanger inlet and outlet nozzles during 1EOC16 with no detected indications.

This finding was of more than minor significance because a failure to examine the 1A ND heat exchangers as required by the ASME Code is related to the “Equipment Performance” attribute of the “Mitigating Systems” cornerstone and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was evaluated using Phase 1 of the NRC IMC 0609, “Significance Determination Process,” and was determined to be of very low safety significance. This finding directly involved the cross-cutting area of Human Performance under the “Proper Work Planning” aspect of the “Work Control” component, in that the licensee did not properly plan and coordinate a work activity consistent with nuclear safety. Inadequate planning for 1A RHR HX inlet and outlet nozzle UT examinations resulted in the availability of only one (of two) required calibration blocks. (Section 1R08)

- Green. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for the licensee's failure to identify and implement effective corrective actions to prevent recurring deficiencies associated with the erection of scaffolding around safety related equipment. For the examples identified by the inspectors, the licensee removed or adjusted the scaffolding to correct the condition.

  The inspectors determined that the licensee's repeated failure to erect scaffolding in accordance with the Duke Scaffold Manual and implement effective corrective actions to prevent recurrence was a performance deficiency. The inspectors determined that the performance deficiency was more than minor in that multiple occurrences were identified of scaffolding being located in a manner where safety-related equipment could be adversely impacted without the appropriate engineering evaluation or approval. In accordance with Appendix B, "Issue Screening," of IMC 0612, the inspectors determined that the finding was of more than minor significance since the finding was associated with the equipment performance and human performance attributes of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of equipment that respond to initiating events to prevent undesirable consequences. This finding directly involved the cross-cutting area of Problem Identification and Resolution under the "Appropriate and Timely Corrective Actions" aspect of the "Corrective Action Program" component, in that ineffective corrective actions were established resulting in additional scaffolding deficiencies being identified over an 18 month period. The licensee has entered this issue into the corrective action program as PIP C-06-8183 and has identified scaffold construction and usage as an adverse trend requiring additional focus in 2007. (Section 4OA2.2)

B. Licensee-Identified Violations.

One violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period operating at 100 percent Rated Thermal Power. On November 9, 2006, power was reduced to 95 percent to support main steam safety valve testing. The unit was removed from service on November 11, 2006 for the 1EOC16 refueling outage. The unit's reactor went critical on December 30 and the generator was placed on-line on December 31. Power ascension continued through the end of this inspection period.

Unit 2 began the inspection period operating at 100 percent Rated Thermal Power. On December 8, 2006, power was reduced to 50 percent to allow for the 2A main feedwater pump to be removed from service to repair the main thrust bearing that had exhibited elevated temperatures. The unit returned to 100 percent power on December 11, 2006 and remained at 100 percent Rated Thermal Power 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 reviewed the licensee's preparations for adverse weather associated with cold ambient temperatures for the following three risk significant systems. This included field walkdowns to assess the material condition and operation of freeze protection equipment (e.g., heat tracing, instrument box heaters, area space heaters, etc.), as well as other preparations made to protect plant equipment from freeze conditions. In addition, the inspectors conducted discussions with operations, engineering, and maintenance personnel responsible for implementing the licensee's cold weather protection program to assess the licensee's ability to identify and resolve deficient conditions associated with cold weather protection equipment prior to cold weather events. Documents reviewed during this inspection are listed in the Attachment to this report.

- Standby Shutdown Facility
- Nuclear Service Water Pump House
- Refueling Water Storage Tanks

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed walkdowns of the following seven system alignments to verify that critical portions of equipment alignments remained operable while the redundant components or 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 to this report.

- Protection of the 1A Diesel Generator (DG), 1A 4.16 kV Vital Bus (1ETA) switchgear, Standby Shutdown Facility (SSF) and switchyard with the 1B DG out of service for planned maintenance
- Protection of the 2A Auxiliary Feedwater (CA) pump and the 2ETA breaker associated with the 2A CA pump while preventive maintenance was being performed on the 2B CA pump and related valves
- Protection of the 1B DG, 1B 4.16kV vital bus (1ETB) switchgear, SSF and switchyard with the 1A DG out of service for planned maintenance
- Protection of the 1ETB, 1B KF pump and support equipment, and the B train of Nuclear Service Water (RN) with the 1A 4.16kV vital bus and associated equipment removed from service for maintenance during the Unit 1 refueling outage (full core offload in the spent fuel pool)
- Protection of the equipment identified in the Outage Schedule Change package required to support the start of “B” train RN cross-over piping work following the failure of the #4 journal bearing on the 1A DG
- Protection of Plant Equipment Identified in the Risk Management Action Plan implemented to address both Unit 1 DGs being declared inoperable due to bearing issues
- Protection of plant equipment when entering mid-loop conditions on Unit 1 to support vacuum refill of the reactor coolant system

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors walked down accessible portions of the following nine 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 to this report.

- Unit 1 Essential Battery Rooms; Auxiliary Building 554 foot elevation
- Unit 1 "A" DG Room
- Unit 1 CA pump room, Auxiliary Building 543 foot elevation
- Unit 1 "A" train essential 4.16kV switchgear room; Auxiliary Building 577 foot elevation
- Unit 1 Electrical Penetration Room; Auxiliary Building 574 foot elevation; Rooms 491 and 491A
- Unit 2 "B" train essential 4.16kV switchgear room; Auxiliary Building 560 foot elevation
- Unit 2 Spent Fuel Pool; Auxiliary Building 605 foot elevation
- Unit 2 Spent Fuel Pool Purge Unit; Auxiliary Building 636 elevation
- Nuclear Service Water Pump Structure

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, Individual Plant Examination, and flood analysis documentation associated with internal plant areas to determine the effect of flooding. The inspectors reviewed the licensee's internal flood protection features for the following two areas. The internal areas were selected and walked down based on the flood analysis calculations. Through observation and design review the inspectors reviewed sealing of doors, holes in elevation penetrations, sump pump operations, and potential flooding sources. The inspectors reviewed the corrective action program documents to verify that the licensee was identifying issues and resolving them. Documents reviewed during this inspection are listed in the Attachment to this report.

- Auxiliary building; 522 foot elevation and the CA pump pits and rooms
- Standby Shutdown Facility

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Piping Systems and Containment ISI

a. Inspection Scope

Between November 27 and December 1, 2006, the inspectors reviewed the implementation of the licensee's ISI program used for monitoring the reactor coolant system and risk significant piping system boundaries for degradation. The inspectors selected a sample of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI required examinations for review.

The inspectors conducted an on-site review of nondestructive examination (NDE) activities to evaluate compliance with Technical Specifications and the applicable editions of ASME Section V and XI (1989 Edition; No Addenda, for examinations credited to the second 10-year ISI interval, and 1998 Edition; 2000 Addenda, for examinations credited to the third 10-year ISI interval), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of ASME Section XI, IWB-3000 or IWC-3000 acceptance standards.

The inspectors reviewed final reports for the NDE inspections of the welds described below to verify that the evaluation and disposition of indications was in accordance with the applicable version of ASME Section XI, IWB-3000.

- Phased array UT examination of Pressurizer surge line weld overlay at 1-PZR-W1SE (ASME Class 1).
- Liquid Penetrant examination of Pressurizer surge line weld overlay at 1-PZR-W1SE (ASME Class1).

The inspectors directly observed a sample of welding activities performed during the refueling outage for ASME Class 1 piping. The inspectors reviewed welding procedures, procedure qualification records, and welder qualification records, for the following welds:

- Weld overlay at 1PZR-W4ASE, Pressurizer safety line nozzle, 6-inch diameter, (ASME Class 1).

The inspectors reviewed a sample of examinations with recordable indications that were accepted by the licensee for continued service. The inspectors verified indications were within the acceptance criteria established by ASME B&PV Code, Section XI, IWB-3000.

- UT of 1-RPV-W05, Reactor Vessel Upper Shell to Middle Shell Circ. Weld, (ASME Class 1).
- UT of 1-RPV-W17, Reactor Vessel Outlet Nozzle to Shell Weld at the 202-degree location, (ASME Class 1).

b. Findings

Introduction: The inspectors identified a finding involving an NCV of 10 CFR Part 50.55a(g)(4)(ii) having very low safety significance (Green) for the licensee's failure to perform a UT of the 1A ND heat exchanger inlet and outlet nozzles in accordance with the requirements of Section XI of the ASME B&PV Code for the second 10-year ISI interval.

Description: The inspectors reviewed Catawba Nuclear Station Unit 1 Relief Request Number 05-CN-001 which had been submitted to the NRC on February 17, 2005 for the licensee's inability to obtain the examination coverage requirements (90% per Code Case N-460) of the 1A ND heat exchanger inlet and outlet nozzle-to-shell welds.

The licensee requested relief from the examination coverage required by the ISI Code of Record for the second 10-year ISI interval (Section XI of the ASME Code, 1989 Edition). The licensee performed UT examinations of 1A ND heat exchanger inlet and outlet nozzle welds on October 28, 2003. Because of differences in the thickness between the nozzles and shell materials, two different calibration blocks were required and only the calibration block associated with the nozzles was available. This permitted examination only from the nozzle side of the weld and resulted in 14.25 percent coverage vice the required 90 percent. Consequently, these examinations could not be credited to the second 10-year ISI interval and the relief request was denied. The limited UT examinations found no indications that the structural integrity of the welds were unacceptable for service. A liquid penetrant test was also performed on the subject welds with 100 percent coverage, which found no significant material degradation that could represent a structural integrity concern. The licensee also documented a determination of operability in PIP C-06-05142.

Because the licensee did not perform a complete UT examination of the 1A ND heat exchanger inlet and outlet nozzle-to-shell welds required by the ASME Code, the inspectors determined that the applicable requirements of the ASME Code, Section XI, Subsection IWC, Item C2.21 were not met for the second 10-year ISI interval for Unit 1.

Analysis: The inspectors determined that the failure of the licensee to perform a complete UT examination of the 1A ND heat exchanger inlet and outlet nozzle welds, as required by the ASME Code, was a performance deficiency. This finding was of more than minor significance because the ND heat exchangers are a part of the decay heat removal flow path during LOCA (i.e. ECCS recirculation) and non-LOCA conditions. A failure of the ND heat exchanger inlet or outlet welds due to material degradation could result in a challenge of the ND system boundary and the ND system's ability to remove decay heat from the reactor core. Therefore, a failure to examine the ND heat exchangers as required by the ASME Code is related to the "Equipment Performance" attribute of the "Mitigating Systems" cornerstone and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was evaluated using Phase 1 of the NRC IMC 0609, "Significance Determination Process SDP," and determined to be of very low safety significance (green) because it was a qualification deficiency that did not result in a loss of operability. This finding directly involved the cross-cutting area of

Human Performance under the “Proper Work Planning” aspect of the “Work Control” component, in that the licensee did not properly plan and coordinate a work activity consistent with nuclear safety. Inadequate planning for 1A ND HX inlet and outlet nozzle UT examinations resulted in the availability of only one (of two) required calibration blocks.

Enforcement: 10 CFR 50.55a(g)(4)ii requires that inservice examination of components and system pressure tests must comply with the requirements of the ASME Code (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147). ASME Code, Section XI, Subsection IWC, Article IWC-2500, Table IWC-2500-1, Item C2.21 requires a volumetric examination (UT was performed in this case) of all nozzles at terminal ends of piping runs each inspection interval. Contrary to this requirement, on 10/28/2003, the licensee did not perform a complete UT examination of the 1A ND heat exchanger inlet and outlet nozzles. These examinations were limited in their coverage and could not be credited to the examination requirements for the third 10-year ISI interval. Because of the very low safety significance of this finding, and the issue was entered into the licensee’s corrective action program (PIP C-06-05142), and the licensee completed a 100 percent UT examination of the 1A ND heat exchanger inlet and outlet nozzles during 1EOC16 with no detected indications, it is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy: NCV 05000413/2006005-01, Failure to Perform Adequate Examinations of 1A ND Heat Exchanger Inlet and Outlet Welds.

.2 Boric Acid Corrosion Control (BACC) Program

a. Inspection Scope

Between November 27 and December 1, 2006, the inspectors reviewed the licensee’s BACC activities to ensure verify the program was being implemented in accordance with commitments made in response to NRC Generic Letter 88-05 “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary” and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and the results of the licensee’s Mode 3 containment walkdown inspection from this outage. The inspectors also conducted an independent walk-down of the reactor building to evaluate compliance with licensee BACC program requirements and verify that degraded or non-conforming conditions, such as boric acid leaks identified during the Mode 3 containment walkdown, were properly identified and corrected in accordance with the licensee’s corrective action program.

The inspectors reviewed a sample of engineering evaluations completed for evidence of boric acid found on systems containing borated water to verify that the minimum design code required section thickness had been maintained for the affected components. Specifically, the inspectors reviewed the following evaluations:

- PIP C-05-07338, 1FW-001A active boron leak at packing gland (Refueling Water System).
- PIP C-06-03049, Dried boron on the mechanical seal flanges of the 2A and 2C reactor coolant pumps (Reactor Coolant System).
- PIP C-05-06604, 2NV-232; Active boron required engineering evaluation (Chemical and Volume Control System).

The inspectors performed a review of ISI related problems, including welding, BACC and SG ISI, that were identified by the licensee and entered into the corrective action program as Problem Investigation Process (PIP) documents. The inspectors reviewed the PIPs to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Resident Quarterly Observation.

a. Inspection Scope

The inspectors observed Simulator Exercise Guide OP-CN-LOR-S-54 to assess the performance of licensed operators. The exercise included three scenarios where a loss of shutdown cooling occurred in various points of a typical refueling outage. The inspection focused on high-risk operator actions performed during implementation of the normal and abnormal operating procedures, and the incorporation of lessons-learned from previous plant and industry events. Through observations of the critique conducted by training instructors throughout the simulator session, the inspectors assessed whether appropriate feedback was provided to the licensed operators regarding identified weaknesses.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee’s effectiveness in performing the following five 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 to this report.

- Preventative maintenance activities and inspections on the 2A Residual Heat Removal (ND) pump and selected valves in the 2A ND flowpath
- Post maintenance testing associated with the pre-outage work on the 1A DG
- Problem investigation and base metal weld repairs of American Society of Mechanical Engineers Code Class 2, Section III piping near valve 1 NV-11A
- Repair work on the 1A DG following failure of the #4 bearing during break-in run
- Repair and testing associated with the failure of Unit 2 Containment Air Return System Air Return Dampers 8, 9 and 10 to open during surveillance testing

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 12 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 (NSD) 415, Operational Risk Management (Modes 1-3), and NSD 403, Shutdown Risk Management (Modes 4,5,6, and No Mode), for appropriate guidance to comply with 10 CFR 50.65 (a)(4).

- Construction of Independent Spent Fuel Storage Installation haul road turning pads and potential impact on required ground coverage over the RN supply and return headers
- Excavation of the "B" RN supply headers for external inspections
- Assessment and subsequent rescheduling of the excavation to inspect the 10 inch Unit 2 DG RN supply and return headers
- Deferment of the 1A RN-to-CA flush evolution based on the need to open the "A" Emergency Core Cooling System (ECCS) containment sump for measurement and inspection and the resulting white Outage Risk Assessment Model (ORAM) risk profile with both activities conducted at the same time
- Assessment of planned and in-progress work activities following the identification of a potential crack in DG 1A cylinder head 7L
- Replacement of failed Unit 2 Electro-Hydraulic communication cards using a Complex Evolution Plan
- Risk Management Actions to address the increased risk resulting from the loss of the 2B KF pump
- Protection of equipment identified in the Outage Schedule Change package required to allow cross-train work to take place on RN piping associated with the cross-tie modification

- Protection of plant equipment through the development and implementation of a Risk Management Action Plan following the identification of a potential common mode failure mechanism associated with the bearings in the Unit 1 A and B diesel generators
- Review and rescheduling of maintenance activities to support allowing Engineered Safeguards Features testing to start immediately following the completion of 1A DG repair activities and prior to the start of planned maintenance on the 1B DG
- Performance of a surveillance run of the 1B DG when the 1A DG was inoperable and the B train equipment was being protected
- Scheduling of Unit 2 downpower to 50 percent power to support repair of the 2A main feedwater pump

b. Findings

Introduction: A Green NCV of 10 CFR 50.65(a)(4) was identified by the inspectors for the licensee's inadequate recognition, assessment and management of the increased risk resulting from the failure of the single operable spent fuel pool cooling pump with the opposite train's emergency diesel generator (DG) inoperable and the recently unloaded Unit 1 reactor core in the spent fuel pool.

Description: On November 17, 2006, Unit 1 began to offload the reactor core into the spent fuel pool as part of their scheduled activities associated with refueling outage 1EOC16. The core was fully offloaded by 0400 on November 19, 2006. Spent fuel pool cooling was provided by the 1B KF pump. The 1A KF pump was not available per the definition contained in NSD 403; Shutdown Risk Management (Modes 4, 5, 6 and No-Mode) per 10 CFR 50.65(a)(4), because the associated emergency power source; i.e., the 1A DG, was removed from service for routine maintenance activities. This condition was allowed per Technical Specifications and placed the unit in a Yellow Defense in Depth (DID) risk profile.

At approximately 1100 on November 21, 2006, the 1B KF pump exhibited rapidly increasing temperatures on the inboard pump bearing and the operators manually secured the pump. The 1A KF pump was started to provide cooling to the spent fuel pool. While the 1A KF pump could be placed in service due to electrical power being cross-tied from Unit 2 in support of the outage, it did not have an emergency power supply that would ensure that spent fuel pooling cooling could be provided in the event of a loss of off-site power or a trip of Unit 2. Therefore the 1A KF pump could not be considered available. According to the DID worksheets contained in NSD 403, this configuration; i.e., no trains of KF available, placed Unit 1 in a Red DID condition. Operations personnel incorrectly applied the guidance contained in NSD 403 and credited the 1A KF pump as being available which resulted in the risk condition being disseminated to attendees at the 1130 Outage Meeting as being Orange. While the importance of returning the 1B KF pump to service was emphasized, the need to assess other ongoing outage-related work was not discussed or placed on-hold to ensure the overall risk profile did not degrade further. In addition, a Risk Management Action Plan was not developed to address the emergent equipment failure.

By mid-afternoon on November 21, the licensee determined that the 1A KF pump could not be considered "available" based on the 1A DG having been removed from service for maintenance and re-evaluated the DID worksheet. This review identified that Unit 1 was in a Red DID condition. Confirmation of this decision was obtained through discussions with the PRA group in the General Office. NSD 403 defines a Red DID condition during shutdown as one in which "A key safety function is severely threatened. Immediate restoration is required."

NSD 417; Nuclear Facilities / Generation Status Communications, requires that immediate notifications be made of an entry into any red zone on the DID worksheet to ensure the appropriate station and Nuclear Business Unit management personnel are aware of the condition. This provides the requisite oversight to ensure the appropriate corrective actions and risk management actions are developed and implemented in an expeditious manner. The Red DID condition was discussed within the Outage Control Center (OCC); however, the notifications required by NSD 417, were not made which adversely affected the ability of the senior management team to be aware of and address the emergent situation. The OCC staff and Operations personnel opted to use the regularly scheduled outage update meetings to communicate the status of the 1B KF pump repair efforts to the management team.

NSD 403 requires that Risk Management Actions be developed and implemented to ensure the overall station risk is controlled and the cause of the unplanned entry into a Red DID condition is rectified as expeditiously as possible. A draft Risk Management Action Plan was provided to attendees at the 1600 Outage Meeting; however, it contained limited detail in terms of actions required for implementation. Attendees raised a number of concerns that the draft plan did not address including the following:

- Several required SSC's were not identified as requiring protection including the Spent Fuel Pool ventilation system, some "A" train equipment supporting the remaining running KF pump and electrical power supplies.

- The "time-to-boil" in the spent fuel pool in the event a loss of cooling occurred was not included in the Risk Management Action plan or known by those in attendance.

- A full understanding of work currently in progress or planned for the next 12 hours was not readily available nor had an assessment been performed to determine the potential impact this work could have on the overall plant risk based on the resulting changes to equipment status caused by the pump failure.

The Risk Management Action Plan was to be revised to include more detail pertaining to questions raised during the 1600 meeting and implemented at the start of the night shift. At approximately 1900, discussions were held between the Resident Inspectors and the OCC staff to determine the status of the 1B KF pump Risk Management Action Plan. Maintenance and Engineering personnel had performed an inspection of the failed pump and determined that the probable cause of the elevated bearing temperatures was the bearing seal retaining plate having come loose which came in contact with the bearing assembly. OCC personnel stated that a "success path" had been identified in the repair of the 1B KF pump and its return to service was imminent. As a result of this anticipated restoration, development of the Risk Management Action Plan had been placed on hold. This is contrary to the expectations contained in NSD 403 which requires that corrective actions and a Risk Management Action Plan be developed and implemented in parallel to effectively manage the increased risk when in a Red DID condition.

At approximately 2200, discussions were again held between the Resident Inspectors and the OCC for an update on the status of the 1B KF pump. OCC personnel reported that the pump repairs had not been successful and that additional repair plans were being formulated. The Risk Management Action plan was still in the development stage despite having been in a Red DID condition for approximately 12 hours.

At the 0730 Site Direction Meeting on November 22, 2006, an update on the 1B KF pump was provided. Additional risk management actions had been developed; however, aspects such as the projected time-to-boil for the current time since shutdown and an assessment of on-going work in the plant were still missing from the Risk Management Action Plan. Attendees at the meeting raised a number of questions which required follow-up and a 0930 meeting was scheduled to address the concerns that had been voiced. A structured round-the-clock meeting schedule was established at this time to ensure the Red DID condition was resolved expeditiously.

During the 0930 meeting, a detailed Risk Management Action Plan was provided to the station and clear roles and responsibilities defined going forward. The senior management team was responsible for this focused direction and remained engaged throughout the repairs and restoration of the 1B KF pump. During the period between the failure of the 1B KF pump and the implementation of the final Risk Management Action Plan presented at the 0930 meeting, planned work throughout the plant was allowed to continue without any additional barriers having been put in place to protect equipment or ensure workers were aware of the current risk condition of the unit. This included work that was in-progress on the Nuclear Service Water (RN) supply headers in the Auxiliary Building and fuel cleaning / inspection activities being conducted by contract personnel in the Unit 1 Spent Fuel Pool.

Specific actions added to the Risk Management Action Plan and implemented following the 0930 meeting - approximately 22 hours after the pump failure - to control the increased risk during this period included the following:

Actions to provide increased risk awareness and control:

- Developed and distributed a communication package to all site personnel describing the current plant condition and plans to resolve the issue
- Posted signage at the entrance to the protected area informing personnel that the plant was in a Red DID condition
- Secured the spent fuel pool operating deck access door to minimize the potential to adversely affect the KF system in operation
- Provided increased oversight of work in-progress in the auxiliary building on the RN system in close proximity to the KF system

Actions to reduce the time that the Unit 1 Spent Fuel Pool cooling was degraded:

- Verified that additional parts were available in the event a subsequent failure of a KF pump occurred
- Developed a test and monitoring plan to ensure the 1B KF pump functioned properly following repairs before declaring it operable

Actions to minimize the magnitude of the risk increase

- The OCC staff evaluated the scheduled work to identify any potential impact on the operating 1A KF pump
- The 1B train of spent fuel pool ventilation was added to the protected equipment list
- On-going work activities in the spent fuel pool were evaluated for potential impact on the operating train of KF
- Operations reviewed AP/1/A/5500/026; Loss of Refueling Cavity or Spent Fuel Pool Level, for actions that would be required in the event a loss of spent fuel pool cooling occurred
- Operations visually inspected and verified each cut of the RN piping being performed in the auxiliary building during the period the 1B KF pump was inoperable to ensure protected systems were not impacted
- A table containing times to boil based on initial spent fuel pool temperatures for a period that would cover the expected duration of the pump repairs was developed by Engineering

The 1B KF pump was returned to service at 2200 on November 22 and following a period during which the performance of the 1B KF pump was monitored, it was declared operable at 1000 on November 23. The Risk Management Action plan terminated at that time.

Analysis: The performance deficiency associated with this issue was the less-than-adequate recognition, assessment, and management of the increased risk associated with the period of the Unit 1 refueling outage during which the single operable spent fuel pool cooling pump had failed with the opposite train's emergency diesel generator inoperable and the recently unloaded Unit 1 reactor core in the spent fuel pool placing the unit in a RED DID risk condition. Aspects which demonstrated this performance deficiency included the following:

- Operations and Outage Control Center personnel reviewing the guidance for the DID sheet were uncertain as to how the failure of the 1B KF pump combined with the inoperable 1A DG should be factored into an overall shutdown risk value. The initial assessment incorrectly assigned an Orange risk value to the plant conditions rather than the actual risk value of Red as defined in NSD 403.

- Station and corporate management were not notified in a timely manner once the Red DID condition was identified as required by NSD 417. Information on the pump failure, repairs and the elevated risk was provided through the normal outage communication channels.

- Maintenance and outage-related activities were allowed to proceed without additional oversight or review. Work on the nuclear service water piping which was the assured spent fuel pool makeup supply continued. In addition, this work was being performed in close proximity to KF system piping without any increased attention or protection put in-place. Ultrasonic cleaning of spent fuel and inspections of fuel assemblies in-progress in the spent fuel pool were allowed to continue with no communication of the elevated risk provided to the teams conducting the work. Access control into the spent fuel pool building was not implemented until approximately 20 hours after the failure of the 1B KF pump.

- Engineering did not provide Operations personnel with current spent fuel pool “time-to-boil” data for the period bounded by the unit shutdown and the KF pump repair activities until approximately 18 hours after the failure of the 1B KF pump.

- A comprehensive Risk Management Action Plan was not developed and implemented as required by station procedures for approximately 22 hours due to the primary focus being placed on the repair of the pump without the full consideration of other activities in progress or equipment requiring protection to control the increased risk caused by the 1B KF pump failure.

The finding was more than minor because the deficiency is consistent with IMC 0612, Appendix B, Section 3, Minor Screening Question (5)(i). Specifically, the licensee failed to expeditiously develop and implement risk management actions to address the elevated risk the unit was in based on the 1B KF pump failure and other equipment out of service or in an outage alignment; i.e., core in the spent fuel pool and the 1A DG disassembled. The inspectors completed a Phase 1 screening of the finding using Appendix K of IMC 0609, "Maintenance Risk Assessment and Risk Management Significance Determination Process," and determined that the performance deficiency represented a finding of very low risk significance (Green), based on the resulting magnitude of the calculated Incremental Core Damage Probability being below 1E-6. This was derived from discussions with the Region II Senior Reactor Analysts based on the time to boil in the Spent Fuel Pool being >24 hours which allows for operator actions to mitigate the effect of a postulated loss of cooling scenario. This finding has been entered into the licensee’s Corrective Action Program as PIPs C-06-7829 and C-06-7840. This finding directly involved the cross-cutting aspect of Human Performance under the “Safety Significant / Risk Significant Decisions” aspect of the “Decision Making” component, in that the licensee failed to adequately recognize, assess and manage the increased risk resulting from the failure of the 1B KF pump during outage conditions on Unit 1.

<u>Enforcement:</u> 10 CFR 50.65, “Requirements for monitoring the Effectiveness of Maintenance at Nuclear Power Plants”, paragraph (a)(4) requires in part, that prior to performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. NSD 403; Shutdown Risk Management (Modes 4, 5, 6 and No-Mode) per 10 CFR 50.65(a)(4), implements the requirements of 10 CFR 50.65(a)(4) during outage periods.

Contrary to the above, on November 21 and 22, 2006, the licensee did not recognize, assess, and manage the increased risk resulting from the failure of the single operable spent fuel pool cooling pump with the opposite train's emergency diesel generator inoperable and the recently unloaded Unit 1 reactor core in the spent fuel pool, a RED Defense In Depth risk condition, as required by 10 CFR 50.65 (a)(4).

Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as PIPs C-06-7829 and C-06-7840, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000413,414/2006005-02: Inadequate Recognition, Assessment and Management of the Increased Shutdown Risk Associated With the Failure of the 1B KF Pump with the Core in the Spent Fuel Pool and the 1A DG Inoperable.

1R15 Operability Evaluations

a. Inspection Scope

For the 12 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 to this report.

- PIP C-06-6676; Non-bounding Assumptions in Byron's Steam Generator Tube Rupture Analysis Resulted in Changes to Catawba's Operating Procedures
- PIP C-06-6839; Failure of Mode Selector switch on the 2B DG prevented the DG from being placed in the maintenance mode
- PIP C-06-6884; Scaffold clamp fell into the "A" RN valve pit during the disassembly of a scaffold
- PIP C-06-7126; 1A DG load unexpectedly increased to approximately 6837 kW during the performance of a post maintenance test
- PIP C-06-7748; Nuclear Service Water pond intake structure underwater inspection results
- PIP C-06-07604; Operability evaluation of 1B and 2B RN return headers following the cutting of 1B Containment Spray (NS) Heat Exchanger RN return line next to valve 1RN229B
- PIP C-06-7708; Containment Vessel Corrosion at Pipe Chase Floor
- PIP C-06-8309; Crack found on a turning vane in the 1A train of Auxiliary Building Ventilation and missing turning vanes on the 1B train of Auxiliary Building Ventilation
- PIP G-06-535; Torque wrench calibration sticker shows range that is unacceptable per QA inspection procedure

- PIP C-06-8041; Results of the 1A1 Component Cooling (KC) Pump head curve test indicate a possible pump interaction problem with the 1A2 KC Pump
- PIP C-06-8742; Hydrogen igniters 22, 42 and 46 failed SR 3.6.9.3 due to low temperature and a fuse was found failed in the power circuitry
- PIP C-06-8805; 1C Steam Generator main steam safety valve had minor leakage during restart activities

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

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

- Nuclear Station Modification CN 11441/00, Installation of main steam isolation valve air close assist upgrades and associated air manifold

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed 13 examples of the post-maintenance testing listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee's test procedures to verify that the procedures adequately tested the safety function(s) that may have been affected by the maintenance activities, that the acceptance criteria in the procedures 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 tests 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 to this report.

- PT/1/A/4350/002B; Diesel Generator 1B Operability Test, Rev. 110, following maintenance and activities on the 1B diesel generator
- PT/2/A/4250/003B; Auxiliary Feedwater Motor Driven Pump 2B Performance Test, Rev. 35, following maintenance activities on the 2B CA pump

- PT/2/A/4200/010A; Residual Heat Removal Pump 2A Performance Test; Rev. 46, following inspections and maintenance activities associated with the “2A” train of ND
- PT/1/A/4350/006A; 4160 Essential Power System Train A Test; Rev. 10, following maintenance activities on the 1A diesel generator
- PT/2/A/4350/002B; Diesel Generator 2B Operability Test; Rev. 85, following maintenance activities on 2B Diesel Generator
- PT/0/A/4400/022B; Nuclear Service Water Pump Train B Performance Test; Rev. 69, following maintenance activities on RN Pump 1B discharge check valve
- PT/1/A/4200/026; NS Valve Inservice Test; Rev. 57, following maintenance activities on valve 1NS-18A
- PT/1/A/4400/020; FW Valve Inservice Test, Enclosure 13.6, 1FW-55B Valve Inservice Test; Rev. 35
- Functional testing of the 1B KF pump following bearing seal ring replacement and associated repairs on November, 23, 2006
- PT/1A/4200/007A; Centrifugal Charging Pump 1A Test; Rev. 49, following maintenance activities on pump NV-1A
- Restoration to service and monitoring of performance of the 2A main feedwater pump following the repair of the turbine thrust bearing
- PT/1/A/4400/003 A; KC Train 1A Performance Test; Rev. 71, following the replacement of the 1A1 KC Pump rotating element
- PT/1/A/4350/002B; Diesel Generator 1B Operability Test; Rev. 110, Following maintenance on 1B DG during 1EOC16

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors performed the inspection activities described below for the Unit 1 refueling outage. The inspectors confirmed that, when the licensee removed equipment from service, the licensee maintained defense-in-depth commensurate with the outage risk control plan for key safety functions and applicable technical specifications, and that configuration changes due to emergent work and unexpected conditions were controlled in accordance with the outage risk control plan. Documents reviewed are listed in the attachment to this report.

- Reviewed the status and configuration of electrical systems to verify that those systems met TS requirements and the licensee’s outage risk control plan.
- Reviewed system alignments to verify that the flow paths, configurations, and alternative means for inventory addition were consistent with the outage risk plan.
- Reviewed the outage risk plan to verify that activities, systems, and/or components which could cause unexpected reactivity changes were identified in the outage risk plan and were controlled.
- Reviewed reactor coolant system (RCS) pressure, level, and temperature instruments to verify that the instruments provided accurate indication and that allowances were made for instrumentation errors.

- Observed decay heat removal parameters to verify that the system was properly functioning and providing cooling to the core.
- Reviewed selected control room operations to verify that the licensee was controlling reactivity in accordance with the technical specifications.
- Observed licensee control of containment penetrations to verify that the requirements of the technical specifications were met.
- Reviewed the licensee's plans for changing plant configurations to verify that technical specifications, license conditions, and other requirements, commitments, and administrative procedure prerequisites were met prior to changing plant configurations.
- Reviewed RCS boundary leakage and the setting of containment integrity.
- Examined the containment prior to reactor startup to verify that debris had not been left which could affect performance of the containment sumps.

b. Findings and Observations

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed the 17 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. The inspectors reviewed PIP C-06-3520, which had been initiated in May 2006 to address questions raised regarding the acceptance criteria contained in the surveillance procedure for testing the ice condenser lower inlet doors, along with past test data and an analysis performed by Westinghouse conducted to establish the design basis of the test's acceptance criteria. Documents reviewed are listed in the Attachment to this report.

Surveillance Tests
- PT/2/A/4400/006B; NS Heat Exchanger 2B Heat Capacity Test; Rev. 32
- PT/2/A/4350/002B; Diesel generator 2B Operability Test (24-hour run); Rev. 085
- PT/1/A/4400/006A; NS Heat Exchanger 1A Heat Capacity Test; Rev. 40
- PT/0/A/4150/030; RCCA Bank Repositioning; Rev. 20
- IP/1A/3200/001B; Solid State Protection Train B Periodic Testing; Rev. 2
- PT/2/A/4350/002B; Diesel Generator 2B Operability Test; Rev. 85
- IP/1/A/3200/001B; Solid State Protection System (SSPS) Train B Periodic Testing; Rev. 2
- MP/0/A/7150/072; Main Steam Safety Valve Setpoint Test; Rev. 17
- PT/1/A/4350/002B; Diesel Generator 1B Operability Test; Rev. 110

- PT/2/A/4150/001 D; Reactor Coolant System Leakage Calculation; Rev. 59
- PT/1/A/4200/009; Engineered Safety Features Actuation Periodic Test; ‘B’ Train - Blackout Only; Rev. 176

In-Service Tests
- PT/2/A/4200/004C; Containment Spray Pump 2B Performance Test; Rev. 32

Ice Condenser Systems Testing
- MP/0/A/7150/006; Ice Condenser Lower Inlet Door Inspection and Testing (As-Found); Rev. 27
- MP/0/A/7150/006; Ice Condenser Lower Inlet Door Inspection and Testing (As-Left); Rev. 28

Containment Isolation Valve Testing
- PT/1/A/4200/041D; Containment Hydrogen Sample and Purge Isolation Valve Leak rate Test; Rev. 13 - Testing of Penetration 332 for 1VY15B and 1VY16
- PT/1/A/4200/001I; Containment Isolation Valve Leak Rate Test of 1RN-485, Rev. 10 - Testing of Penetration M230
- PT/1/A/4200/001I; Containment Isolation Valve Leak Rate Test of 1NV-90 (NC Pumps Return Containment Isolation Relieving Check Valve), Rev. 10 - Testing of Penetration M256

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion III, Design Control; and Criterion XI; Test Control, for the licensee’s failure to have design documentation to support the ice condenser lower inlet door surveillance procedure test acceptance limits.

Description: In May of 2006, the inspectors reviewed completed lower inlet door test data contained in procedure MP/0/A/7150/006; Ice Condenser Lower Inlet Door Inspection and Testing, and questioned the calculated frictional torque value documented in the surveillance procedure that was used to satisfy T.S. requirement 3.6.13.6. On several of the doors tested, the measured closing torque value was greater than the measured opening torque value which resulted in a calculated negative frictional torque value; i.e., Frictional Torque = Opening Torque - Closing Torque. The acceptance criteria in the T.S. bases and the plant surveillance procedure for the calculated frictional torque was stated as “≤40 in-lbs.” The inspectors requested the design bases documents that supported the acceptability of calculated frictional torque values being negative; however, the licensee was not able to provide any design documents that supported the ability to accept negative frictional torque values as satisfying the TS surveillance requirement.

As a result of the questions raised by the inspectors, the licensee contracted with the vendor, Westinghouse, to develop a formal design document that provided technical justification for the opening, closing and frictional torque values contained in the station’s surveillance procedures. The calculation was received by station engineering personnel and incorporated into the Catawba UFSAR, ice condenser surveillance procedure, and TS Bases document in December of 2006.

Analysis: The function of the ice condenser is to protect containment integrity by dissipating the heat from a design basis accident. The design documents for the ice condenser should contain the basis used to establish the acceptance criteria contained in the surveillance tests performed to verify the operability of the ice condenser lower inlet doors under the full range of postulated accident scenarios. The test acceptance limits in the surveillance test procedure being used at Catawba to verify torque values were acceptable at the 40-degree open position did not contain a lower bound and as a result, did not provide assurance that operability could be assured over the full range of calculated values if negative numbers were obtained.

The inspectors concluded that the finding was greater than minor following the review of IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," in that an excessively high closing torque could adversely impact the ability of the lower inlet door to modulate properly in the event of a small-break LOCA; however, with no lower limit for negative values in the surveillance test's acceptance criteria, this condition might not have been identified and corrected prior to returning the unit to power operation. The finding is associated with the Barrier Integrity cornerstone and is associated with the integrity of the reactor containment structure; i.e., the ice condenser's ability to control internal pressure following a LOCA event, to protect the public from radio-nuclide releases.

The issue was determined to be of very low safety significance (green) because the acceptance criteria contained in the design bases document received from the vendor bounded the calculated frictional torque values that had been recorded during the performance of past LID surveillance tests on both Catawba units. This finding directly involved the cross-cutting area of Human Performance under the "Complete Documentation and Component Labeling" aspect of the "Resources" component, in that the licensee failed to maintain complete, accurate and up-to-date design documentation and procedures.

Enforcement: 10CFR 50 Appendix B Criterion III, Design Control, requires, in part, that measures be established to assure that the design basis as defined in 10 CFR 50.2 for safety-related structures, systems and components is correctly translated into specifications, procedures and instructions.

10CFR 50 Appendix B Criterion XI, Test Control, requires, in part, that a test program shall be established and performed in accordance with written procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Contrary to the above, prior to November 2006, the licensee failed to establish a technical bases for the acceptance criteria used to satisfy the Technical Specification surveillance requirements pertaining to the 40-degree torque testing of the ice condenser lower inlet doors.

The failure to have test acceptance criteria based on design documents for the ice condenser lower inlet doors is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000413,414/2006-05-03: Failure to implement adequate design control for ice condenser lower inlet doors. This issue has been entered into the licensee's corrective action program as PIPs C-06-3520 and C-06-7212.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following two temporary station modifications to determine whether the individual modification was properly installed; the modification did not affect system operability, drawings and procedures were appropriately updated; and post-modification testing was satisfactorily performed. Documents reviewed are listed in the Attachment to this report.

- Minor Modification CD101222, Digital Rod Position Indication (DRPI) cable for Shutdown bank C, E-3 position to bypass bulkhead connector and directly tie the DRPI data cabinet with the coil stack
- Minor Modification CD 101223, Relocate processor board from primary to secondary acoustic monitor for Pressurizer Relief Valve NC001 acoustic leak detection system

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupational Radiation Safety (OS) Cornerstone

2OS1 Access Controls To Radiologically Significant Areas

.1 Access Controls

a. Inspection Scope

Licensee program activities for monitoring workers and controlling access to radiologically significant areas and tasks were inspected. The inspectors evaluated procedural guidance; directly observed implementation of administrative and established physical controls; assessed worker exposures to radiation and radioactive material; and appraised radiation worker and technician knowledge of, and proficiency in implementing radiation protection program activities.

During the inspection, radiological controls for selected operations and maintenance activities were observed and discussed. Briefings and/or radiation control field activities were observed for on-going maintenance, plant modifications and refueling preparatory work. Inspector evaluations included, as applicable, Radiation Work Permit (RWP) details, use and placement of dosimetry and air sampling equipment, electronic dosimeter set-points, and monitoring and assessment of worker dose from direct radiation and airborne radioactivity source terms. Effectiveness of established controls were assessed against area radiation and contamination survey results, and occupational doses received. Physical and administrative controls and their implementation for extra-high radiation area locations and for storage of highly activated material within the spent fuel pool areas were evaluated through discussions with licensee representatives, direct field observations, and record reviews.

Occupational worker adherence to selected RWPs and Health Physics Technician proficiency in providing job coverage were evaluated through direct observations of staff performance during job coverage and routine surveillance activities, review of selected exposure records and investigations, and interviews with licensee staff. Radiological postings and physical controls for access to designated high radiation areas and extra high radiation area locations within auxiliary building and spent fuel pool areas were evaluated during facility tours. In addition, the inspectors independently measured radiation dose rates and evaluated established posting and access controls for selected auxiliary building and containment locations and equipment including waste storage facilities; liquid waste processing; outdoor radioactive waste storage and upper containment general areas. Occupational exposures associated with direct radiation, potential radioactive material intakes, and from discrete radioactive particle or dispersed skin contamination events for calendar year (CY) 2005 and year-to-date 2006 were reviewed and discussed.

Radiation protection program activities were evaluated against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; UFSAR details in Section 12, Radiation Protection; TS Sections 5.4, Procedures; and 5.7, High Radiation Area; and approved licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in Sections 2OS1 of the report Attachment.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

Licensee Corrective Action Program (CAP) documents associated with access controls to radiologically significant areas were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with NSD-208, Problem Investigation Process (PIP), Rev. 27. Licensee audits, self-assessments and PIP documents related to access controls that were reviewed and evaluated in detail for this program area are identified in Section 2OS1 of the report Attachment.

The inspectors completed 21 of 21 required samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

.1 As Low As Is Reasonably Achievable (ALARA)

a. Inspection Scope

The inspectors evaluated ALARA program guidance and its implementation for ongoing Refueling Outage tasks. The inspectors reviewed and discussed with licensee staff, ALARA work plan documents including dose estimates and prescribed ALARA controls for selected outage work activities expected to incur significant collective doses. This reviews and discussions included ECCS Sump Mod, PZR Alloy 600 Weld Overlays, Shielding, Mechanical Valves, and Reactor Head Work activities. The inspectors reviewed the implementation of dose reduction initiatives for high person-rem expenditure tasks. These elements of the ALARA program were evaluated for consistency with the methods and practices delineated in applicable licensee procedures.

The implementation and effectiveness of ALARA planning and program initiatives during work in progress were evaluated. The inspectors made direct field or remote video observations of Unit 1 work activities involving: sump mod; pressurizer allow 600 weld overlay, preparations of radioactive shipments; and on-going work in the auxiliary building. The inspectors interviewed radiation workers and Health Physics Technician staff to assess their understanding of dose reduction initiatives and their current and expected final accumulated occupational doses at completion of the task.

Projected RWP dose expenditure estimates were compared to actual dose expenditures, and noted differences were discussed with cognizant ALARA staff. These estimate vs. actual dose expenditures comparisons covered calendar year (CY) 2005 and 2006 from January to October. Changes to dose budgets relative to changes in job scope also were identified and discussed. The inspectors attended pre-job briefings and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel. In addition, the inspectors reviewed air sampling results and internal dosimetry assessments for adequacy of respiratory protection and engineering controls.

Implementation and effectiveness of selected program initiatives with respect to source-term reduction were evaluated. Shutdown chemistry program actions and cleanup initiatives, including their resultant effect on containment vessel and auxiliary area and equipment dose rate trending data were reviewed and compared to previous refueling outage data. The effectiveness of selected shielding packages installed for the current outage was assessed through completion of independent radiation surveys and comparison to applicable licensee survey records and expected planning data.

The plant collective exposure histories for CY 2004 and 2005, taken from data reported to the NRC pursuant to 10 CFR 20.2206, were reviewed and discussed with licensee staff, as were established goals for reducing collective exposure. The inspectors reviewed the applicable guidance and examined dose records of declared pregnant workers during CYs 2005 and 2006 to evaluate current gestation doses for declared pregnant workers.

ALARA activities were evaluated against the requirements specified in 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; and approved licensee procedures. In addition, licensee performance was evaluated against Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable, and Regulatory Guide 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Section 2OS2 of the report Attachment.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

Licensee corrective action documents associated with ALARA activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with the corrective action program. Specific self-assessments and audits were reviewed and evaluated in detail for this inspection area are identified in Section 2OS2 of the report Attachment.

The inspectors completed 18 samples. (14 required samples and 4 optional samples based on three year rolling average occupational radiation exposure ranking.)

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation

.1 Waste Processing and Characterization

a. Inspection Scope

The inspectors reviewed and discussed the currently installed radioactive waste (radwaste) processing system as described in the UFSAR Section 11. In addition, stored and disposed radwaste types and quantities as documented in Effluent Release Reports for CYs 2004 and 2005 were discussed with responsible licensee representatives.

The operability and configuration of selected liquid and solid radioactive radwaste processing systems and equipment were evaluated. Inspection activities included document review, interviews with plant personnel, and direct inspection of processing equipment and piping. The inspectors directly observed equipment material condition and configuration of liquid and solid radwaste processing systems. The radwaste processing equipment was inspected for general condition and licensee staff was interviewed regarding equipment function and operability. The licensee's policy regarding abandoned radwaste equipment was reviewed and discussed with cognizant licensee representatives. Chemistry staff was interviewed to assess knowledge of radwaste system processing operations. Procedural guidance involving transfer of resin and filling of waste packages was reviewed for consistency with the licensee's Process Control Program and UFSAR details.

Licensee radionuclide characterizations of each major waste stream were evaluated. For dry active waste, primary resin, secondary resin, and filters, the inspectors evaluated Process Control Program and licensee procedural guidance against 10 CFR 61.55 and the Branch Technical Position on Radioactive Waste Classification details. Part 61 data and scaling factors were reviewed and discussed with licensee representatives for radwaste processed or transferred to licensed burial facilities for the January 1, 2005, through November 16, 2006, period. The licensee's analyses and current scaling factors for quantifying hard-to-detect nuclides were assessed. The inspectors discussed potential for changes plant operating conditions and reviewed selected dry active waste stream radionuclide data to determine if known plant changes were assessed and radionuclide composition remained consistent for the period reviewed.

<u>Transportation</u> The inspectors evaluated the licensee's activities related to transportation of radioactive material. The evaluation included review of shipping records and procedures, assessment of worker training and proficiency, and direct observation of shipping activities.

The inspectors assessed shipping-related procedures for compliance to applicable regulatory requirements. Selected shipping records were reviewed for completeness, accuracy, and consistency with licensee procedures. Training records for individuals qualified to ship radioactive material were checked for completeness. In addition, training curricula provided to these workers were assessed. On November 15, 2006, the inspectors observed the preparation and documentation of a shipment of plant equipment from a vendor. The inspectors directly observed package closure and the performance of radiation surveys for the shipment as well as the preparation of shipment documentation. Responsible staff were interviewed to assess their knowledge of package radiation and contamination controls and applicable limits.

Transportation program guidance and implementation were reviewed against regulations detailed in 10 CFR 71, and 49 CFR 170-189 and applicable licensee procedures listed in the Appendix to this report. In addition, training activities were assessed against 49 CFR 172 Subpart H, and the guidance documented in NRC Bulletin 79-19.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

Licensee CAP documents associated with radwaste processing and transportation activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with NSD - 208, Problem Investigation Process, Rev. 12. Specific assessments and PIP documents reviewed in detail for this inspection area are identified in Section 2PS2 of the report Attachment.

The inspectors completed six of six required samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee data to verify the accuracy of reported performance indicator (PI) data for the periods listed below. To verify the accuracy of the reported PI elements, the reviewed data was assessed against PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Indicator Guideline, Rev. 3.

Initiating Events

- Unplanned Scrams per 7,000 Critical Hours, Unit 1; 4th quarter 2004 through 3rd quarter 2006

The inspectors reviewed the Unplanned Scrams per 7,000 Critical Hours for the period of October 1, 2004 through September 30, 2006 for Unit 1. The inspectors reviewed PIP's and LER's associated with reactor scrams that occurred in that period and verified that the data reported for the PI corresponded to the number of critical hours and reactor scrams that occurred.

Barrier Integrity

- Reactor Coolant System Leakage Calculation; Unit 2; 4th quarter 2004 through 3rd quarter 2006

The inspectors reviewed the Reactor Coolant System Leakage PI results for the period of October 1, 2004 through September 30, 2006 for Unit 2. The inspectors reviewed selected leakage calculation results recorded in the control room logs for Unit 2 and observed the performance of the leak rate calculation surveillance. Documents reviewed are listed in the report Attachment.

Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Effluent Occurrence

The inspectors reviewed the PI results for the period January 2006 through November 2006. The inspectors reviewed selected out of service effluent radiation monitor and compensatory sampling data, abnormal release results as reported in the 2004 and 2005 Annual Effluent Reports, and selected PIP documents related to Radiological Effluent Technical Specification/Offsite Dose Calculation Manual issues. In addition, the inspectors reviewed cumulative and projected doses to the public for the period January 2006 through October 31, 2006. Documents reviewed are listed in Section 4OA1 of the report Attachment.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspectors reviewed the PI results for the period January 2006 through October 2006. For the assessment period, the inspectors reviewed electronic dosimeter alarm records and PIP documents related to controls for exposure significant areas. The reviewed documents reviewed are listed in Sections 2OS1 and 4OA1 of the report Attachment.

The inspectors performed two samples related to radiation protection.

b. Findings

No findings of significance were identified.

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.

.2 Annual Sample Review

a. Inspection Scope

The inspectors reviewed licensee actions to correct two (2) issues determined to require additional attention by station personnel. These issues were 1) quality and consistency of control room logkeeping and 2) installation and usage of scaffolding near safety-related Structure, Systems and Components (SSC's).

Based on the observations made by the inspectors during routine reviews of control room logs and subsequent reviews conducted by licensee personnel, the station determined that control room logs were not being maintained in accordance with Operations Management Procedures governing this activity. The licensee increased the focus on log quality and provided crew-specific coaching when expectations were not being met. The inspectors reviewed the actions implemented by the licensee and assessed how control log quality was affected by these actions.

The review looked at improperly installed scaffolds near safety-related SSC's that have been identified over the past 18 months and the corrective actions taken by the licensee to address this issue. Inspectors reviewed corrective action documents, the corporate Duke Power Scaffold manual and plant-specific procedures used in the implementation of the scaffold program. In addition, personnel involved in the erection and periodic inspections of scaffolding were interviewed to determine what actions had been implemented and their effectiveness. The inspectors evaluated the scaffolding events and associated corrective actions against the requirements of the licensee's corrective action program and 10 CFR 50, Appendix B.

Additionally, the inspectors reviewed the cumulative effects of deficiencies that constituted operator workarounds to determine whether or not they could affect the reliability, availability, and potential for mis-operation of a mitigating system; affect multiple mitigating systems; or affect the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors also assessed whether operator workarounds were being identified and entered into the licensee's corrective action program at an appropriate threshold. Documents reviewed are listed in the attachment.

b. Findings

Introduction: The inspectors identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for the licensee's failure to identify and implement effective corrective actions to prevent recurring deficiencies associated with the erection of scaffolding around safety related equipment.

Description: During the May 2005 Unit 1 refueling outage, NRC inspectors identified three (3) examples of scaffolds that had been erected outside of the requirements specified in the Duke Power Scaffold Manual near safety-related equipment. An assessment of these specific deficiencies determined that while they were violations of a procedure that implemented the requirements of Reg Guide 1.33 for performing maintenance on safety-related equipment, they did not rise above the "more than minor" level based on subsequent engineering evaluations that were performed by the licensee's Civil Engineering group. The licensee did take immediate actions to correct the deficiencies and initiated corrective actions to preclude recurrence.

Over the next six months, additional examples of improperly erected scaffolds were identified by the NRC resident inspectors during routine plant tours. As was done for the examples noted during the Unit 1 refueling outage, the licensee immediately corrected the condition and again, looked at corrective actions to address the practice of scaffold construction.

Based on the number of scaffold construction issues in the area of safety-related equipment, Maintenance initiated trend PIP C-05-6888 in November 2005 to determine the underlying causes for the repeated events and develop additional corrective actions to prevent recurrence. This PIP documented eight specific examples of scaffolds that had been improperly constructed between May and October 2005. All of the corrective actions developed from this trend PIP were completed by July 2006.

During the summer of 2006, NRC inspectors identified two examples of scaffolds that were improperly constructed and had the potential to adversely impact safety-related equipment. The licensee initiated PIPs on these occurrences, immediately corrected the condition and performed an engineering analysis to ensure that the equipment it had been in contact with had not been rendered inoperable during the time the scaffolds had been configured improperly.

On December 1, 2006, the resident inspectors performed a walkdown of equipment that had been designated as "Protected Equipment" to support work being performed on the "B" train of RN. Among the items being protected were the KC pumps. The inspectors identified that scaffolding erected to support work on valve 1KC-82B was improperly installed despite the scaffold tag stating that all requirements for scaffolding near safety-related equipment had been met and verified. One of the scaffolding cross bars was secured in contact with the 1inch KC pump motor cooler inlet line and a large unsecured scaffold plank was laying across the scaffold frame adjacent to valve 1KC-82B. The condition was brought to the attention of the Work Control Center Senior Reactor Operator who initiated actions to correct the condition and generated PIP C-06-8183 to document the condition. The scaffold was repaired; however, the PIP simply stated that the repair had been made and the supervisor was coached on the importance of scaffolding. No additional corrective actions were identified nor was an engineering evaluation performed to determine the potential impact the improperly installed scaffold cross bar had on the KC line.

<u>Analysis</u>: The inspectors determined that the licensee's repeated failure to erect scaffolding in accordance with the Duke Scaffold Manual and implement effective corrective actions to prevent recurrence was a performance deficiency. In accordance with Appendix B, "Issue Screening," of IMC 0612, the inspectors determined that the finding was of more than minor significance since the finding was associated with the equipment performance and human performance attributes of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of equipment that respond to initiating events to prevent undesirable consequences is maintained. The finding was determined to be of very low safety significance because, while improperly installed scaffolding has the potential to adversely affect mitigation systems, the specific examples identified over the last 18 months did not result in an actual loss of safety function of a mitigating system and did

not render equipment inoperable due to a seismic event. This finding directly involved the cross-cutting area of Problem Identification and Resolution under the “Appropriate and Timely Corrective Actions” aspect of the “Corrective Action Program” component, in that ineffective corrective actions were established resulting in additional scaffolding deficiencies being identified over an 18 month period.

Enforcement: 10 CFR, Part 50, Appendix B, Criterion XVI, Corrective Action, states in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this, the licensee has failed to develop and implement effective corrective actions to address long-standing issues related to the erection of scaffolding in the vicinity of safety-related equipment as documented in numerous PIPs. This included a site-wide trend PIP which was closed following the implementation of actions that were intended to address the increasing number of scaffolding events, which could potentially impact safety-related components. Because this finding is of very low safety significance and because it has been entered into the licensee’s corrective action program as PIP C-06-8183, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000413, 414/2006005-04: Failure to Prevent Recurring Scaffolding Installation Deficiencies.

.3 Semi-Annual Review to Identify Trends

.a Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee’s CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors’ review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screenings discussed in section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspectors’ review primarily considered the six month period of July 2006 through December 2006, although some examples expanded beyond those dates when the scope of the trend warranted, particularly in the area of trends identified in previous inspection reports and carried forward to monitor the licensee’s progress. The review also included issues documented outside the normal CAP in major equipment problem lists, plant health team vulnerability lists, Catawba focus area reports, system health reports, self-assessment reports, maintenance rule reports, and Safety Review Group Monthly Reports. The inspectors compared and contrasted their results with the results contained in the licensee’s latest quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensees trend report were reviewed for adequacy.

b. Assessment and Observations

Oversight and Control of Vendors and Contractors Trend Statement

The inspectors followed the actions being implemented by the licensee in response to the inspector-identified trend associated with insufficient management oversight and control of vendors and contractors (non-station personnel). This trend statement was discussed in the following NRC Inspection Reports: 05000413,414/2005005 and 05000413,414/2006003, section 4OA2.3, Semi-Annual Trend Review. Based on the inspectors identification of this trend, the licensee had concluded that a lack of guidance existed in the Duke Nuclear Site Directive 105, Vendor Oversight and Control procedure. The licensee stated in corrective action documentation that this was evident in large projects undertaken at Catawba during the service water project and at Oconee during the steam generator replacement project. Senior Management decided to incorporate specific decision points into the planning and approval process for major projects to ensure oversight controls are considered and developed as part of an overall project development plan. As an interim corrective action, additional oversight has been placed on large projects at Catawba. Despite these interim actions, vendor support has continued to provide challenges to the Catawba organization. For example, the installation of the Unit 1 ECCS sump modification required by the NRC to be completed by the end of 2007 was not ready to be installed as intended during the Fall 2006 refueling outage. The two major contributors to this delay and subsequent request for an implementation extension from the NRC were 1) a less-than-adequate design received from the contractor in charge of the sump design one month from the start of the outage requiring changes to the design, and 2) the failure to receive the sump components due to delays at the fabrication facility which were not identified until just prior to the start of the outage when they were required. Station management has recognized the need for additional attention in the area. Comprehensive changes to the nuclear department's process for controlling vendor-led projects and providing the necessary oversight are planned for the 1st quarter of 2007. The residents will continue to monitor actions taken in this area for improvement in the control and oversight of contractor and vendor personnel conducting work at Catawba.

4OA3 Event Follow-up

.1 Failure of the Unit 1 "A" DG #4 Bearing

a. Inspection Scope

On November 24, 2006, the 1A DG was being run following routine maintenance performed during the refueling outage. Shortly after it was started, the journal bearing for the #4 piston failed catastrophically and the DG was tripped manually by the non-licensed operators conducting the test. The inspectors responded to the site and discussed the failure with engineering, operations and station management personnel. The engine was partially disassembled in order to determine the cause of the failure and repairs were initiated with the assistance of the engine manufacturer and additional contract personnel.

The licensee implemented a Risk Management Action plan to minimize the risk exposure to the plant while repairs were being completed on the 1A DG. The inspectors monitored the repair activities and verified that the risk management actions remained in-place until the 1A DG was returned to service and declared operable.

b. Findings

No findings of significance were identified.

.2 Declaration of an Unusual Event due to Excessive RCS leakage on Unit 1

a. Inspection Scope

On December 27, 2006, Unit 1 was in Mode 3 and increasing reactor coolant system temperature and pressure to support restart from the 1EOC16 refueling outage. At approximately 1215 on December 27, 2006, Unit 1 operators detected an increasing trend in the Reactor Coolant Drain Tank (NCDT) level. Reactor coolant pump (RCP) standpipe levels started to fluctuate and the 1B RCP standpipe level went low. Operators initially diagnosed this as a possible problem with or failure of the #1 pump seal and tripped the 1B RCP. They entered the appropriate abnormal operating procedures for the pump issues and transitioned to AP-10; RCS Leakage, when the NCDT went solid and lifted the relief valve. A Notification of Unusual Event (NOUE) was declared at 1235 based on RCS leakage exceeding 25 gpm. All required notifications to offsite agencies were made within the requisite time frames. The resident inspectors responded to the event and monitored the station's response from the control room and Technical Support Center.

A review of potential input sources to the NCDT led the station to direct the non-licensed operators to check several valves inside of containment and two in-series 2-inch loop drain valves were found to be not fully closed as required. The valves were closed and RCS parameters quickly stabilized. The NOUE was exited at 1600 after conducting additional valve lineup checks, restarting the 1B RCP and performing a transient leak rate calculation.

A similar event occurred in 1997 at Catawba and was reported in Licensee Event Report (LER) 413/1997-011. The licensee is conducting a Root Cause Investigation to determine the cause of the event.

b. Findings

No findings of significance were identified.

.3 (Closed) LER 0500413/2006001-00, Loss of Offsite Power Event Resulted in Reactor Trip of Both Catawba Units from 100% Power

On May 20, 2006, both Catawba units tripped automatically from 100% power following a Loss of Offsite Power event. The event was initiated by the failure of a current transformer in the switchyard which caused a perturbation that cleared both incoming buslines and separated the units from the grid. A NOUE was declared and the Technical Support Center, Operations Support Center and Emergency Operations Facility locations were activated. Offsite power was restored approximately 12 hours after the start of the event and both units were returned to service following the completion of required equipment repairs. An NRC Augmented Inspection Team was dispatched to assess the event and the results of this inspection are documented in Inspection Report 05000413,414/2006-009. The initial notification of the NRC within the required one (1) hour time period was not met by the licensee during this event and is discussed in this LER; however, a Green Licensee-Identified Violation for this issue is contained in IR 05000413,414/2006-003. The LER was reviewed by the inspectors and no additional findings of significance were identified and no additional violations of NRC requirements occurred. The licensee documented the transient, failed equipment and corrective actions in PIP C-06-3864 (Unit 1) and PIP C-06-3865 (Unit 2). This LER is closed.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction (TI) 2515/169, Mitigating Systems Performance Index (MSPI) Verification

a. Inspection Scope

During this inspection period, the inspectors completed a review of the licensee's implementation of the MSPI guidance for reporting unavailability and unreliability of monitored safety systems in accordance with TI 2515/169.

The inspectors examined surveillances that the licensee determined would not render the train unavailable for greater than 15 minutes or during which the system could be promptly restored through operator action and therefore, are not included in unavailability calculations. As part of this review, the recovery actions were verified to be uncomplicated and contained in written procedures.

On a sample basis, the inspectors reviewed operating logs, work history information, maintenance rule information, corrective action program documents, and surveillance procedures to determine the actual time periods the MSPI systems were not available due to planned and unplanned activities. The results were then compared to the baseline planned unavailability and actual planned and unplanned unavailability determined by the Licensee to ensure the data's accuracy and completeness. Likewise, these documents were reviewed to ensure MSPI component unreliability data determined by the licensee identified and properly characterized all failures of monitored components. The unavailability and unreliability data were then compared with performance indicator data submitted to the NRC to ensure it accurately reflected the performance history of these systems.

b. Findings and Observations

No findings of significance were identified.

The licensee accurately documented the baseline planned unavailability hours, the actual unavailability hours and the actual unreliability information for the MSPI systems. No significant errors in the reported data were identified, which resulted in a change to the indicated index color. No significant discrepancies were identified in the MSPI basis document which resulted in: (1) a change to the system boundary, (2) an addition of a monitored component, or (3) a change in the reported index color.

.2 (Closed) NRC Temporary Instruction 2515/150, Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Order EA-03-009) (Unit 1)

a. Inspection Scope

From November 20 to December 1, 2006 the inspectors reviewed the licensee's activities relative to the NDE of the reactor pressure vessel head (RPVH) nozzles, the bare metal visual (BMV) examination of the RPVH nozzles and head surface area, and the visual examination to identify potential boric acid leaks from pressure-retaining components above the RPVH. These activities were reviewed during the Unit 1-Fall 2006 refueling outage, in order to verify licensee compliance with the regulatory requirements of NRC Order EA-03-009 Modifying Licenses dated February 20, 2004 (hereinafter the NRC Order) and gather information to help the NRC staff identify possible further regulatory positions and generic communications.

The inspectors' review of the NDE of RPVH nozzles included: a) review of NDE procedures; b) assessment of NDE personnel training and qualification; c) review of NDE equipment certification and performance demonstration; and d) observation and assessment of ultrasonic (UT) and surface penetrant test (PT) examinations. The inspectors also held discussions with contractor representatives (Areva) and licensee personnel involved in the RPVH examination. Specifically, the inspectors reviewed a sample of NDEs as follows:

- Observed portion of in-process UT scanning of RPVH nozzle No. 29
- Reviewed the UT data sheets and electronic data for RPVH nozzle Nos. 8, 18, 32, 42, 70, 75, and 76
- Reviewed the electronic UT data and PT data sheet for the RPVH vent line penetration
- Reviewed the results of the UT examination performed to assess for leakage into the annulus between the RPVH penetration nozzle and the RPVH low-alloy steel (interference fit zone) for penetration Nos. 8, 18, 32, 42, 70, 75, and 76
- Reviewed training and qualification records, including qualification and certification procedures, for NDE personnel who performed the above volumetric and surface examinations
- Reviewed certification, performance demonstration, and calibration records for NDE equipment used to perform the above volumetric examinations
- Reviewed Areva's examination procedures used to perform the above volumetric and surface examinations.

The inspectors' review of the BMV examination for the RPVH nozzles and head surface area included: a) review of procedures used to perform the examination; b) assessment of personnel training and qualification; c) direct observation of portion of the examination; and d) review of final report and disposition of indications.

The inspectors' review of the visual examination to identify potential boric acid leaks from pressure-retaining components above the RPVH consisted of the review of licensee procedures used to meet this requirement and the results from the visual examinations performed in the Unit 1-Fall 2006 refueling outage.

The inspectors also reviewed the licensee's effective degradation years calculation, which was performed to determine the RPVH's susceptibility category and its examination requirements.

b. Observations and Findings

1) Verification that the examinations were performed by qualified and knowledgeable personnel.

The inspectors reviewed personnel training and qualifications to verify that volumetric and surface NDEs were performed by trained and qualified personnel. All examiners were qualified in accordance with the ASME Code and had additional training on RPVH examination, as required in Areva's "Written Practice for the Qualification and Certification of NDE Personnel" document.

2) Verification that the examinations were performed in accordance with approved and demonstrated procedures.

Catawba's RPVH (Unit 1) has 78 control rod drive mechanism (CRDM) penetrations and 1 vent line penetration. Fifty three (53) of the 78 penetrations contain thermal sleeves and the remaining 25 penetrations have open bores. All penetration nozzles, including the vent line, were examined by remote automated UT from the inside diameter surface in accordance with Areva approved procedures 54-ISI-604-001 for open bore penetrations, 54-ISI-603-002 for sleeved penetrations, and 54-ISI-605-01 for small bore penetrations.

In addition to the CRDM and vent line penetrations, Catawba's RPVH has 4 auxiliary head adapter penetrations. These penetrations consist of an Alloy 600 nozzle welded to the top of the RPVH with a dissimilar metal full penetration weld. These welds were not examined as part of the NDEs required to meet the NRC Order. However, these welds were included within the scope of the Inservice Inspection Program as required by Section XI of the ASME Code.

RPVH penetrations with thermal sleeves and some open bore penetrations were examined with the Time of Flight Diffraction (TOFD) technique using a blade probe containing one set of 50 degree/5 MHz/L-Wave transducers circumferentially oriented for axial flaws (COAF). The transducer set was contained in a single inspection housing. Assessment of leakage into the interference fit zone was employed by analyzing the pattern and amplitude of the backwall reflection from the TOFD transducers set up.

RPVH thermocouple penetrations (open bore) were examined with a 0 degree/5 MHz/L-Wave transducer, one TOFD set of 30 degree/5 MHz/L-Wave transducers axially oriented for circumferential flaws, one set of 60 degree/2.25 MHz/S-Wave transducers axially oriented for circumferential flaws, one TOFD set of 45 degree/5

MHz/L-Wave transducers COAF, and one set of 60 degree/2.25 MHz/S-Wave transducers COAF. All transducer sets were contained in a single rotating inspection housing. Assessment of leakage into the interference fit zone was employed by analyzing the pattern and amplitude of the backwall reflection from the TOFD and 0 degree transducers set up.

The vent line penetration nozzle was examined with a set of 0 degree/5 MHz/L-Wave transducers, one set of 45 degree/5 MHz/S-Wave transducers (CW and CCW beam direction), and one set of 70 degree/5 MHz/S-Wave transducers (up and down beam direction). All transducer sets were contained in a single rotating inspection housing. Assessment of leakage through the J-groove weld was employed by performing a PT examination on the surface of the J-groove weld in accordance with Areva procedure 54-PT-200-06.

The inspectors found that Areva examination procedures for CRDM nozzles were demonstrated to be able to detect and size flaws in the RPVH nozzles in accordance with Electric Power Research Institute (EPRI) NDE Center's protocol contained in "Materials Reliability Program: Demonstration of Vendor Procedures for the Inspection of Control Drive Mechanism Head Penetrations (MRP-89)." Areva's equipment demonstration took place from August 14 to August 24, 2006. Areva had performed a similar demonstration in 2002, as documented in MRP-89. However, because Areva modified its equipment including changing the essential variables of the demonstration in 2002, the demonstration was repeated. The 2006 demonstration was performed with three RPVH nozzle mockups with multiple tube flaws representing the expected field degradations. These mockups were different from the ones used during the demonstration performed in 2002 (i.e. demonstration documented in MRP-89). The demonstration adopted security portions from the EPRI Performance Demonstration Initiative protocol by restricting the access to the mockups and making them available to Areva only when the EPRI NDE personnel were present. EPRI letter to Duke Energy Corporation, dated October 19, 2006, documents the comparison of the recent Areva's equipment demonstration with the previous demonstration performed in 2002.

The letter states that the scatter observed is within the variability of the examination and the reliability of the examinations conducted with the new instrumentation will be comparable to the previous demonstration.

The procedure used for the RPVH vent line was not demonstrated under a specific program such as the EPRI MRP. This procedure was developed with NDE techniques similar to the CRDM procedures with regard to basic fundamental ultrasonic requirements. The procedure used for the PT examination of the vent line weld surface was developed in accordance with the ASME Code.

3) Verification that the licensee was able to identify, disposition, and resolve deficiencies.

All indications of cracks or interference fit zone leakage were required to be reported for further examination and disposition as specified in Areva's NDE procedures. Based on observation of the examination process and discussions with vendor's personnel, the inspectors considered that deficiencies would be appropriately identified, dispositioned, and resolved. UT indications associated with the fabrication of the J-groove weld and nozzle tube material were identified at several RPVH penetrations. These indications did not exhibit service related crack characteristics and were documented for future reference. Two surface PT indications in the vent line J-groove weld required repair. In addition, the licensee identified and repaired a loose guide funnel in a thermocouple penetration (see observation 7).

4) Verification that the licensee was capable of identifying the primary water stress corrosion cracking (PWSCC) and/or RPVH corrosion phenomenon described in the NRC Order.

   The NDE techniques employed for the examination of RPVH CRDM nozzles had been previously demonstrated under the EPRI MRP/Inspection Demonstration Program as capable of detecting PWSCC type manufactured cracks. Based on the review of performance demonstration documents, observation of in-process examinations, and review of NDE data, the inspectors considered that the licensee was capable of identifying PWSCC and/or corrosion as required by the NRC Order.

5) Evaluation of the RPVH condition (e.g. debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions).

   A BMV examination was performed per licensee's procedure MP/0/A/7150/042D by engineering personnel and two VT-2 qualified inspectors. All RPVH penetrations were inspected either by direct visual examination or visual examination using a mirror on a pole and flashlights. The CRDM shroud was removed and the examiners were able to have access to essentially 100% of the required examination surface. No evidence of boron deposits indicating active leakage from the annular gaps around the penetrations was observed. The licensee did identify general surface corrosion in the dome area of the RPVH and light boron stains in some CRDM penetrations, but they were not indicative of active RCS leakage. In addition, the examiners identified a small amount of boron residue at a CRDM lower canopy seal weld. The licensee performed a PT examination and found no indications of leakage through that canopy seal weld. The licensee compared the results from this BMV examination with the previous one and found no changes that would indicate pressure boundary leakage.

   The inspectors witnessed part of the BMV examination and performed an independent assessment of the RPVH condition and found no indications of leakage from the RPVH nozzles or significant corrosion of the RPVH top surface area around the penetration nozzles.

6) Evaluation of the licensee's ability to identify and characterize small boron deposits, as described in NRC Bulletin 2001-01.

As noted above, the licensee was able to have access to essentially 100% of the required examination surface. The examination procedure established requirements for the illumination and resolution of the examination equipment. Per procedure, the light intensity (minimum of 50 ft-candles) must allow the examiner to see a 0.105 inch lower case character height at a six ft distance. Based on the inspector's assessment of the BMV examination implementation, the review of personnel qualifications, the review of the BMV examination procedure, and the review of the licensee's observations captured in the examination report; the inspectors considered that the licensee had the ability to identify and characterize small boron deposits in the examination area.

7) Evaluation of the extent of material deficiencies (i.e., cracks, corrosion, etc.) that required repair.

No examples of CRDM penetration tube flaws requiring repair were identified during the NDEs and the BMV examination. As indicated above, UT indications were identified at several RPVH penetrations and they were dispositioned as fabrication indications (not service related).

The licensee did identify two rounded indications during the PT examination performed on the vent line J-groove weld area. The indications were located in the J-groove weld (.25" and .125" in diameter) and did not show characteristics of PWSCC. These indications were ground to a depth where the minimum ASME Code requirements for the J-groove depth were met. The inspectors witnessed portions of the repair process and reviewed a recorded video of the final PT performed after the indications were removed.

In addition, the licensee identified a material deficiency that required repair in one of the RPVH thermocouple penetrations (not pressure boundary related). During the volumetric examination of the RPVH penetrations, a loose guide funnel was found in one of the thermocouple penetrations. The guide funnel is threaded to the penetration tube and then fillet welded. This fillet weld does not have a pressure boundary function; it prevents the funnel from becoming loose during operation. The existing fillet weld was discovered cracked after further visual inspection. The licensee implemented repair activities by making a new fillet weld. The inspectors reviewed the work order and welding procedures related to the repair and found no issues of significance.

8) Evaluation of any significant impediments to effectively perform each examination method (e.g., centering rings, insulation, thermal sleeves, nozzle distortion, etc.)

The volumetric examination coverage extended from a minimum of 2-in above the highest point of the J-groove weld to the maximum coverage possible below the lowest point of the J-groove weld, which resulted to be more than 1-inch for all nozzles, except for thermocouple penetration No. 78.

The examination coverage for penetration No. 78 was 0.70-inch below the lowest point at the toe of the J-groove weld. The examination coverage limitation was due to the nozzle length, the weld profile on the downhill side of the nozzle, and the tapered tip of the thermocouple nozzle. At the time of the NRC inspection, the licensee was working on a request for relaxation from the NRC Order requirements.

The inspectors reviewed Dominion Engineering Calculation C-3217-00-01, which contains the axial and hoop stress analysis for Catawba's RPVH nozzles. The analysis determined the distance below the J-groove weld where the stresses reach 20 ksi tension in penetrations with a set up angle of 0, 15.8, 29.3, 43.8, and 47.0 degrees. The inspectors reviewed the coverage obtained for a sample of RPVH penetrations at different set up angles to verify that the distance below the lowest point of J-groove weld to reach 20 ksi was bounded by the examination coverage. In addition to UT coverage limitations for thermocouple penetration No. 78, no issues concerning the UT coverage below the J-groove weld for the remaining penetrations were found during the NRC inspection.

The BMV examination required the removal of the CRDM shroud and the RPVH mirror insulation to obtain the examination coverage required by the NRC Order. Some pieces of insulation could not be removed, but they were lifted as necessary to perform the examination. No issues concerning the BMV examination coverage were found during the NRC inspection.

The inspectors considered that the examination coverage requirement of the NRC Order was met for the NDE activities reviewed during the NRC inspection. The licensee did not experience any significant impediment that would preclude the effective performance of the volumetric and BMV examinations. The only exemption was the UT coverage limitation on thermocouple penetration No. 78, which was in process to be submitted to the NRC for further review.

9) Evaluation of the basis for the temperatures used in the susceptibility ranking calculation.

The inspectors reviewed the susceptibility ranking calculation and the basis for the RPVH temperatures used in the calculation. The calculation determined the RPVH Effective Degradation Years and susceptibility ranking since the first operating cycle until the current operating cycle using best estimated values of effective full power days. This calculation has been updated at the end of every operating cycle since the NRC Order was effective. The temperature used for the calculation was the reactor coolant system cold leg temperature. The use of this temperature was based on the RPV upper internals temperature documented on WCAP-13493, "Reactor Vessel Closure Head Penetration Key Parameters Comparison," and WCAP-9404, "Study of Reactor Vessel Upper Head Region Fluid Temperature."

10) Verification that the methods used for disposition of NDE identified flaws were consistent with NRC flaw evaluation guidance.

No indications considered to be penetration nozzle flaws were found during the RPVH examinations. As indicated above, UT indications were identified at several RPVH penetrations and they were dispositioned as fabrication indications (not service related). The rounded indications found in the vent line weld were removed and J-groove dimensions required by the ASME Code were maintained.

11) Evaluation of the existing procedures to identify potential boric acid leaks from pressure-retaining components above the RPVH and the licensee's followup actions for indications of boric acid leaks.

The inspectors reviewed Procedure MP/1/A/7150/042, "Reactor Vessel Head Removal and Replacement," which was implemented, in part, to conduct inspection activities required by the NRC Order to identify potential boric acid leaks from pressure-retaining components above the RPVH. This procedure has steps to inspect above and through the CRDM shroud windows for evidence of leakage every refueling outage. The licensee generated corrective action document PIP C-06-07904 to implement enhancements for this procedure, in order to clearly specify the components that are required to be examined every refueling outage. The licensee also generates a model work order every refueling outage to inspect pressure-retaining components above the head. The work order provides instruction to inspect the following components for leakage: 78 CRDM housing welds, 1-inch diameter RPVH vent nozzle to stainless steel butt weld, auxiliary head adaptors, and canopy seal welds. The examination activities discussed above are controlled by the RPVH removal/installation procedure and the model work order, except when a BMV examination is scheduled for the refueling outage. In that case, the BMV examination procedure covers the examination of the penetration nozzles, the head surface area, and the pressure retaining components above the head.

The inspectors performed an independent assessment of the RPVH condition and held discussions with licensee personnel to confirm followup actions taken for any evidence of boric acid leaks above the RPVH. The inspectors considered that the implementation of the procedures and the work order mentioned above met the requirements of the NRC Order.

.3 Independent Spent Fuel Storage Installation (ISFSI)

a. Inspection Scope

The inspectors reviewed design calculations, drawings, and specifications pertaining to construction of the ISFSI haul path road from the Unit 1 and 2 fuel handling buildings to the new ISFSI facility.

Documents reviewed were the Technical Requirements for the ISFSI Transporter Haul Road; CNS ISFSI Haul Path Evaluation Calculation, Document 32-5053646-03; the CNS ISFSI RN Bridge Micropile Specification; Engineering Change CD500624, Vehicle Crossing for Large RN Piping; and Drawing Numbers 504795E, 504796E, and 9011459E.

The inspectors walked down the haul road and examined completed improvements which were constructed to protect existing buried piping and cable trenches along the proposed ISFSI transporter haul route. A crossing structure supported by drilled micropiles will be constructed over the safety related service water lines at the one location where the ISFSI transporter will cross the piping.

The inspectors identified several issues which required clarification. The licensee documented these issues in a PIP report. The issues were as follows:

1) The need to clearly mark the prepared transporter road so that the loaded ISFSI transporter will not be inadvertently driven off the road.

2) The calculation which checked the adequacy of the 10 foot diameter component circulating water piping assumed the piping would be pressurized when the loaded ISFSI transporter passed over the piping. Controls need to be implemented (e.g. procedure step) to ensure that the piping is pressurized when transporter load is over the component circulating water piping.

3) The seismic design criteria for the bridge over the service RN piping supported by the micropiles needs to be clearly defined in the design calculation.

4) The actual thickness of the RN cooling supply piping to the diesel generators needs to be verified to confirm assumptions in the design calculation.

5) The RN piping loading calculations should consider a strip loading acting on the buried piping when the transporter is moving parallel to the piping, in addition to a point load when the transporter is passing over the RN piping.

b. Findings

The inspectors concluded that the controls for construction of the ISFSI haul road to protect buried piping were adequate. No findings of significance were identified.

4OA6 Meetings, Including Exit

On December 1, 2006, the inspectors discussed results of the onsite radiation protection inspection, Routine ISI and Boric Acid Corrosion Control Program inspection, and the TI2515/150 reactor vessel inspection with Mr. J. Morris, Site Vice President, and his staff to discuss the results of the radiation protection inspection.

On January 11, 2007, the resident inspectors presented the inspection results to Mr. B. Pitesa 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 or destroyed.

4OA7 Licensee-Identified Violations

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

10 CFR Part 50.55a(g)(4), “Codes and Standards,” requires, in part, that components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements set forth in Section XI of the editions of the ASME Boiler and Pressure Vessel Code and addenda which the facility has committed to in their ASME Section XI program. During the second and third intervals, the licensee was committed to the 1989 edition, and the 1998 edition and 2000 addenda respectively.

Contrary to the above, the licensee failed to meet section IWA-2420, “Inspection Plans and Schedules”, of the 1989 edition, and 1998 edition and 2000 addenda of the ASME Section XI code, in that they had failed to identify over 30 components which were required to be examined by their Inservice Inspection (ISI) Program. The licensee identified this violation during an operating experience review for a previous violation at another facility (05000269,270,287/2006003-02). The current violation was identified in the licensee’s corrective action program as PIP C-06-05445. As part of their corrective actions the licensee re-evaluated the scheduling of inspections of welds and supports included in their ISI Program, and has completed or scheduled examinations for the missed weld and support examinations. The finding is not suitable for SDP evaluation, but has been reviewed by NRC management and is determined to be a finding of very low safety significance because no SSCs were found to be inoperable as a result of the completed exams.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
K. Adams, Human Performance Manager
T. Alley, Duke General Office
E. Beadle, Emergency Planning Manager
S. Beagles, Chemistry Manager
E. Brewer, Operations Training Manager
M. Bryan, Civil Design Engineer
W. Byers, Security Manager
T. Cabe, Radiation Protection Supervisor- ALARA
W. Callaway, Engineering
J. Ferguson, Safety Assurance Manager
J. Foster, Radiation Protection Manager
W. Green, Reactor and Electrical Systems Manager
M. Hacker, Level III Examiner, Areva
G. Hamrick, Mechanical, Civil Engineering Manager
R. Hart, Regulatory Compliance Manager
M. Hatley, Weld Overlay Coordinator
A. Hogge, General Office - ISI Plan Manager
G. Hudson, ISI Engineer
D. Jamil, Catawba Site Vice President
B. Kimray, Radiation Protection- Sr. Scientist
A. Lindsay, Training Manager
J. McConnell, Shift Operations Manager
J. Morris, Catawba Site Vice President
C. Orr, Work Control
J. Pitesa, Station Manager
L. Reed, Modifications Engineering Manager
R. Repko, Engineering Manager
G. Spurlin, LOR Training Supervisor
G. Strickland, Regulatory Compliance
C. Trezise, Operations Superintendent
M. Webster, Manager of RPVH examination team, Areva
D. Whitaker, Duke General Office

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000413/2006005-01	NCV	Failure to perform adequate examinations of 1A ND heat exchanger inlet and outlet welds (Section 1R08).
05000413,414/2006005-02	NCV	Inadequate Recognition, Assessment and Management of the Increased Shutdown Risk Associated With the Failure of the 1B KF Pump with the Core in the Spent Fuel Pool and the 1A DG Inoperable (Section 1R13).

05000413,414/2006005-03	NCV	Failure to implement Adequate Design Control for Ice Condenser Lower Inlet Doors (Section 1R22).
05000413,414/2006005-04	NCV	Failure to Prevent Recurring Scaffolding Installation Deficiencies (Section 4OA2.2).

Closed

05000413/2006001	LER	Loss of Offsite Power Event Resulted in Reactor Trip of Both Catawba Units from 100% Power (Section 4OA3.4).
05000413,414/2515/169	TI	Mitigating Systems Performance Index Verification (Section 4OA5.1).
05000413/2515/150-2	TI	Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Order EA-03-009) (Section 4OA5.2).

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection
Nuclear System Directive 317, Freeze Protection Program, Rev. 03
Freeze Protection Readiness Preparation for August 2006 / Winter 2007 Season
PT/0/B/4700/038; Cold Weather Protection; Rev. 025
PT/0/B/4700/039; Hot Weather Protection; Rev. 009
PT/0/B/4350/008; Heat Tracing Alignment Verification; Rev. 038
IP/0/B/3560/008; Preventive Maintenance and Operational Check of Freeze Protection Heat Trace and Instrument Box Heaters; Rev. 47
IP/0/B/3560/009; Operational Check for Winter Months and Extreme Cold Weather Surveillance of Freeze Protection Heat Trace and Instrument Box Heaters (EHT/EIB) Systems; Rev. 010
Nuclear System Directive 317, Freeze Protection Program, Rev. 03 IP/0/B/3560/009; Operational Check for Winter Months and Extreme Cold Weather Surveillance of Freeze Protection Heat Trace and Instrument Box Heaters (EHT/EIB) Systems; Rev. 010
PIP C-05-1316; Deviations identified during the 2005 Catawba Freeze Protection Assessment conducted by the General Office
PIP C-05-1318; Areas of improvement identified by the Freeze Protection Assessment Team
PIP C-05-6797; Current items for weather related surveillance activities for Operations
Operator Aid Computer Alarm Response Information for points C1P0118 (Dry Bulb Ambient Temperature - Unit 1)
Operator Aid Computer Alarm Response Information for points C1P1821 (Wet Bulb Ambient Temperature - Unit 1)
Operator Aid Computer Alarm Response Information for points C2P0118 (Dry Bulb Ambient Temperature - Unit 2)
Operator Aid Computer Alarm Response Information for points C2P1821 (Wet Bulb Ambient Temperature - Unit 2)

Section 1R04: Equipment Alignment
Risk Management Actions for the 1B KF Pump Being out of Service
Risk Management Actions for Both Unit 1 DGs Being Inoperable

Section 1R05: Fire Protection
Pre-Fire Plan for Fire Strategy Area 17; Unit 1 Cable Room, Auxiliary Building 574 level, Rooms 491 and 491A
Pre-Fire Plan for Fire Strategy Area 3; Unit 1 CA Pump Room, Auxiliary Building 543 level, Room 250
Pre-Fire Plan for Fire Strategy Area 25; 1A Diesel Generator Room, Diesel Generator Building, Room 1A
Pre-Fire Plan for Fire Strategy Area 10; Unit 1 Battery Rooms, Auxiliary Building 554 level, Rooms 350 - 356
Pre-Fire Plan for Fire Strategy Area 15; Unit 1 Essential Switchgear Room, Auxiliary Building 577 level, Rooms 495 & 496
Pre-Fire Plan for Fire Strategy Area 7; Unit 2 Essential Switchgear Room, Auxiliary Building 560 level, Rooms 362 & 363
Pre-Fire Plan for Fire Strategy Area 47; Unit 2 Spent Fuel Pool; Purge Unit, Auxiliary Building 636 level, Room 802
Pre-Fire Plan for Fire Strategy Area 23; Unit 2 Spent Fuel Pool; Auxiliary Building 605 level, Room 614
Pre-Fire Plan for Fire Strategy Area

Section 1R06: Flood Protection
CN-938-15; Electrical Equipment Layout Outdoor Area Buried Cable and Conduit Plan, Revision 1
CN-1938-01; Electrical Equipment Layout Outdoor Area General Plan, Revision 67
CN-1390-07; Miscellaneous Yard Structures Cable Trenches Concrete and Reinforcing Plan, Sections and Details, Revision 29
CN-1390-06; Miscellaneous Yard Structures Cable Trenches Layouts, Revision 26
CN-1390-24; Miscellaneous Yard Structures Cable Trench Concrete and Reinforcing Plan, Sections and Details, Revision 1
CN-1022-01; Powerhouse Yard Area Grading Plan, Revision 74A
CN-1022-01; Powerhouse Yard Area Grading Plan, Revision 74
CN-1022-01; Powerhouse Yard Area Grading Plan, Revision 73A
CN-1022-01; Powerhouse Yard Area Grading Plan, Revision 72A
CN-1022-02; Construction Yard Area Grading Plan, Revision 14
CN-1022-03; Cooling Tower Yard Area Grading Plan, Revision 26
CN-1022-02; Standby Nuclear Service Water Dam Area Grading Plan, Revision 23
CN-1022-06; Grading Sections and Details Sheet 1, Revision 23
CN-1022-07; Grading Sections and Details Sheet 2, Revision 11
CN-1022-08; Waste Water Treatment System Grading Plan, Sections and Details, Revision 26
CN-1022-09; Grading Sections and Details Sheet 3, Revision 4
CN-1022-11; Earthwork Site Backfill Requirements Layout and Notes, Revision 11
CN-1022-12; Solid Waste Landfill No. 2 Plan and Sections, Revision 8
CN-1022-13; Flood Protection Requirements for Local Intense Probable Maximum Precipitation, Revision 0

CN-1022-17; Powerhouse Yard Drainage Layout, Revision 6
CN-1022-17; Powerhouse Yard Drainage Layout, Revision 6A
CN-1022-18; Cooling Tower Yard Area Construction Yard Area Waste Water Treatment System Area Drainage Layout, Revision 0
CN-1209-11.08; Flood Boundary Wall and Floor Locations Architectural Plan at EL 537+0, 543+0 and 550+0, Revision 2
CN-1209-11.09; Flood Boundary Wall and Floor Locations Architectural Plan at EL 560+0, 554+0 and 568+0, Revision 1
CN-1209-11.10; Flood Boundary Wall and Floor Locations Architectural Plan at EL 574+0 and 577+0, Revision 1
CN-1209-11.11; Flood Boundary Wall and Floor Locations Architectural Plan at EL 594+0, Revision 2
CN-1209-11.12; Flood Boundary Wall and Floor Locations Architectural Plan at EL 606+10, 608+0, 609+0, 610+0 and 611+0, Revision 1
CN-1680-109; Flooding and Pressure Seal Installation, Revision 8
CN-1261-2.4; Standby Shutdown Facility Embedded Pipe Plan, Sections and Details, Rev. 1
CN-1560-2.0; Flow Diagram of Standby Shutdown Diesel System (AD), Revision 2
CN-1583-2.0; Flow Diagram of Conventional Waste Water (WC), Revision 15
CN-2565-2.2; Flow Diagram of Liquid Radwaste System (WL), Revision 30
CN-1565-2.2; Flow Diagram of Liquid Radwaste System (WL), Revision 34
CN-1565-1.4; Flow Diagram of Liquid Radwaste System (WL), Revision 30
CN-1565-1.1; Flow Diagram of Liquid Radwaste System (WL), Revision 29
CN-1609-7.0; Flow Diagram of Diesel Generator Room Sump Pump System (WN), Revision 10
CN-2609-7.0; Flow Diagram of Diesel Generator Room Sump Pump System (WN), Revision 8
CN-1604-2.1; Flow Diagram of Service Building Sump Pump System (WB), Revision 10
CNC-1114.00-00-0040; Yard Drainage Results of PMP, Revision 22
CNC-1206.03-00-0001; Flood Levels for Structures Outside of the Reactor Building, Revision 19
CNS-1435.00-00-0003; Design Specification for Mechanical and Electrical Penetration Fire, Flood, and Pressure Seals, Revision 3
PIP C-06-4447; Site topography has changed since original design such that the surface water drainage system as described in UFSAR 2.4 needs to be surveyed and evaluated
PIP C-00-2859; CSRG requested to perform assessment of the roofing modification process as it pertains to the Unit 2 water intrusion reactor trip
PIP C-06-7300; Error in CA pump room flooding calculation

Section 1R08: Inservice Inspection Activities
Engineering Support Document, Boric Acid Control Program, Rev. 3
QAP 9.6, Welding Services Incorporated Liquid Penetrant Inspection Procedure, Rev. 10
SI-UT-126, Structural Integrity Phased Array Ultrasonic Testing Procedure, Rev. 0
PT/1/A/4150/001 H, Inside Containment Boric Acid Check, Rev. 13
WSI WPS 03-08-T-801-102840, Welding Procedure Specification for weld overlay
WSI PQR-03-08-T-801, Procedure Qualification Record for weld overlay
NSD 203, Operability, Rev. 18
PIP C-06-08118, Items identified as a result of boric acid walkdown, 11/29/2006*
PIP C-06-03049, Dried boron on RCPs 2A and 2B mechanical seal flanges, 4/17/2006
PIP C-05-07338, 1FW-001A active boron leak at packing gland, 12/07/2005
PIP C-05-06604, 2NV-232 “active” boron required engineering evaluation, 10/31/2005

PIP C-06-07442, Mode 3 boric acid walkdown, 11/11/2006
PIP C-06-02612, ER308 weld filler material used on SA312 and SA182 TP36 piping material, 4/5/2006
PIP C-06-08086, Wrong filler material used during welding process, 11/29/2006
PIP C-06-02414, The Service Water Project has identified an emerging trend related to materials problems, 3/30/2006
PIP C-06-01627, Piping joints welded with incorrect filler material, 3/7/2006
PIP C-06-00411, Two inch piping welded with incorrect filler material, 1/17/2006
PIP C-06-05445, Missed ISI examinations during the 2nd interval, 7/26/2006
PIP G-06-00256, Missed ISI examinations (General Office PIP), 6/21/2006
PIP C-06-04726, NSAL-06-8 Pressurizer heater sleeve cracking, 6/21/2006
PIP C-05-06872, Incorrect supports installed on 1RNPT9520, 11/10/2005
PIP C-06-04368, ASME Section XI relief request, 6/6/2006
PIP C-06-08054, Pre heat hold point not documented on welding process control, 11/28/2006
PIP C-05-04844, EPRI MRP-139 program requirements
Letter from Duke Energy Corp to USNRC, SUBJ: REQUEST FOR RELIEF 05-CN-001, dated February 17,2005
Letter from Duke Energy Corp to USNRC, SUBJ: REQUEST FOR RELIEF 05-CN-001 REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION, dated November 28, 2005
Catawba Nuclear Station UFSAR, Sections 5.4 and 6.3
Weld Data Sheet, NW-4A, Pressurizer Safety Line weld overlay
WSI Traveler No 103441-003, NW-4A weld overlay, Rev. 0
Root Cause Failure Analysis Report, Missed Class B and C ISI Examinations Third 10-year ISI Interval, Rev. 0

Section 1R11: Licensed Operator Requalification
LOR Task Requirement Guide OP-CN-LOR-S-54; Loss of ND (AP/19 Case I, Case III and Case IV); Rev. 01

Section 1R12: Maintenance Effectiveness
PT/2/A/4200/010A; Residual Heat Removal Pump 2A Performance Test; Rev. 46
PT/1/A/4350/006A; 4160 Essential Power System Train A Test; Rev. 10
PT/1/A/4350/002A; DG 1A Operability Test, Rev. 112
PT/1/A/4350/002C; Available Power Source Operability Check, Rev. 23
PIP C-00-4947; Potential problem with operation of the DG tied to the grid with the non-emergency trips bypassed
PIP C-06-8330; Cut marks in pipe wall downstream of letdown orifice isolation valve 1 NV-11A
WO 01717483 01; 1NV 011A; I/R Valve Leaking By Seat
Radiography Film for base metal repairs of American Society of Mechanical Engineers Code pressure boundary piping near 1 NV-11A
PT/2/A/4450/005B; Containment Air Return Fan 2B and Hydrogen Skimmer Fan 2B Performance test, Rev. 37
PIP C-06-8039; Unexpected continuation of 72-hour TSAIL entry due to Containment Air Return System dampers 8, 9 & 10 not opening during testing

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Complex Maintenance Plan WO 0171534-01; Unit 2 Digital Turbine Control Work on Communication Cards; Execution Date: 10/26/06
PIP C–06-8302; Documentation of a major schedule change associated with the start of Engineered Safeguards Features testing following the completion of repairs on the 1A DG
PIP C-06-8273; Questions regarding the protection of the B train equipment with the 1A DG inoperable due to a bearing failure
NSD 213; Risk Management Process, Rev. 06
NSD 403; Shutdown Risk Management (Modes 4, 5, 6 and No Mode) per 10CFR50.65(a)(4); Rev. 15 and Rev 16
NSD 417; Nuclear Facilities / Generation Status Communications; Rev. 06
OMP 2-18; Equipment Protection and Quarantine, Rev. 066
PIP C-06-7840; Unplanned Red DID status on Spent Fuel Pool Cooling due to elevated 1B KF pump temperatures
PIP C-06-7829; KF pump 1B inboard bearing failure
CNC-1201-30-00.0045; Appendix C; Catawba Unit 1 EOC16 Time to Boil in the Spent Fuel Pool
Catawba Unit 1 Unified Operational Logs for the period of November 21 - 23, 2006
Risk Management Actions Plans for the 1B KF pump being out of service

Section 1R15: Operability Evaluations

PIP C-04-4871; DG 2B would not go into Maintenance Mode until the 3rd attempt
PIP C-00-4947; Potential procedure issues related to non-emergency trips of a DG tied to the grid during testing
PIP C-99-4765; 1B DG tripped on overcurrent while loading during an operability test
Selected Licensee Commitment 16.8-5; Diesel Generator Supplemental Testing Requirements
WO 01720158 01; Containment Coatings Inspection to support PIP C-06-7708 PIP C-06-8288; Torque wrench used below allowable QA range
PIP C-06-8292; Torque wrench used below allowable QA range
CNC-1223.23-00-0033; Supporting Calculation for Response to NRC IE Bulletin 88-04, "Potential Safety Related Pump Loss" (Component Cooling Pumps); Rev. 12
CD101217; Minor Design Change to replace 30 amp fuses with 40 amp fuses in the power supply to Unit 1 H2 Igniters
IP/1/A/3170/001; H2 Igniter Surveillance Test date performed on 12/17/06
IP/2/A/3170/001; H2 Igniter Surveillance Test date performed on 12/26/06
PIP C-06-8562; Unexpected entry into Tech Specs due to the 2A H2 Igniter being declared inoperable
Tech Spec 3.6.9; Hydrogen Ignition System
PIP C-06-8288; Torque wrench used below allowable QA range
CNC-1223.23-00-0033; Supporting Calculation for Response to NRC IE Bulletin 88-04, "Potential Safety Related Pump Loss" (component cooling pumps)

Section 1R17: Permanent Plant Modifications

PIP C-06-08355; Redundant solenoid valves added to MSIVs not functioning as designed
PIP C-06-08577; MSIV 'A' train stroke test
PIP C-06-8555; Results of investigation into the failure of main steam isolation valves 1SM3 and 1SM5 to pass the post modification testing of CN-114411
TT/1/A/9300/052; Post Installation Test of Modification CN-11441/00; Rev. 0

Section 1R19: Post-Maintenance Testing
PT/1/A/4350/006A; 4160 Essential Power System Train A Test; Rev. 10
PT/1/A/4350/002A; DG 1A Operability Test, Rev. 112
PT/1/A/4350/002C; Available Power Source Operability Check, Rev. 23
PIP C-00-4947; Potential procedure issues related to non-emergency trips of a DG tied to the grid during testing
PIP C-06-08161; Shaft vibration on end of 1A NV pump is higher than value prior to 1EOC16 maintenance
PIP C-06-0628; Inspection of the 1B emergency diesel generator during 1EOC16
PT/1/A/4400/003F; Head curve test for KC pumps 1A1, 1A2, 1B1, 1B2, Rev. 14

Section 1R20: Refueling and Outage Activities
1EOC-16-IRT Unit 1 Outage Risk Assessment
Site Directive 3.1.30, Unit Shutdown Configuration Control (Modes 4, 5, 6 or No Mode), Rev. 34
Nuclear System Directive, NSD-403, Shutdown Risk Management (Modes 4, 5, 6 or No Mode), per 10CFR50.65(a)(4); Rev. 15 and 16\
OP/1/A/6150/006, Draining The Reactor Coolant System, Rev. 69
PT/1/A/4350/003, Electrical Power Source Alignment Verification, Rev. 45
OP/1/A/6200/005, Spent Fuel Cooling System, Rev. 74
PT/0/A/4150/037, Fuel / Component Movement Accounting, Rev. 9
PT/1/A/4200/002C, Containment Closure Verification (Part I); Rev. 75
PT/1/A/4200/002I, Containment Closure Verification (Part II); Rev. 33
PT/1/A/4200/002J, Containment Closure Verification Penetration Status Change; Rev. 10
OP/0/A/6100/014, Penetration Control for Modes 5 and 6; Rev.31
OP/1/A/6150/001, Filling and Venting the Reactor Coolant System, Enclosure 4.16, Reactor Coolant System Vacuum Refill Without Solid Operation; Rev. 95
OP/1/A/6150/006, Draining the Reactor Coolant System; Rev.68
Enclosure 4.2, Decreasing the NC System Level
Enclosure 4.3, Increasing the NC System Level
Enclosure 4.10, Requirements for Operation with the NC System Level Below 16%
Enclosure 4.12; Reduced Inventory Posting Requirements
OP/0/A/6550/015; Receipt, Inspection and Storage of New Fuel, Rev. 30
PT/0/A/4150/29A; New Fuel and Component Inspection, Rev. 2
PT/1/A/4550/001F, Preparation for New Fuel Receipt, Rev. 3
OP/1/A/6550/006, Transferring Fuel with the Spent Fuel Manipulator Crane; Rev. 58
OP/1/A/6550/007, Reactor Building Manipulator Crane Operation; Rev. 32
OP/1/A/6550/008, Fuel Transfer System Operation; Rev. 9
MP/0/B/7150/012, Refueling Canal Cleanliness; Rev. 7
PT/1/A/4550/001C, Refueling Communications Test; Rev. 7
PT/1/A/4550/001D; Reactor Building Manipulator Crane Load test; Rev. 17
PT/1/A/4550/001E; Spent Fuel Building Manipulator Crane Load test; Rev. 11
PT/0/A/4550/003C, Core Verification; Rev. 9
PT/0/A/4150/022, Total Core Reloading; Rev. 39
Unit 1 1EOC16 Core Reload Verification videotape
PT/0/A/4200/002, Containment Cleanliness Inspection; Rev.26
SM/0/A/8510/008, Ice Condenser FME Inspection; Rev. 3
PT/0/A/4150/019; 1/M Approach to Criticality; Rev.33

PT/0/A/4150/001J, Zero Power Physics Testing; Rev. 1
PT/0/A/4150/001, Controlling Procedure for Startup Physics Testing; Rev. 40
OP/1/A/6100/001, Controlling Procedure for Unit Startup; Rev. 212
OP/1/A/6100/003, Controlling Procedure for Unit Operations; Rev. 97
OP/1/B/6300/001, Turbine Generator Startup; Rev.83
OP-CN-JITT-ZPPT/Turbine; Just In Time Training Package; Initial Startup / Zero Power Physics Testing / Turbine On-Line; Rev. 7
OP/1/A/6100/002; Controlling Procedure for Unit Shutdown; Rev. 159
SM/0/A/8510/008; Ice Condenser FME Inspection; Rev. 3
MP/1/A/7150/042; RX Vessel Head Removal & Replacement, Rev. 48
Catawba Unit 1 Spent Fuel Pool Assembly Location Map, Cycle 16, Rev. 82
CNEI-0400-28, Catawba 1 Cycle 17 Final Core Map, December 2006 based on CNC-1553.05-00-0450, Rev. 2
PIP C-06-7212; Based on a letter from Westinghouse, the upper limit for the lower inlet door 40-degree acceptance criteria is being changed
PIP C-06-07433; During Shutdown for 1EOC16, NC Boron (as sampled at the NV Mixed Bed Demin inlet) did not trend as expected.
PIP C-06-07438; A significant area of unit 1 pipe chase, at approximately 260 degrees and between elevation 552 feet and 591 feet, is covered with a white powder substance (boron).
PIP C-06-07440; An area inside unit 1 pipe chase was found covered with a white powder substance, which appears to be boric acid. Engineering to determine source of leak and to evaluate need for additional corrective action.
PIP C–06-7741; Results of additional Unit 1 ECCS sump inspection following removal of the impingement plates
PIP C-06-07442; This PIP is written to document findings while performing Unit1 Mode 3 Inside Containment Boric Acid Check for 1EOC16.
PIP C-06-07445; Ice Condenser initial 1EOC16 Mode 5 walkdown
PIP C-06-07448; Critique from C shift for Power decrease from 94% to Mode 3 at 500 degrees along with JITT to support plant activities
PIP C-06-07462; Discrepancy between Actual DID Sheet condition and Planned DID Sheet condition for Mode 5 HDH with Equipment Hatch Open. There was no documented review of the cause of the discrepancy or the impact on outage risk.
PIP C-06-07474; Documentation of Equipment Hatch Emergency Closure Drill.
PIP C-06-07484; All prerequisites were not completed prior to addition of hydrogen peroxide to NC system.
PIP C-06-07506; PM (NDE inspection) of the Spent Fuel Manipulator Crane Gripper was not performed as required in parallel with the Fuel Transfer System PM. It shows up in the schedule a day after the Transfer System PM was completed.
PIP C-06-7455; Engineering & Maintenance walkdown of the Unit 1 Ice Condenser
PIP C-06-8783; Miscellaneous FME items found during the USSI/OPS NOT/NOP containment walkdown

Section 1R22: Surveillance Testing
Tailgate Briefing Package for PT/0/A/4150/030; RCCA Bank Repositioning
Work Order 01125002; Perform ice condenser lower door tests

Section 1R23: Temporary Plant Modifications
PIP C-06-8754; Indication error found during cable connections for DRPI
PIP C-06-8763; Temporary Design Change CD101222 authorized installation of temporary cable for use with Shutdown Bank C, E-3 position
PIP C-06-8782; Acoustic monitor for 1NC1 must be swapped from primary monitor to secondary monitor due to failed parts
PIP C-06-8765; Work in area of 1NC1, 1NC2, and 1NC3 damaged cables for acoustic monitor system

Section 2OS1: Access Control To Radiologically Significant Areas

Procedures, Guidance Documents, and Manuals
Health Physics Procedure (HP)/0/B/1000/058, Diving Operations, Rev. 2
Radiation Protection Administrative Procedure (RA)/0/1100/001, Radiation Protection Routines, Rev. 13
SH/0/B/2000/012, Access Controls for High, Extra High, and Very High Radiation Areas, Rev. 7
SH/0/B/2000/0005, Posting of Radiation Control Zones, Rev. 4
SH/0/B/2000/003, Preparation of a Radiation Work Permit, Rev. 6
SH/0/B/2000/006, Control of Radioactive Material and Use of Radioactive Material Tags, Rev.4
SH/0/B/2001/001, Internal Dose Assessment, Rev. 2
SH/0/B/2000/009, Neutron Dose Tracking, Rev.2
SH/0/B/2001/002, Investigation of Unusual Dosimetry Occurrence or Possible Overexposure, Rev. 5
SH/0/B/2002/001, Multiple Dosimetry, Rev. 5
SH/0/B/2000/007, Placement of Personnel Dosimetry for Non-Uniform Radiation Fields, Rev. 1
Nuclear Policy Manual, Nuclear System Directive: 501, Temporary Storage of Radioactive Material in the Spent Fuel Pool
Nuclear Policy Manual, Nuclear System Directive: 507, RP Manager=s Best
Radiation Protection Management Procedure 2.4, EHRA and VHRA Documentation and Locking Hardware Control Guidelines.
Radiation Protection Management Procedure 6.1, Passive Monitoring-Implementation Process, Rev. 0
Standard Radiation Protection Management Procedures for Oconee, McGuire and Catawba Nuclear Stations, (SRPMP) 2-1, ED Alarms, Rev. 0

Records and Data Reviewed
RWP 4, Receipt / Shipment of Miscellaneous Radioactive Material (excluding radioactive waste and spent fuel)
RWP 11, Routine Spent Fuel Pool Area Activities (Excluding Refueling)
RWP 1413, Vessel Flange Cleaning and Inspection
RWP 1414, Canal and Cavity Decontamination
RWP 1417, Fuel Transfer System and Blind Flange Work.

Corrective Action Program (CAP) Documents
PIP C-06-02775, Airborne radioactivity during vessel flange cleaning resulted in significant plateout on the operating deck restricting access.
PIP C-06-06848, Forms identifying non-fuel items in spent fuel pool were not completed correctly.

PIP C-06-06885, Door to room 306 found unlocked.
PIP C-06-07548, Low risk work orders have been issued which direct workers to bypass RP. Legacy work order templates had carried an old practice forward.
Email referring to how the problem in PIP C-06-07548 was immediately corrected.
PIP C-06-07682, Several contract workers will require administrative dose limit extensions prior to starting work.
PIP C-06-07990, Radiation Protection Human Performance Assessment -3rd quarter 2006

Section 2OS2: ALARA Planning and Controls

General Documents
Catawba Nuclear Station ALARA REPORT, dated 02/03/05, 03/07/05, 04/07/05, 05/08/05, 06/03/05, 07/11/05, 08/03/05, 09/01/05, 10/05/05, 11/02/05, 12/27/05, 01/05/06, 02/01/06, 03/07/06, 04/01/06, 05/03/06, 06/05/06, 07/06/06, 08/03/06, 09/05/06, 10/03/06, and 11/03/06.
ALARA Committee Minutes, dated 02/23/06, 06/28/06, 09/13/06, and 10/18/06.
ALARA Committee Meeting Agenda, dated 02/23/06, 06/28/06, 09/13/06, and 10/18/06.
1EOC15 Mass Shielding Installation and Removal, ALARA Planning Worksheets (Complete ALARA Package for RWP=s 1125, 1166, 1442, and 1623; including post job review), multiple dates.
1EOC16 In Service Inspection during Refueling Outage, ALARA Planning Worksheet (Complete ALARA Package for RWP's 1130, 1426, and 1616), dated 10/27/06
1EOC16 In Service Inspection of Head, ALARA Planning Worksheet for RWP's 1453, dated 10/27/06
1EOC16 In Service Inspection of Reactor Vessel Threads & Nozzle Belt, ALARA Planning Worksheet for RWP's 1462, dated 10/27/06
1EOC16 ECCS Sump Modification, ALARA Planning Worksheet (Complete Package) for RWP's 1178, dated 10/27/06
1EOC16 Unit 1 PZR Alloy 600 Weld Overlay Modifications during Refueling Outage, ALARA Planning Worksheet (Complete ALARA Package for RWP's 1171, and 1467), dated 10/27/06
1EOC16 Shutdown Sequence / Crudburst Contingency Plan, not dated.
Crud burst Teem Meeting Agenda, dated 10/18/06
Crud burst Teem Meeting Minutes, dated October 2006
C-06-04560 CNS to review and evaluate PIP C06-02305 MNS Source Term Reduction Assessment
Crud Burst Team Action Items, dated March 14, 2005
Zinc Injection Action Register (not dated)
RPS-13-06 ALARA Planning and Controls Assessment
Catawba Nuclear Station DOSE PLANNING REPORT (Estimate 2006) (for groups 108, 110, 111, and 105), dated 11/26/2006, 11/27/2006, 11/27/2006, and 11/29/2006 respectively
Collective Daily Job Dose History for ECCS Sump Mod, dated 11/27/2006
Collective Daily Job Dose History for PZR weld overlay, dated 11/27/2006
High Radiation Work Area Pre-Job Brief (ALLOY 600 WELD OVERLAY), dated 11/14/2006

Problem Investigation Process Documents
C-06-07524, Skin Contamination
C-06-07527, C-06-07504, C-06-07455, C-06-07436, Clothing Contaminations
C-06-05448, RP Audit Findings

C-04-00731, Air Actuator Valve work during 1EOC14 exceeded it's exposure estimate by >25%
C-05-05672, Estimate for continuous header vent mod exceeded estimate by > 25%

Section 2PS2: Transportation of radioactive material

Shipping Records
05-27, 8-120 HIC of RBT Resin
06-04, 14-215 Cask of Dry Active Waste
06-17, Two boxes, one Type A package and the other a surface contaminated object.
06-18, HIC containing filters (individually characterized)
06-27, Nine boxes of green is clean trash
06-122, Unit 1 pressurizer power operated relief valves

Corrective Action Documents
PIP-C-06-07350, Concern about who requires the DOT HAZMAT Training in 49CFR172.700-704.
Self Assessment RPS-17-06, Test of Emergency Responder INFOTRAC, PIP C-06-05545, 7/25/06
Self Assessment RPS-15-06, NRC Prep Audit of Radioactive Material Processing and Transportation Using NRC Inspection Plan 71122.02, PIP C-06-06293, 8/14 - 17/ 2006
Self Assessment RPS-03-06, RMC INFOTRAC Process Testing, PIP C-06-00577, 1/23/06

4OA1: Performance Indicator Verification

Records
Standard Radiation Protection Management Procedure SRPMP 10.1, NRC Performance Indicator Data Collection Validation, Review and Approval, Rev 1.
One procedure with numerous attachments for each month from January 2006 to November 2006
Catawba Nuclear Station Units 1 and 2, 2004 Annual Radioactive Effluent Release Report
Catawba Nuclear Station Units 1 and 2, 2005 Annual Radioactive Effluent Release Report
PT/2/A/4150/001D; NC System Leakage Calculation, Rev. 59
Standard Radiation Protection Management Procedure SRPMP 10.1, NRC Performance Indicator Data Collection Validation, Review and Approval, Rev 1.
One procedure with numerous attachments for each month from January 2006 to November 2006
Catawba Nuclear Station Units 1 and 2, 2004 Annual Radioactive Effluent Release Report
Catawba Nuclear Station Units 1 and 2, 2005 Annual Radioactive Effluent Release Report

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

PIP C-05-04201; NRC Resident Inspector on 7/7/2005 identified scaffold that was past its expiration date and checked as seismically erected without the required information checked on the back of the form.
PIP C-05-04384; Scaffold erected in Unit 1CA pump room tied off with rope.
PIP C-05-05379; There are numerous scaffolds present in the Auxiliary Building that are not in use and need to be removed as they present a safety impact on safe operation of the plant. They also show that there are some break downs in the processes controlling scaffolds.

PIP C-05-05653; Scaffold not appropriately secured.
PIP C-05-06878; Scaffold tag did not represent scaffold as built. Seismic section of scaffold tag incorrectly completed as meeting all requirements on seismic checklist.
PIP C-05-07052; Seismic section of scaffold tag at RN not sufficiently completed One of the scaffolds at the RN has a handrail that is making contact with a one inch RN pipe.
PIP C-05-07648; Scaffold tied off to the isolated phase bus. Tie offs to electrical equipment is not permitted under the Scaffold Manual or CSWP 3.4.
PIP C-05-07649; Scaffolding tied off to Isolated Phase Bus.
PIP C-06-01988; Scaffold does not have appropriate scaffold tags attached
PIP C-06-02701; Scaffold erected outside Ice Condenser Bay 2 Lower Inlet Doors is tied off with wire to the spring housings.
PIP C-06-03092; Maintenance "Lessons Learned" Review of OE22344-Station Wide Inspection of Scaffold Installations at Susquehanna
PIP C-06-06018; This PIP is to document issues related to erected scaffolds for tracking and trending by the Maintenance Civil section. This PIP will be used to evaluate possible trends with scaffolds being modified, moved, etc. following the initial inspections performed and to trend behaviors on the end users with regards to completed scaffolds.
PIP C-06-06235; During a routine scaffold inspection, scaffold on 522 room 113 was founded to be tied incorrectly against a narrow electrical conduit.
PIP C-06-07975; Scaffold at 1NI78 was blocking free operation of hand wheel. It was identified and corrected.
PIP C-06-08183; NRC resident identified a scaffold erected for 1KC-82 in room 300 of aux bldg which did not meet two inch seismic requirement.
PIP C-06-7006; Several examples of the control room logs not being maintained in compliance with OMP 2-17 have been noted.
Control Room Logs; Unit 1 and Unit 2
NSD 506; Operator Workarounds, Rev. 4
Catawba Operator Workaround List

Section 4OA3: Event Follow-up

Risk Management Actions for both Unit 1 DGs Inoperable; 1EOC16 Materials Engineering & Lab Services Report for CNS 2A DG Lube Oil Particles
Material Issue Ticket 0103388; DG Bearings
Material Receiving Inspection Report of DG Bearings, 7/2/96
PIP C-06-07946; 1A DG tripped on High Vibration and Abnormal Noises, 11/24/06
PIP C-06-8135; Allow use of Undersized Bearings on Number 4 Connecting Rod Journal in 1A DG
PIP C-06-8368; Bearing failure investigation of 1A DG determined suspect bearing also installed in 1B DG
PIP C-06-8554; 1A DG field voltage erratic during break-in runs
PIP C-06-8563; Voltage regulator swings and abnormal noise observed during 1A DG run
Catawba Technical Specification T.S. 3.6.13; Ice Condenser Doors
PIP C-06-3250; NRC questioned 1EOC15 As-Left 40-degree lower inlet door test results
Duke Power Calculation DPC-1201.17-00-006; Design and Licensing Basis for Ice Condenser Lower Inlet Door Technical Specification Surveillance Requirements, 40-degree Opening, Closing and Frictional Torques

Westinghouse Calculation DPC-06-81; Ice Condenser Design Basis and Safety Function for Catawba and McGuire
Duke Power Company UFSAR Change Package 07-015; Update UFSAR sections 6.7.8.3 and 6.7.20 pertaining to the surveillance testing conducted on the ice condenser lower inlet doors

Section 4OA5: Other Activities

Reactor Oversight Program MSPI Basis Document, Catawba Nuclear Station, Revision 1
MSPI Derivation Reports for systems providing input to the MSPI calculations; Unit 1 and Unit 2; dated 10/30/06
PT/1/A/4200/009A; Auxiliary Safeguards Test Cabinet Periodic Test, Enclosures 13.25 and 13.26; Rev. 176
PT/2/A/4200/013G; NI Valve Inservice Test; Rev. 34
PT/2/A/4200/013I; NV Valve Inservice Test; Rev. 26
PT/2/A/4200/021, KC Valve Inservice Test, Rev. 46
PT/1/A/4200/020; FW Valve Inservice Test; Rev. 21
ND System Health Report; 2006T2
NI System Health Report; 2006T2
NV System Health Report; 2006T2
RN System Health Report; 2006T2
EPC System Health Report; 2006T2
KC System Health Report; 2006T2
Selected Control Room Logs, January 2005 through September 2006
Selected Tech Spec Action Item Log entries, January 2005 through September 2006
PIP C-06-4759; Estimated demand / run hours and basis for the RN system revised to account for summer system alignment for MSPI
NEI 99-02; Appendix F, Rev. 04
IP 60853, On-Site Fabrication of Components and Construction of An ISFSI
CNS ISFSI RN Bridge Micropile Specification, Document 51-9012384-000
CNS ISFSI Haul Path Evaluation Calculation, Document 32-5053646-03
Engineering Change No. CD500624, Vehicle Crossing Structure for Large RN Piping
FRAMATONE Drawing No. 5047955E, Catawba Nuclear Station ISFSI Project Haul Path and Upgrades, Plan, Sections, & Details, Rev. A
FRAMATONE Drawing No. 5047956E, Catawba Nuclear Station ISFSI Project Haul Path and Upgrades, Plan, Sections, & Details, Rev. A
FRAMATONE Drawing No. 9011459E, Catawba Nuclear Station ISFSI Project Haul Path RN Bridge, Rev. A
Technical Requirements for Procurement CNR-1140.04-00-0001, Rev. 1, CD-500920: ISFSI Transporter Haul Road
PIP C-06-06857, RN ISFSI Lessons Learned
PIP C-06-06921, Clarifications Required for ISFSI Haul Road (NRC Identified)
54-ISI-30-04, “Written Practice for the Qualification and Certification of NDE Personnel,” Rev. 4
54-PT-200-06, “Color Contrast Solvent Removable Liquid Penetrant Examination of Components,” Rev. 6
54-ISI-604-001, “Automated Ultrasonic Examination of Open Tube RPV Closure Head Penetrations,” Rev.1

54-ISI-603-002, “Automated Ultrasonic Examination of RPV Closure Head Penetrations Containing Thermal Sleeves,” Rev. 2
54-ISI-605-001, “Automated Ultrasonic Examination of RPV Closure Head Small Bore Penetrations,” Rev. 1
51-9026779-001, “RPV Head Penetration Inspection Plan and Coverage Assessment for Catawba Unit 1 and McGuire Unit 2," Rev. 1
MP/1/A/7150/042, “Reactor Vessel Head Removal and Replacement,” Rev. 48
MP/0/A/7150/042D, “Reactor Vessel Head Penetration Visual Inspection,” Rev. 3