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

July 28, 2005

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

SUBJECT: CATAWBA NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000413/2005003 AND 05000414/2005003 AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION INSPECTION REPORT 07200045/2005001

Dear Mr. Jamil:

On June 30, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Catawba Nuclear Station. The enclosed integrated inspection report documents the inspection findings which were discussed on July 7, 2005, with members of your staff.

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

This report documents two NRC-identified findings of very low safety significance (Green) which were determined to be violations of NRC requirements. However, because of their very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C., 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C., 20555-0001; and the NRC Resident Inspector at the Catawba Nuclear Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system

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

Sincerely,

/RA/

Michael E. Ernstes, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket Nos.: 50-413, 50-414, 72-45
License Nos.: NPF-35, NPF-52

Enclosure: Integrated Inspection Report 05000413/2005003, 05000414/2005003, and 07200045/2005001 w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-413, 50-414, 72-45
License Nos:	NPF-35, NPF-52
Report No:	05000413/2005003, 05000414/2005003, 0720045/2005001
Licensee:	Duke Energy Corporation
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	4800 Concord Road York, SC 29745
Dates:	April 1, 2005 - June 30, 2005
Inspectors:	E. Guthrie, Senior Resident Inspector A. Sabisch, Resident Inspector L. Cain, Visiting Resident Inspector R. Chou, Reactor Inspector (Section 4OA5.4) A. Vargas, Reactor Inspector (Section R08 and 4OA5.2) S. Vias, Senior Reactor Inspector (Section R08 and 4OA5.2)
Approved by:	Michael E. Ernstes, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

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Enclosure

SUMMARY OF FINDINGS

IR 05000413/2005-003, IR 05000414/2005-003, IR 0720045/2005-001; 4/1/2005 - 6/30/2005; Catawba Nuclear Station, Units 1 and 2; Operability Evaluations

The report covered a three month period of inspection by three resident inspectors (one visiting) and three regional-based inspectors: two senior reactor inspectors and one reactor inspector. Two, Green, non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Technical Specification (TS) 5.4.1.a, written procedures, because the licensee failed to implement adequate post maintenance testing following maintenance in 1RN-38B, 1B Nuclear Service Water (RN) pump discharge valve, electric valve operator control circuit.

  The finding was determined to be greater than minor because 1RN-38B, 1B RN pump discharge valve, was not capable of performing its intended function, which caused the 1B nuclear service water (RN) pump to be inoperable. The inoperability resulted in potential impact on reactor safety and adversely affected the availability and reliability of a mitigating system performance attribute of the reactor safety cornerstone. The finding was determined to be of very low safety significance, using the significance determination phase 1 worksheet, because the inoperability of 1RN-38B and the 1B RN pump did not result in the loss of safety function of the RN train in excess of its TS allowed outage time. This finding involved the cross-cutting aspect of human performance since individuals did not determine adequate post maintenance testing to verify that the valve could perform its intended function following the fuse replacement (Section 1R15b.1).

- Green. A non-cited violation was identified for inadequate design control as required by 10 CFR 50, Appendix B, Criterion III, in that, the licensee found that they had incorrectly assumed that the Unit 1 and Unit 2 containment sump suction valves needed to function under a maximum 20 pound per square inch pressure differential (psid) and then implemented periodic testing under their Generic Letter 89-10 Motor Operated Valve (MOV) testing program to ensure the valves would open against this psid. Subsequent licensee analysis determined that the valves could experience up to 364 psid during specific accident conditions. Because this violation appeared to be of greater significance than the licensee's initial characterization of the issue, this finding is being treated as an NRC-identified violation in accordance with NRC Enforcement Guidance. This finding involved the cross-cutting aspect of human performance since individuals did not determine the proper design parameters and conditions for all required accident scenarios.

This finding was greater than minor because it affected an objective and attribute of the Reactor Safety Mitigating Systems Cornerstone for availability and reliability, in that excessive psid across the containment sump suction valves could prevent the valves from opening and providing a required injection supply source to the emergency core cooling system pumps. The finding was assessed using the significance determination process for Reactor Inspection Findings for At-Power Situations. The evaluation determined that the finding exceeded the threshold that required evaluation under Phase 3 of the significance determination process. The Phase 3 analysis conducted by the Regional Senior Reactor Analyst, determined the finding to be of very low safety significance because the dominant factor in the analysis was that the need for sump recirculation would have to coincide with a degraded grid condition and such an initiating event frequency was sufficiently low enough to conclude the deficiency was Green. (Section 1R15b.2).

B. Licensee-identified Violation

A 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. The violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

- TS 5.5.8 requires that an In-Service Test (IST) Program shall be established, implemented and maintained for American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components. The station's IST Program identified components requiring testing and the frequency at which they were to be tested. The nuclear service water system-to-containment penetration valve injection system supply check valves were within the scope of the IST Program and were required to be tested on a quarterly basis. Contrary to this, on April 20, 2005, the licensee discovered that the work orders used to schedule the required testing had been inadvertently deleted and the tests had not been performed within the specified frequency and allowable grace period. This event is documented in the licensee's corrective action program as Problem Investigation Process Report (PIP) C-05-2089. This finding was of very low safety significance because the valves were subsequently tested successfully which ensured that the nuclear service water system would have been available to supply makeup to the containment penetration valve injection system if required to do so.

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the inspection period operating at 100 percent Rated Thermal Power (RTP). The unit was removed from service for the fifteenth end-of-cycle refueling outage on May 7, 2005. The unit returned to service on June 6, 2005 and remained at 100 percent RTP through the end of the inspection period.

Unit 2 began the inspection period operating at 100 percent RTP. On May 11, 2005, reactor power was reduced to approximately 10 percent RTP and the main turbine removed from service to repair an electro-hydraulic control fluid leak on the #5 Combined Intermediate Valve (CIV). Following repairs, the unit was returned to service on May 13, 2005 and remained at 100 percent RTP through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

Warm Weather Preparation

d. Inspection Scope

The inspectors reviewed the licensee's preparations for adverse weather associated with high ambient temperatures. This included field walkdowns to assess the material condition and operation of ventilation and cooling equipment as well as other preparations made to protect plant equipment from high ambient temperature conditions. Risk significant systems reviewed included portions of the standby shutdown facility and the nuclear service water pump house structure. In addition, the inspectors conducted discussions with operations, engineering, and maintenance personnel responsible for implementing Catawba's hot weather preparation program to assess the licensee's ability to identify and resolve deficient conditions associated with hot weather protection equipment prior to actual hot weather being experienced at the site. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors verified the critical portions of equipment alignments for selected trains that remained operable while the redundant trains were inoperable. The inspectors reviewed plant documents to determine the correct system and power alignments, as well as the required positions of selected valves and breakers. The inspectors verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact mitigating system availability. Documents reviewed are listed in the Attachment to this report. The inspectors verified the following four partial system alignments:

- 1B diesel generator with the 1A diesel generator inoperable due to a failed component in the diesel field excitation circuitry
- On-site and off-site power with the standby shutdown facility removed from service for planned maintenance on the SLXG bus and support equipment
- 2A component cooling water (KC) train safety-related equipment with the 2B KC train out of service for planned maintenance
- 1A diesel generator and associated electrical buses with Unit 1 in mode 6 and the 1B diesel generator removed from service for planned maintenance

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors conducted one detailed walkdown/review involving the alignment and condition of the Unit 2, nuclear service water (RN) system. The inspectors utilized licensee procedures, as well as licensing and design documents to verify that the system (i.e., pump, valve, and electrical) alignment was correct. During the walkdowns, the inspectors also verified that: valves and pumps did not exhibit leakage that would impact their function; major portions of the system and components were correctly labeled; hangers and supports were correctly installed and functional; and essential support systems were operational. In addition, pending design and equipment issues were reviewed to determine if the identified deficiencies significantly impacted the system's functions. Items included in this review were: the operator workaround list, the temporary modification list, system Health Reports, and outstanding maintenance work requests/work orders. A review of open Problem Identification Process reports (PIP) was also performed to verify that the licensee had appropriately characterized and prioritized RN-related equipment problems for resolution in the corrective action program. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

Fire Protection Walkdowns

a. Inspection Scope

The inspectors walked down accessible portions of the plant to assess the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The inspectors observed the fire protection suppression and detection equipment to determine whether any conditions or deficiencies existed which could impair the operability of that equipment. The inspectors selected the areas based on a review of the licensee's safe shutdown analysis probabilistic risk assessment, sensitivity studies for fire related core damage accident sequences, and summary statements related to the licensee's 1992 Initial Plant Examination for External Events submittal to the NRC. Documents reviewed/generated during this inspection are listed in the Attachment to this report. The inspectors toured the following eight areas important to reactor safety:

- Unit 2, Mechanical Penetration Room, 560 foot elevation
- Unit 2, Volume Control Tank Room, 560 foot elevation
- Unit 2, Electrical Penetration Room, 560 foot elevation
- Unit 2, 4160 Volt 'B' Essential Switchgear Room, 560 foot elevation
- Unit 1, B Emergency Diesel Generator Room and Sequencer Hallway, 560 foot elevation
- Unit 1, Main Transformer Yard
- Unit 2, Main Transformer Yard
- Unit 2, Spent Fuel Pool Purge Unit Room, 636 foot elevation

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Piping Systems ISI

a. Inspection Scope

The inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries for Unit 1. The inspectors selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in

Section 71111.08-03 of inspection procedure 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the onsite inspection period.

The inspectors conducted an on-site review of the following types of nondestructive examination activities to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements. Specifically, the inspectors observed and performed record review of the following examinations;

Liquid Penetrant (PT):

- Safety Injection Elbow to Pipe, Weld # B09.0011.157A/1NI32-4
- Safety Injection Elbow to Pipe, Weld # B09.0011.158A/1NI32-5
- Safety Injection Elbow to Pipe, Weld # B09.0011.156A/1NI32-3

Ultrasonic Examination (UT):

- Steam Generator Inlet, Inlet Nozzle Inner Radius, Weld # B03.140.005/1SGC-INLET
- Steam Generator Inlet, Outer Nozzle Inlet Radius, Weld # B03.140.006/SGC-OUTLET
- Steam Generator Inlet, Nozzle to Safe End, Weld #B05.070.005/1SGC-INLET-W5SE
- Chemical and Volume Control (NV) System, Pipe to Elbow, Weld #C05.021.107/INV97-7

The inspectors also reviewed examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service to verify that the licensee's acceptance was in accordance with the Section XI of the ASME Code. The inspectors reviewed pressure boundary welds for Code Class 1 or 2 systems which were completed during the previous refueling outage, to verify that the welding acceptance and pre-service examinations were acceptable.

The inspectors performed a review of piping system ISI related problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. Additionally, the inspectors' review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspectors evaluated the threshold for identifying issues through interviews with licensee staff and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspectors performed these reviews to ensure compliance with 10 CFR 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.

.2 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

The inspectors reviewed the Unit 1 BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary.

The inspectors conducted an on-site record review and direct observation of the BACC visual examination activities to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requirements. In particular, the inspectors verified that the visual examinations focused on locations where boric acid leaks can cause degradation of safety significant components and that degraded or non-conforming conditions were properly identified in the licensee's corrective action system. The inspectors performed observations and record reviews of the visual examinations, reviewed the visual examination procedures and examination records for the BACC examination conducted.

The inspectors reviewed licencee corrective actions implemented for evidence of boric acid leakage to confirm that they were consistent with requirements of Section XI of the ASME Code and 10 CFR 50 Appendix B Criterion XVI. Specifically, the inspectors reviewed:

- PIP C-04-02962, 2B Containment Spray (NS) system found to have leakage
- PIP C-04-03868, small amount on valve body to bonnet joints
- PIP C-04-04262, through wall leak on Boric Acid Tank #2
- PIP C-05-0206, Atlantic Quality Control (QC) Inspector does not have current eye exam

b. Findings

No findings of significance were identified.

.3 Steam Generator (SG) Tube ISI

a. Inspection Scope

The inspectors reviewed the Unit 1 SG tube examination activities conducted pursuant to Technical Specification (TS) 5.5.9, Steam Generator Program, which incorporated the Generic License Change Package (GCLP) and the ASME Code Section XI requirements.

The inspectors reviewed the SG examination scope, expansion criteria, eddy current testing (ET) acquisition procedures, ET analysis procedures, the SG Operational

Assessment, in-situ tube pressure testing procedures, and records and examination reports to confirm that:

- The SG tube ET (Eddy Current Testing) examination scope was sufficient to identify tube degradation confirming that the ET scope completed was consistent with the licensee's procedures and plant TS requirements. Additionally, the inspectors reviewed the SG tube ET examination scope to determine that it was consistent with TS requirements and included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions and support plates.

- The licensee adequately followed-up on a new tube degradation mechanism other than what was predicted in the SG tube degradation assessment, specifically indications from loose parts on the secondary side.

- The SG tube repair criteria and process (plugging and sleeving) was consistent with TS requirements and the licensee was only applying the TS plugging limit at tube wear locations.

- The ET probes and equipment configurations used to acquire ET data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with TS requirements.

- The licensee adequately examined for loose parts indications and secondary side tube condition.

- The licensee adequately evaluated for any contractor deviations from their ET data acquisition or analysis procedures or with TS requirements.

The inspectors performed a review of SG ISI related problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. Additionally, the inspectors' review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspectors evaluated the threshold for identifying issues through interviews with licensee staff and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed a simulator exercise conducted on April 20, 2005, to assess the performance of licensed operators. The scenario, PTRQ Task Requirement Guide, Loss of Residual Heat Removal (ND) (AP/19 Case I, Case II, and Case V), involved a loss of the residual heat removal system while the plant was shutdown, the primary system was not fully filled, and a large vent path existed. The inspection focused on high-risk operator actions performed during implementation of the emergency operating procedures, emergency plan implementation and classification, and the incorporation of lessons learned from previous plant events. Through observations of the critique conducted by training instructors following the exam session, the inspectors assessed whether appropriate feedback was provided to the licensed operators regarding identified weaknesses.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee's effectiveness in performing routine maintenance activities. This review included an assessment of the licensee's practices pertaining to the identification, scope, and handling of degraded equipment conditions, as well as common cause failure evaluations and the resolution of historical equipment problems. For those systems, structures, and components scoped in the maintenance rule per 10 CFR 50.65, the inspectors verified that reliability and unavailability were properly monitored, and that 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. The inspectors conducted this inspection for the degraded equipment conditions associated with the two items listed below. Documents reviewed are listed in the Attachment to this report.

- 1A diesel generator troubleshooting and repair of the generator field excitation circuitry
- Unit 2, containment air return fan 2B, discharge damper valve 2ARF-D-4 timer did not stroke the valve in the required time period

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 six emergent and planned 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)," for appropriate guidance to comply with 10 CFR 50.65 (a)(4). Documents reviewed are listed in the Attachment to this report.

- Planned and emergent work during performance of the Complex Maintenance Evolution Plan for 1A diesel generator pre-outage battery modification work
- Severe weather predicted concurrent with the 1A diesel generator being out of service
- Planned and emergent work during performance of the Complex Maintenance Evolution Plan for 1B diesel generator pre-outage battery modification work
- Replacement of the 1A diesel generator batteries when a 2A diesel generator battery cell was found to have dropped below the administrative limit
- Work on the 1A RN pump performed while the 1A diesel generator battery replacement project was in progress
- Failure of the 1B RN pump discharge valve (1RN-38B) to stroke in the open direction caused inoperability of RN pump

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope

On May 7 and 8, 2005, the inspectors observed operator performance during the shutdown of Unit 1 for the end-of-cycle, fifteenth (1EOC15) refueling outage. The inspectors observed licensed operators use of procedures, control room pre-evolution briefings and plant equipment manipulations during the power reduction, manual reactor trip and portions of the subsequent plant cooldown. Documents reviewed are listed in the Attachment to this report.

On May 11, 2005, the inspectors observed operator performance following the identification of an electro-hydraulic control fluid leak on the Unit 2 low pressure turbine #5 CIV. The leak necessitated the rapid reduction of power and tripping of the main turbine. The reactor remained critical and was stabilized at approximately 10 percent RTP. The inspectors verified operator actions and use of procedures in stabilizing the unit. In addition, the inspectors reviewed selected trend graphs for parameters following

the rapid power reduction to verify the plant responded as expected. Documents reviewed are listed in the Attachment to this report.

On May 13, 2005, the inspectors observed operator performance returning the Unit 2 main generator to service. The generator was tripped due to indications of high vibration on the #6 bearing. The inspectors observed licensed operators use of procedures, the control room pre-evolution briefing and plant equipment manipulations following the manual trip of the main generator.

On June 5, 2005, the inspectors observed operator performance during the start-up of Unit 1 following the 1EOC15 refueling outage. The inspectors observed licensed operators' use of procedures, control room pre-evolution briefings, and plant equipment manipulations during the reactor approach to criticality and performance of portions of zero power and startup physics testing. Documents reviewed are listed in the Attachment to this report

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability evaluations to verify that the operability of systems important to safety were properly established, that the affected components or systems remained capable of performing their intended safety function, and that no unrecognized increase in plant or public risk occurred. Operability evaluations were reviewed for the eight issues listed below. Documents reviewed are listed in the Attachment to this report.

- Re-analysis of the reactor coolant pump seal volume available to mitigate a loss of seal cooling event (PIP C-05-1033)
- Numerous spurious fire detection system alarms caused by aluminum oxide buildup (PIP C-05-2006)
- Nuclear Instrument Power range detector uncertainty (PIP C-05-1754)
- Residual heat removal (ND) system containment sump suction isolation valve differential pressure across the valve was found to be significantly higher than design values during some postulated accident conditions (PIP C-05-2259)
- Diesel engine starting air storage tank inlet check valves failing to close due to foreign material in the air lines (PIP C-05-2943)
- Control Area Chilled Water (YC) chiller 'A' control panel indication for "Compressor Starter Overload" illuminated during 'A' train operation (PIP C-05-3011)
- Cavitation noise observed on KC pump 2A2 (PIP C-05-3397)
- 1RN38B, 1B RN pump discharge valve did not close when the pump was secured as it should have (PIP C-05-2591)

b. Findings

(1) 1B RN Pump Discharge Valve

Introduction: A Green Non-Cited Violation (NCV) was identified for the failure to implement adequate post maintenance testing following maintenance on 1RN-38B, 1B RN pump discharge valve electric valve operator control circuit, in accordance with TS 5.4.1.a.

Description: On May 12, 2005 the licensee had found that 1RN-38B did not close, as expected, when the pump was secured. Troubleshooting determined that a fuse was blown in the control circuit of the valve. The licensee performed an operability assessment for this condition and determined that the 1B RN pump remained operable since the pump was able to perform its intended function. An additional concern was addressed for the condition when the pump was secured with the discharge valve remaining open, that the potential existed for reverse rotation of the pump and system flow diversion. The licensee verified that the 1B RN pump discharge check valve was stopping reverse flow, so the RN system remained operable. The inspectors reviewed the condition and agreed that the pump and system remained operable.

The licensee replaced blown fuses in the 1RN-38B electric valve operator control circuit. The valve was cycled closed, following the maintenance, as the 1B RN pump was secured. On May 13, 2005, the 1B RN pump discharge valve did not open, as expected, when the pump was started. The operators noted the discharge valve did not open and secured the pump immediately. The operators declared the pump inoperable and entered the appropriate TS based on plant conditions. Based on operable plant equipment at the time of the 1B RN pump being declared inoperable, the operators entered TS 3.0.3, due to a condition prohibited by TS. TS 3.0.3 was entered because both trains of control room chillers were inoperable, one was already inoperable for maintenance and the other became inoperable when the 1B RN pump was declared inoperable. The licensee remained in TS 3.0.3 for less than one hour while the RN system was aligned for single header operation, which restored operability to one of the control room chillers.

The inspectors found that the licensee had replaced the blown fuses in the 1RN-38B valve control circuit on May 12, 2005, and cycled the valve in the close direction only. Cycling the valve closed was considered the post maintenance retest, by the licensee. The licensee's Work Process Manual, WPM 501, section 501.5.1, Post Maintenance Testing, stated in part that "all appropriate testing (installation tests, functional verifications or retests) performed following maintenance to verify that a system, structure, or component may be considered operable or returned to service. Post-Maintenance Testing shall verify that "the system, structure, or component can perform its intended function, that any original deficiency has been addressed...". The inspectors determined that work order, 98727279, 1RN-038B I/R Valve Fail To Close, specified a post maintenance retest be performed using periodic test PT/1/A/4200/013C, RN Valve Inservice Test. The periodic test procedure stated, in part, that the purpose of the procedure was "following maintenance or repair on any valve covered by this procedure that could affect the valve operability the valve shall be

retested to verify stroke time, operability, valve fail-safe actuator verification as necessary." The inspectors found that the performance of PT/1/A/4200/013C would not have adequately retested the valve since the circuit that was integral with the fuse replacement was jumpered out of the circuit during the implementation of the procedure. The inspectors concluded that an adequate retest was not developed for this maintenance activity and adequate post maintenance testing was not performed to verify the valve could perform its intended function following the fuse replacement.

Analysis: The finding is greater than minor because the 1RN-38B was not capable of performing its intended function, which caused the 1B RN pump to be inoperable. The inoperability resulted in potential impact on reactor safety and adversely affected the availability and reliability of a mitigating system performance attribute of the reactor safety cornerstone. The finding was determined to be of very low safety significance, using the significance determination process (SDP) phase 1 worksheet, because the inoperability of 1RN-38B and the 1B RN pump did not result in the loss of safety function of the RN train in excess of its TS allowed outage time. This finding involved the cross-cutting aspect of human performance since individuals did not determine adequate post maintenance testing to verify that the valve could perform its intended function following the fuse replacement.

Enforcement: Technical Specification 5.4.1.a requires that written procedures be established, implemented and maintained covering applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 including maintenance procedures. Work Process Manual, WPM 501, Post Maintenance Testing requires the performance of a post maintenance test to verify the 1RN-38B, 1B RN pump discharge valve, could perform its intended function. Contrary to the above, on May 12, the licensee failed to implement Work Process Manual, WPM 501, Post Maintenance Testing when the 1RN-38B, 1B RN pump discharge valve, was not adequately post maintenance tested following maintenance in the electric valve control circuitry to verify the valve could perform its intended function. Because this issue was of very low safety significance and was placed in the corrective action program as PIP C-05-2655, this violation is being treated as a non-cited violation in accordance with Section VI.A.1 of the Enforcement Policy, and is identified as NCV 05000413/2005003-01, Inadequate Post Maintenance Testing on 1RN-38B, 1B RN Pump Discharge Valve.

(2) Containment Sump Suction Valves

Introduction: A Green, non-cited violation was identified for inadequate design control as required by 10 CFR 50, Appendix B, Criterion III, in that, the licensee incorrectly assumed that the Unit 1 and Unit 2 containment sump suction valves needed to function under a maximum 20 pounds per square inch pressure differential (psid) and then implemented periodic testing under their Generic Letter 89-10 MOV testing program to ensure the valves would open against this psid. Subsequent analysis determined that the valves could experience up to 364 psid during specific accident conditions.

Description: The containment sump suction valves were designed to open when Refueling Water Storage Tank (FWST) water level reached the low-level setpoint

coincident with a safety injection signal to provide a continuous makeup supply to the Emergency Core Cooling System (ECCS) during a Loss of Coolant Accident (LOCA). The licensee was required to periodically ensure the valves would open against the expected accident condition psid as prescribed in the licensee's Generic Letter 89-10 MOV Testing Program. Catawba Nuclear Station (CNS) Calculation, CNC-1205.19-00-0044, "Generic Letter 89-10 MOV Calculation, safety injection (NI) System: 1(2)NI185B and 1(2)NI184B," assumed an opening force of 20 psid. The four valves (two per unit) were tested on a regular basis; however, the acceptance criteria defined in the procedure was the 20 psid value. On April 28, 2005, the licensee determined that if the residual heat removal (ND) pumps were secured following an auto-start signal and component cooling water (KC) was not established to the ND heat exchangers during a small-break LOCA where reactor coolant system pressure did not decrease to the injection pressure of the ND system, that the pressure trapped in the suction piping could result in an elevated psid being developed across the containment sump suction valves. This concern could have potentially prevented the valves from opening automatically when required.

The licensee's failure to consider the scenario where the ND pump would be secured without KC aligned to the heat exchangers, thereby allowing the ND suction piping to pressurize resulted in inadequate testing being performed on the containment sump suction valves to ensure they would open against the psid postulated to be experienced during all possible accident conditions.

The inspectors reviewed the licensee's proposed compensatory actions and raised questions related to personnel safety and the ability to perform these actions under all accident conditions. Based on the apparent safety significance, the inspectors initiated the SDP.

The licensee subsequently performed an analysis to determine if the containment sump suction valves would have been able to open against the maximum postulated psid by using the lowest 4160 volt bus voltages recorded over the previous two years. Based on these calculations, the four (4) valve motor operators would have produced sufficient torque to open the valves against the maximum postulated psid they could have had to operate under during accident conditions where a demand to open signal may have existed.

Analysis: The performance deficiency associated with this finding was that the licensee did not adequately analyze all possible configurations the RHR system could be placed in during accident conditions to ensure the appropriate psid was used in establishing the test criteria in the station's Generic Letter 89-10 MOV testing program and subsequently verify operability of the containment sump suction valves against this psid. This finding was greater than minor because it affected an objective and attribute of the Reactor Safety Mitigating Systems Cornerstone, in that excessive psid across the containment sump suction valves could prevent the valves from opening and providing a required injection supply source to the ECCS pumps. The finding was assessed using the SDP for Reactor Inspection Findings for At-Power Situations. Based on information available in PIP C-05-2259, the finding was evaluated using the SDP Phase 2 plant notebook and it was determined a Phase 3 evaluation was required. A regional SRA performed a

Phase 3 SDP evaluation and determined the performance deficiency was of very low safety significance (Green). The dominant factor in the analysis was that the need for sump recirculation would have to coincide with a degraded grid condition. Such an initiating event frequency was sufficiently low enough to conclude the deficiency was Green without considering possible recovery actions or alternate accident mitigation strategies. Therefore, the critical assumption was that the containment sump valves would fail to open at their normal failure rate at nominal voltage. This finding involved the cross-cutting aspect of human performance since individuals did not determine the proper design parameters and conditions for all required accident scenarios.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, requires, in part, that "Measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, on April 28, 2005, the licensee determined that they had failed to properly establish the applicable design basis for the maximum differential pressure against which the containment sump suction valves would be required to open under all postulated accident conditions and correctly translate these pressures into specifications and procedures, such as the station's Generic Letter 89-10 MOV Testing Program.

Because this violation appeared to be of greater significance than the licensee's initial characterization of the issue, this finding is being treated as an NRC-identified violation in accordance with NRC Enforcement Guidance. The licensee entered the issue into its corrective action program as PIP C-05-2259. Because this violation was of very low safety significance and because it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000413, 414/2005003-02, Failure to Adequately Evaluate Potential Residual Heat Removal (RHR) System Differential Pressure During Postulated Accident Conditions In Generic Letter 89-10 MOV Testing Program.

1R16 Cumulative Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative Catawba Nuclear Station Operator Workaround List for potential affects on the functionality of mitigating systems. The workarounds were reviewed to determine: (1) if the functional capability of the system or human reliability in responding to an initiating event was affected; (2) the affect on the operator's ability to implement abnormal or emergency procedures; and (3) if operator workaround problems were captured in the licensee's corrective action program. Aggregate impacts of the identified workarounds on each individual operator watch station were also reviewed. Documents reviewed for this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- 2B chemical and volume control (NV) pump in-service test following seal repair
- Post maintenance testing following the repair of the excitation circuit on the 1A Diesel Generator (DG)
- Functional testing of the volume control tank normal makeup system following replacement of the Unit 2 boric acid turbine flow meter (2NVFT5450)
- 1A diesel generator operability test following planned maintenance during 1EOC-15
- Operability test on the 1A NV pump following 1EOC15 preventive and corrective maintenance
- Stroke testing on Unit 1 safety injection (NI) valves following maintenance
- 1B diesel generator operability test following planned maintenance during 1EOC-15
- In-Service Testing of 1B diesel generator starting air check valves, 1VG-5 and 1VG-7, following maintenance to correct the failure of the initial surveillance due to foreign material in the air line
- Performance test of the ‘A’ control room area chiller following maintenance to address tripping of the chiller due to a failed refrigerant temperature switch

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors evaluated Unit 1 outage activities to ensure that the licensee considered risk in developing and implementing outage schedules; adhered to administrative risk reduction methodologies developed to control plant configuration; developed mitigation strategies for losses of key safety functions; and adhered to operating license and Technical Specification (TS) requirements that ensure defense-in-depth. The following specific areas were reviewed:

- Review of Outage Plan - Prior to the outage, the inspectors reviewed the licensee's outage risk control plan, attended risk briefings and Plant Operational Review Committee meetings, and verified that the licensee appropriately considered risk, industry experience, and previous site specific problems. The inspectors reviewed the licensee's contingency actions for losses of key safety functions, and verified that the licensee maintained key safety function status and controls throughout the outage. The inspectors reviewed the Unit 1 outage risk assessment CN-05-007, 1EOC-15-IRT Pre-Outage Review, Shutdown Risk Assessment.

- Outage Configuration Management - The inspectors assessed the licensee's management of configuration control and the risk associated with outage activities by reviewing the licensee's implementation of Site Directive 3.1.30, Unit Shutdown Configuration Control (Modes 4, 5, 6 or No Mode) and Nuclear Site Directive, NSD-403, Shutdown Risk Management (Modes 4, 5, 6 or No Mode) per 10CFR50.65(a)(4). This assessment included verification that the licensee maintained defense-in-depth commensurate with the outage risk control plan for key safety functions and applicable TS when risk significant equipment was removed from service. The inspectors also assessed whether configuration changes due to emergent work and unexpected conditions were controlled in accordance with the outage risk control plan, and if control room operators were cognizant of plant configuration.

- Electrical Power - The inspectors reviewed the status and configurations of electrical systems for compliance with TS requirements and the licensee's outage risk control plan. The inspectors verified that switchyard activities were controlled commensurate with safety and were consistent with the licensee's outage risk control plan. The inspectors reviewed Site Directive 3.1.30, Unit Shutdown and CN-05-007, 1EOC-15-IRT Pre-Outage Review, Shutdown Risk Assessment. The inspectors also reviewed the implementation of PT/1/A/4350/003, Electrical Power Source Alignment Verification.

- Clearance Activities - The inspectors verified that tags were properly hung and that associated equipment was appropriately configured to support the function of the clearance and that the tags were properly removed when the equipment was returned to service. Specifically, the inspectors reviewed the tag removal and equipment restoration of the fuses and breakers associated with the four Unit 1 reactor coolant pump motors under tagout 05-00538.

- Spent Fuel Pool Cooling System Operation - The inspectors verified that outage work was not impacting the ability of operators to operate the spent fuel pool cooling system during and after core offload. This verification included the review of OP/1/A/6200/005, Spent Fuel Cooling System, the review of control room indications specific to the spent fuel cooling system and the spent fuel pool, and discussions with control room licensed operators.

- Inventory Control - The inspectors reviewed flow paths, configurations, and alternative means for inventory addition to verify they were consistent and

maintained in accordance with the outage risk plan, CN-05-007, 1EOC-15-IRT Pre-Outage Review, Shutdown Risk Assessment. The inspectors reviewed reactor vessel inventory controls to verify they were adequate to prevent inventory loss.

- Reactor Coolant System Instrumentation - The inspectors verified that reactor coolant system level and temperature instruments were installed and configured to provide accurate indication, and that instrumentation error was properly addressed. This verification included a review of OP/1/A/6150/006, Draining The Reactor Coolant System, and the observation of lowering reactor water level activities.

- Reduced Inventory and Mid-Loop Conditions - The inspectors reviewed the licensee's commitments from Generic Letter 88-17, Loss of Decay Heat Removal, and confirmed they were adequately implemented. The inspectors verified that the configuration of plant systems during reduced inventory and mid-loop conditions were in accordance with Generic Letter 88-17 commitments. The inspectors observed control room activities during mid-loop conditions and verified that licensed operators could maintain required reactor vessel level. The inspectors reviewed the following documents and their implementation:

  - OP/1/A/6150/001, Filling and Venting the Reactor Coolant System, Enclosure 4.16, Reactor Coolant System Vacuum Refill Without Solid Operation
  - OP/1/A/6150/006, Draining the Reactor Coolant System; Enclosure 4.2, Decreasing the NC System Level; Enclosure 4.3, Increasing the NC System Level, and Enclosure 4.10, Requirements for Operation with the NC System Level Below 16%
  - Site Directive 3.1.30, Unit Shutdown Configuration Control (Modes 4,5,6 or No Mode).

- Reactivity Control - The inspectors reviewed reactivity control to verify that proper control was maintained in accordance with the TS and Site Directive 3.1.30, Unit Shutdown Configuration Control (Modes 4,5,6 or No Mode) and NSD 403, Shutdown Risk Management (Modes 4, 5, 6 or No Mode) per 10CFR50.65(a)(4). Potential reactivity changes were identified in the outage risk plan, CN-05-007, 1EOC-15-IRT Pre-Outage Review, Shutdown Risk Assessment, and were reviewed to verify proper controls.

- Containment Closure - The inspectors verified that the licensee controlled containment penetrations in accordance with the refueling operations TS, and that containment closure could be achieved when needed. The inspectors observed containment penetration control activities conducted in the control room and in the plant. The inspectors reviewed the following documents and their implementation:

  - Site Directive 3.1.30, Unit Shutdown Configuration Control (Modes 4,5,6 or No Mode)
  - NSD 403, Shutdown Risk Management (Modes 4, 5, 6 or No Mode)

per 10CFR50.65(a)(4)
- PT/1/A/4200/002C, Containment Closure Verification (Part I)
- PT/1/A/4200/002I, Containment Closure Verification (Part II)
- PT/1/A/4200/002J, Containment Closure Verification Penetration Status Change
- OP/0/A/6100/014, Penetration Control for Modes 5 and 6

- Refueling Activities - The inspectors reviewed fuel handling operations to verify they were performed in accordance with approved fuel handling procedures. Specifically, the inspectors observed portions of the new fuel receipt inspection, verified the positions of randomly selected new fuel assemblies and verified that these assemblies were tracked and placed in the correct position and orientation during movement from the new fuel vault area into the spent fuel pool. The inspectors also observed the coordination and movement of several fuel assemblies from the reactor vessel to the spent fuel pool during core offload as well as the coordination and movement of several fuel assemblies from the spent fuel pool to the reactor vessel during core reload. The inspectors also reviewed the completed total core reload procedure (PT/0/A/4550/003C) and viewed the videotape of the final fuel assembly in-core position verification. The inspectors reviewed the following documents and their implementation:

  - PT/0/A/4150/037, Fuel/Component Movement Accounting
  - OP/1/A/6550/006, Transferring Fuel with the Spent Fuel Manipulator Crane
  - OP/1/A/6550/007, Reactor Building Manipulator Crane Operation
  - OP/1/A/6550/008, Fuel Transfer System Operation
  - MP/0/B/7150/012, Refueling Canal Cleanliness
  - PT/1/A/4550/001C, Refueling Communications Test
  - PT/0/A/4150/017, Total Core Unloading
  - PT/0/A/4150/017, Total Core Unloading Tailgate Briefing
  - PT/0/A/4550/003C, Core Verification
  - PT/0/A/4150/022, Total Core Reloading
  - PT/0/A/4150/022, Total Core Reloading Tailgate Briefing

- Monitoring of Heatup and Startup Activities - The inspectors reviewed TS, license conditions, commitments, and administrative procedure prerequisites for mode changes to verify they were met for changing plant configurations. The inspectors performed a walkdown of primary containment prior to reactor startup to verify that debris had not been left which could affect performance of the containment sumps. In addition, the inspectors conducted a walkdown of the upper and lower ice condenser areas to verify that debris had not been left which could affect ice condenser performance. The inspectors observed the reactor startup, the approach to criticality and portions of the power ascension program. The inspectors reviewed the following documents and their implementation:

  - PT/0/A/4200/002, Containment Cleanliness Inspection
  - SM/0/A/8510/008, Ice Condenser FME Inspection
  - PT/0/A/4150/001J, Zero Power Physics Testing
  - PT/0/A/4150/001, Zero Power Physics Testing tailgate briefing

- PT/0/A/4150/001, Controlling Procedure for Startup Physics Testing
- OP/1/A/6100/001, Controlling Procedure for Unit Startup
- OP/1/A/6100/003, Controlling Procedure for Unit Operations
- OP/1/B/6300/001, Turbine Generator Startup

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed the surveillance tests listed below to verify that TS surveillance requirements and/or Selected Licensee Commitment requirements were properly complied with, and that test acceptance criteria were properly specified. The inspectors verified that proper test conditions were established as specified in the procedures, that no equipment preconditioning activities occurred, and that acceptance criteria had been met. Additionally, the inspectors also verified that equipment was properly returned to service and that proper testing was specified and conducted to ensure that the equipment could perform its intended safety function following maintenance or as part of surveillance testing. Additional documents reviewed during this inspection are listed in the Attachment to this report. The following six activities were reviewed:

Surveillance Tests:

- PT/1/A/4200/013H, NI/NV Check Valve Test

Containment Isolation Valve Test:

- PT/1/A/4200/001 C, As Left Containment Isolation Valve Leak Rate Test; Enclosure 13.16, Penetration Number M323 As Left Type C Leak Rate Test for valves 1KC-429B (Reactor Building Drain Header Containment Isolation Valve) and 1KC-47 (Reactor Building Drain Header Pressure Equalization Check Valve)

In-Service Test:

- PT/1/A/4200/007C, Standby Makeup Pump #1 Performance Test
- PT/2/A/4200/004B, Containment Spray Pump 2A Performance Test

Ice Condenser Surveillance Tests:

- MP/0/A/7150/005, Ice Basket Weight Determination; Unit 1, Bay 19
- MP/0/7/7150/006, Ice Condenser Lower Inlet Doors Inspection and Testing

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed and evaluated the licensee's performance during an emergency drill conducted on June 21, 2005. The inspectors observed licensee activities occurring in the Control Room Simulator and in the Technical Support Center. The NRC's assessment focused on the timeliness and location of classification, the notification and protective action recommendations process activities, and the licensee's expectations of response. The performance of the emergency response organization was evaluated against applicable licensee procedures and regulatory requirements. The inspectors attended the post-exercise critique for the drill to evaluate the licensee's self-assessment process for identifying potential deficiencies relating to failures in classification and notification, as well protective action recommendation process activities. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

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

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

.2 Annual Sample Review

dd. Inspection Scope

The inspectors selected one PIP for detailed review. PIP C-05-3120 involved an unexpected diversion of letdown flow, during cation bed flushing operations, that resulted in a Unit 2 Volume Control Tank (VCT) level decrease of approximately 6 percent over a 10-minute span and an automatic VCT inventory makeup. The PIP was reviewed to determine whether the full extent of the issues were identified, an

appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the PIP against the requirements of the licensee's corrective action program document and 10 CFR 50, Appendix B.

b. Findings

No findings of significance were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's Corrective Action Program (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 nominally considered the six month period of January 2005 through June 2005, although some examples expanded beyond those dates when the scope of the trend warranted. 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. Some of the items reviewed are listed in the Attachment to this report. 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.

ee. Assessment and Observations

No findings of significance were identified. In general, the licensee has identified trends and has appropriately addressed the trends with their CAP. The inspectors had previously observed a trend associated with inadequate procedure use and adherence that the licensee had not previously fully recognized. This trend was identified based on actual inspector observations of several activities performed by various organizations of licensee personnel. The inspectors continued to engage the licensee regarding lack of PIP generation for issues that involve human performance. The inspectors have noted and discussed with the licensee several occasions where PIPs were not generated or not identified as a human performance concern for issues that involved some aspect of human performance.

The inspectors found that station management, in response to the inspector identified trend, had begun implementing a site wide focus initiative in January 2005 covering the following aspects: self reporting; conducting cross disciplinary management observations focusing on procedure use and adherence behaviors; a common cause problem evaluation/assessment on procedure use and adherence conducted by work control, operations, and maintenance; and the generation of a 'human performance

report card' that includes measures for tracking procedure use and adherence indicating group and team error rates (colored from green to red) and employing feedback and reward incentives. The site wide focus initiative was found by the inspectors to be comprehensive. According to station management, the initiative was developed and implemented, to create a sustained culture change.

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

NCV 05000413/2005003-01, Inadequate Post Maintenance Testing on 1RN-38B, 1B RN Pump Discharge Valve, described in Section 1R15.1, contained elements of human performance since individuals did not determine adequate post maintenance testing to verify that the valve could perform its intended function following the fuse replacement.

NCV 05000413, 414/2005003-02, Failure to Adequately Evaluate Potential RHR System Differential Pressure During Postulated Accident Conditions In Generic Letter 89-10 MOV Testing Program, described in Section 1R15.2, contained elements of human performance since individuals did not determine the proper design parameters and conditions for all required accident scenarios.

4OA5 Other Activities

.1 Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles Inspection - Temporary Instruction (TI) 2515/150, Bare Metal Visual Examination

The inspectors independently reviewed photographs taken during the Unit 1 reactor vessel head inspection. The inspectors verified that the individuals involved in the head inspection were qualified examiners based on classroom training, examination, and practical testing by reviewing licensee document Form QA-140E for the individuals involved in the inspection. The inspectors reviewed the completed inspection procedure MP/0/A/7150/042 D, Reactor Vessel Head Penetration Visual Inspection. The procedure described the inspection criteria to identify signs of nozzle penetration leakage. The inspectors reviewed the licensee's evaluation and disposition of the areas identified by the examiners as requiring engineering evaluation. The inspectors observed that the head area had minor dirt and grit deposits and some evidence of boron on the head. The licensee determined through a review of past inspection video tape records and procedural documentation that all identified boron deposits were old, and that they originated from locations above the head. The inspectors assessed the adequacy of the conditions under which this inspection was performed (i.e., lighting, removal of insulation, and absence of obstruction for viewing the nozzle penetrations). No nozzle penetrations were identified to be leaking nor were any deficiencies identified that needed repair.

.2 TI 2515/160, Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)

The inspectors reviewed the licensee's 60-day response to NRC Bulletin 2004-01, dated July 27, 2004. The inspectors verified that the licensee's examinations conducted during the Unit 1 refueling outage were consistent with the licensee's response. The

inspectors observed the Bare Metal Visual (BMV) examination performed on the following sample of the welds that fall under the scope of the bulletin:

- Pressurizer Spray Line
- 1NC002- Safety Relief Valve Line
- 1NC002- Safety Relief Valve Line
- 1NC003- Safety Relief Valve Line
- PORV Line
- Pressurizer Manway

For each of the examination methods used during the outage, was the examination:

- Performed by qualified and knowledgeable personnel? The inspectors verified that the examination personnel were VT-1 and VT-2 qualified in accordance with the licensee written practice, and response to Bulletin 2004-01.

- Performed in accordance with demonstrated procedures? The inspectors reviewed the licensee's BMV examination procedure for compliance to inspection requirements, and to ensure that it contained specific instructions related to the identification, disposition, and resolution of deficiencies.

- Able to identify, disposition, and resolve deficiencies? Through application of qualified procedures and examination personnel, the licensee was able to identify, disposition, and resolve any boric acid indications.

- Capable of identifying the leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01? The inspectors verified that the licensee's examination personnel were capable of identifying any leakage in pressurizer penetration nozzles or steam space piping components.

- What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)? There were no viewing obstructions, the insulation was completely removed from the identified components and there was some presence of insulation material on the surface.

- How was the visual inspection conducted (e.g., with video camera or direct visual by the examination personnel)? The examination was conducted by the direct visual examination technique.

- How complete was the coverage (e.g., 360° around the circumference of all the nozzles)? The licensee was able to view the entire circumference, 360 degrees, around each component.

- Could small boron deposits, as described in the Bulletin 2004-01, be identified and characterized? The examination personnel were appropriately trained and qualified to identify small boron deposits as described in the bulletin.

- What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair? There were no deficiencies identified that required repair.

- What, if any, impediments to effective examinations, for each of the applied methods, were identified (e.g., centering rings, insulation, thermal sleeves, instrumentation, nozzle distortion)? There were no impediments for an effective examination.

- If volumetric or surface examination techniques were used for the augmented inspections examinations, what process did the licensee use to evaluate and dispose any indications that may have been detected as a result of the examinations? In accordance with the licensee's response, only a BMV examination was conducted this outage, and there were no indications identified that required further examination.

- Did the licensee perform appropriate follow-up examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system? There were no indications of boric acid leaks from pressure-retaining components in the pressurizer system.

.3 (Closed) TI 2515/163, Operational Readiness Of Offsite Power

During this report period, inspectors collected data from licensee maintenance records, event reports, corrective action documents and procedures, and through interviews of station engineering, maintenance, and operations staff, as required by TI 2515/163. Appropriate documentation of the results was provided to headquarters staff for further analysis, as required by the TI. This completes the Region II inspection requirements in this TI for the Catawba Nuclear Station.

.4 Independent Spent Fuel Storage Installation (ISFSI)

a. Inspection Scope

The inspectors examined installation of the reinforcing steel, observed the concrete pour, and reviewed documents for the ISFSI Cask Storage Pad.

The inspectors examined reinforcing steel and wooden concrete forms for ISFSI pads 1 and 2, section numbers 1 and 2 to ensure that they were installed within cleanliness and tightness requirements, and that the licensee had measured the reinforcing steel diameter, spacing, splice length, and the concrete minimum protection coverage in accordance with the requirements of the design drawings and the American Concrete Institute.

The inspectors witnessed placement of concrete for ISFSI pads 1 and 2, section numbers 1 and 2. The inspectors observed placement activities to ensure that activities pertaining to concrete delivery time, flow distance, layer thickness and concrete consolidation or vibration conformed to industry standards established by the American Concrete Institute. Concrete batch tickets were examined to ensure that the specified

concrete mix was being delivered to the site. The inspectors also witnessed in-process testing and reviewed the results of the concrete for slump, air content, temperature, unit weight, and molding of the concrete cylinders for the compressive strength testing, and reviewed to ensure that concrete samples for the field testing and cylinders for the testing were obtained at the point of placement (end of pump line). The inspectors ensured that concrete field testings were performed and the cylinders were molded in accordance with applicable American Society for Testing and Materials (ASTM) requirements. In addition, the inspectors reviewed activities to ensure that concrete field testing was performed by qualified inspectors from an independent testing company, and that concrete placement activities were continuously monitored by the licensee and contractor engineers and management.

The inspectors also reviewed records documenting inspection of the concrete batch plant and the concrete truck mixers performed by an independent engineering and consulting company. Activities were reviewed to determine if the consulting company's inspection of the trucks and batch plant were performed in accordance with the Plant Certification Checklist of the National Ready Mixed Concrete Association (NRMCA); the batch plant scales were calibrated in accordance with NRMCA recommendations; and mixer efficiency tests were performed on the truck mixers in accordance with ASTM C-94. The inspectors reviewed the concrete mix data to ensure that mix proportions for delivered concrete were selected based on trial concrete mix results, and that the trial mix met concrete strength requirements.

The inspectors also examined and measured the vertical concrete cask (VCC) steel liners on site with the NAC representative and a Duke engineer. The measurements included diameters, dimensions, and fillet weld sizes.

b. Observations

During the inspection the inspectors were told by the licensee that the current concrete pad design could not satisfy seismic requirements of the NAC-UMS general license for the spent fuel cask storage system. The Areva (the engineering consulting company) design calculation document identifier 32-5051258-01, Rev. 0, Catawba Nuclear Station Structural Analysis of Pads for ISFSI Storage Modules documented the maximum pad seismic acceleration of 0.40 g, based on the soil properties from the soil drilling at the pad location, which exceeded the limit of 0.26 g of the general license. The licensee issued PIP C-04-06877 for the cause and resolution. The licensee has contacted the vendor and the NRC Office of Nuclear Material Safety and Security to resolve the problem. Pending the licensee resolution of this seismic issue, this item is identified as a Unresolved Item (URI) 72-45/2005-001-01, ISFSI Concrete Storage Pad Calculated Seismic Acceleration Exceeds the General License Limit.

ff. Findings

No findings of significance were identified. One unresolved item was identified involving a calculated seismic acceleration which exceeds the general license limit.

4OA6 Meetings

Exit Meeting Summary

On July 7, 2005, the resident inspectors presented the inspection results to Mr. L. Keller, Regulatory Compliance Manager, and other members of licensee management, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violation

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

- TS 5.5.8 requires that an In-Service Test (IST) Program shall be established, implemented and maintained for American Society of Mechanical Engineers (ASME) Code Class 1, 2 and 3 components. The station's IST Program identified those components that required testing and the frequency at which they are to be tested. The nuclear service water system-to-containment penetration valve injection system supply check valves were within the scope of the IST Program and were required to be tested on a quarterly basis. Contrary to this, on April 20, 2005, the licensee discovered that the work orders used to schedule the required testing had been inadvertently deleted and the tests had not been performed within the previous 115 days (the specified frequency plus the allowable grace period). This event was documented in the licensee's corrective action program as PIP C-05-2089. This finding is of very low safety significance because the valves were subsequently tested successfully, which ensured that nuclear service water would have been available to supply makeup to the containment penetration valve injection system, if required to do so.

DEC

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Adams, Human Performance Manager
E. Beadle, Emergency Planning Manager
D. Bryan, FANP (Framatone) Civil Engineer
W. Byers, Security Manager
B. Calloway, Allow 600, Boric Acid
C. Cauthen, Steam Generator Maintenance Engineer
T. Daniels, Emergency Planning/Fire Protection
J. Foster, Radiation Protection Manager
R. Glover, Station Manager
W. Green, Reactor and Electrical Systems Manager
G. Hamrick, Mechanical, Civil Engineering Manager
T. Hawkins, ISI Coordinator
D. Jamil, Catawba Site Vice President
L. Keller, Regulatory Compliance Manager
A. Lindsay, Training Manager
S. Magee, Public Relations
G. Mitchell, Emergency Planning
M. Patrick, Work Control Superintendent
J. Pitesa, Operations Superintendent
T. Ray, Safety Assurance Manager
J. Reeves, FANP Project Manager for Independent Spent Fuel Storage Installation
R. Repko, Engineering Manager
F. Smith, Chemistry Manager
R. Smith, Emergency Planning
G. Strickland, Regulatory Compliance Specialist
C. Trezise, Maintenance Superintendent

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

72-45/2005-001-01	URI	ISFSI Concrete Storage Pad Calculated Seismic Acceleration Exceeds the General License Limit

Opened and Closed

05000413/2005003-01	NCV	Inadequate Post Maintenance Testing on 1RN-38B, 1B RN Pump Discharge Valve (Section 1R15b.1)

Attachment

DEC		A-2	
05000413,414/2005003-02	NCV		Failure to Adequately Evaluate Potential RHR System Differential Pressure During Postulated Accident Conditions In Generic Letter 89-10 MOV Testing Program (Section 1R15b.2)
Closed			
2515/163	TI		Operational Readiness Of Offsite Power- Unit 1 and Unit 2 (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Warm Weather Preparation

PT/0/B/4700/039, Return from Cold Weather Protection Alignment
PT/0/B/4700/038, Cold Weather Protection
PIP C-05-3124, Additional guidance is needed in PT/0/B/4700/039 to prepare the station for hot weather conditions
Engineering's proposed additions to PT/0/B/4700/039, Return from Cold Weather Protection Alignment
Action Register Updates for Hot Weather Preparations/Protection
Hot weather protection program routine preventive maintenance work orders associated with safety-related and significant plant systems

PIPs generated as a result of this inspection
PIP C-05-3736, Items identified by NRC during hot weather protection inspection

Section 1R04: Equipment Alignment

OP/0/A/6400/006C, Nuclear Service Water System
Catawba Operations Training Lesson Plan OP-CN-PSS-RN, Nuclear Service Water
Nuclear Service Water System Health Report, 3rd Trimester, 2004
Maintenance Rule SSC Summary Sheets for Nuclear Service Water
Flow Diagrams for Nuclear Service Water System: CN-1574-1.0, 1.1, 1.4, 1.5, 2.0, 2.2, 2.3, 2.8
Flow Diagrams for Nuclear Service Water System: CN-2574-2.0, 2.1, 2.2, 2.3, 2.4, 2.5, 2.6, 2.7
TS 3.7.8 (Nuclear Service Water system) and 3.7.9 (Standby Nuclear Service Water Pond)
OP/1/A/6350/002, Diesel Generator Operation

PIPs generated as a result of this inspection
PIP C-05-3155, Numerous housekeeping and scaffolding issues identified in the 1A diesel generator room while it was protected

Section 1R05: Fire Protection

Pre-Fire Plan for Area AY, Transformer Yard, Unit 1

Pre-Fire Plan for Area AZ, Transformer Yard, Unit 2
Pre-Fire Plan for Area 42, Diesel Generator Building 1B Corridor
Pre-Fire Plan for Area 26, Diesel Generator Building Room 1B
Pre-Fire Plan for Area 47, Unit 2 Spent Fuel Pool Purge Unit Area
Pre-Fire Plan for Area 4, Auxiliary Building, 543 foot elevation
Pre-Fire Plan for Area 11, Auxiliary Building, 560 foot elevation
Pre-Fire Plan for Area 5, Unit 2 Electrical Penetration Room, 560 foot elevation
Pre-Fire Plan for Area 7, Unit 2 Essential Switchgear, 560 foot elevation

PIPs generated as a result of this inspection
C-05-2054, Phone number shown on phone in the Unit 2 switchgear room, 560', is incorrect

Section 1R08: Inservice Inspection Activities

Nondestructive Examination Procedures
NDE 600, Revision 16, Ultrasonic examination of similar metal welds in ferritic and austenitic piping
NDE 25, Revision 20, Liquid penetrant examination
QAL-13, Revision 19, Inservice Inspection Visual Examination,VT-1, VT1C and VT1MC
QAL-14, Revision 25, Inservice Inspection Visual Examination,VT-3, VT3C and VT3MC

Steam Generator
Eddy Current Analysis Guidelines for Duke Power Company's CFR80 Steam Generators, Rev. 7
SGMEP 104, Condition Monitoring, Rev. 4
Eddy Current Acquisition Guidelines for Duke Power Company's CFR80 Steam Generators, Rev. 12
CFR80 Steam Generator Site Technique Validation for Catawba Nuclear Station Unit 1 and McGuire Nuclear Station Units 1 & 2, Rev. 6
MP/0/A/7150/009A Steam Generator Tube Surveillance and Plugging, Rev. 5
SGMEP 104, Condition Monitoring, Rev. 4
SGMEP 105, CFR80 Specific Assessment of Potential Degradation Mechanisms, Rev. 5
Eddy Current Examination Plan, Rev. 2
CFR-80 Steam Generator Tube Integrity Assessment, Catawba Unit 1, 12/4/01
PIP C-05-00291, Track implementation of Generic License Change Package (GCLP), revising Technical Specifications
PIP C-04-04271, Information on the implementation of NRC IN 2004-17, Loose Part Detection and Computerized Eddy Current Data Analysis in Steam Generators
PIP C-04-03664, Notification from Westinghouse that their automated eddy current analysis software missed a short segment of tubing during 2EOC12 steam generator tube inspection
PIP-C-04-05224, SG secondary side inspection scope expansion die to discovery of potential feedwater waterbox modification made during SG manufacturing process
PIP-C-05-01974, Recent Duke and industry experience with loose parts in steam generators warrants evaluation of design change for the CA and CF systems for strainers
PIP-C-04-04941, Eddy current indications found during 2EOC13 steam generator examinations
DPC response to NRC Generic Letter 2004-01, Requirements for Steam Generator Tube Inspections, 10/28/04

Self Assessments
NPA Assessment GO-04-49(NPA)(Out Act)(CNS), Refueling activities, Welding and ISI Outage activities
QATS/ISIM Self-Assessment Planning Form, Catawba Unit 1 Piping Wels- Reactor Coolant (NC) System
Duke Power Company Assessment Report, Fluid Leak Management Program

Section 1R12: Routine Maintenance Effectiveness

Unit Containment Air Return Fan Damper 2ARF-D-4 Troubleshooting Plan
TS 3.6, Containment Systems
Design Basis Specification, Containment Air Return & Hydrogen Skimmer System (VX), CNS-1557.VX-00-001
PIP C-00–6455, VX System agastat relays are approaching end of life
PIP C-03-5722, Review of numerous VX Agastat timer out of tolerance conditions
1A diesel generator troubleshooting plan for excitation circuitry
PIP C-05-1926, Unplanned entry into T.S. 3.8.1 due to 1A DG output breaker tripping

PIPs generated as a result of this inspection
C-05-1921, Recurring problems noted with housekeeping in the diesel generator rooms

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Complex Evolution Plan CN-11447, D/G 1A Pre-Outage Battery Work
Planned maintenance schedule for work weeks 05W14 and 05W15
PIP C-05-1926, Unplanned entry into T.S. 3.8.1 due to 1A DG output breaker tripping
PIP C-05-2591; 1B RN Pump Discharge Valve did not close when pump was secured

PIPs generated as a result of this inspection
C-05-1051, Less than adequate actions taken following ORAM Sentinel identifying the overall plant risk as being RED when combining 1A D/G unavailability with severe adverse weather conditions

Section 1R14: Personnel Performance During Nonroutine Plant Evolutions

OP/1/A/6100/003, Controlling Procedure for Unit Operation, Enclosure 4.2, Power Decrease
Unit 2 Plant Unified Operational Log for the period of May 11-12, 2005
PIP C-04-6109, Large LH leak on CIV #4 required a rapid downpower and removal of the Unit 2 main turbine (11/9/04)
OP/2/B/6100/010E, Annunciator response procedure for panel 2AD-4; window B/5
AP/2/A/5500/009, Rapid Downpower
Selected OAC screen prints associated with the Unit 2 rapid downpower following the #5 CIV Electro-Hydraulic Control fluid leak
PT/0/A/4150/001J, Zero Power Physics Testing
PT/0/A/4150/001J, Zero Power Physics Testing tailgate briefing
PT/0/A/4150/001, Controlling Procedure for Startup Physics Testing
OP/1/A/6100/001, Controlling Procedure for Unit Startup
AP/1/A/5500/002, Turbine Trip

Section 1R15: Operability Evaluations

OAC Temporary Alarm Limits screens for NC Pump 1A, 1B, 1C, 1D, 2A, 2B, 2C AND 2D Number 1 Seal Leakoff Hi Flow Computer Points
ODMI Form associated with PIP C-04-3363, NC pump 1A lower bearing water temperature RTD
loop step change and associated corrective actions
Westinghouse letter DPC-05-12; Reactor Coolant Pump Seal Water Volumes
PIP C-05-2851, Diesel engine starting air storage tank inlet check valves found to be leaking by during performance of PT/1/A/4200/007 as part of post maintenance testing

Section 1R16: Operator Workarounds

Nuclear System Directive 506, Operator Workarounds
Catawba Nuclear Station Operator Workaround Book

Section 1R19: Post-Maintenance Testing

Work Order (WO) 98717584, Repair 2B NV pump seal and perform IWP
PT/2/A/4200/007B, Centrifugal Charging Pump 2B Test
Complex Evolution Plan for the replacement of 2NVFT5450, Boric Acid Turbine Flow Meter
PT/1/A/4350/002 B, Diesel Generator 1B Operability Test
PT/1/A/4350/002 A, Diesel Generator 1A Operability Test
PT/1/A/4200/007A, Centrifugal Charging Pump 1A Test
PT/1/A/4200/013G, NI Valve In-Service Testing; Enclosures 13.10 and 13.12
PT/1/A/4200/077, VG Valve In-Service Testing
PT/0/A/4450/008E, Control Room Area Chillers Performance Test
WO 98729913, Inspect / repair the 'A' YC chiller tripping

Section 1R20: Refueling and Outage Activities

PIP C-05-3026, Damage was identified on the flange surface of the 1D steam generator hot leg man-way cover
PIP C-05-3147, 'B' train of VC rendered inoperable during fuel movement by placing the 1B sequencer in test
PIP C-05-3082, NC level increased to 10.8% during NC system drain down for nozzle dam replacement due to a valve in the ND system leaking by
Partial restoration tagout ID 05-00538; 6.9 kV beakers/NC pump fuses
OP/0/A/6350/010, Operation of Station Breakers and Disconnects; Enclosure 4.10; 6.9 kV circuit breakers - except tie breakers, and Enclosure 4.18; bus potential transformers
OAC trends for points C1A0854 (NC Loop C wide range level) and C1A0905 (NC Loop A wide range level)
PIP C-05-2639, The Unit 1 Closure Component listing is incomplete

<u>PIP's generated during these inspections:</u>
PIP C-05-3050, Chain fall attached to a support in the ceiling above the 'A' reactor coolant pump and contacting adjacent ventilation ducting
PIP C-05-3178, Improper scaffolding built around the Unit 1 KC heat exchangers

PIP C-05-3197, Numerous issues related to scaffolding throughout Unit 1
PIP C-05-3151, Paint chips, oil, and grease noted on the tip of the polar crane main hook
PIP C-05-3442, NRC inspection of containment following completion of the Operations Containment Cleanliness PT

Section 1R22: Surveillance Testing

PIP's generated during these inspections:
C-05-1862, Several issues identified during the performance of PT/1/A/4200/007C, Standby Makeup Pump #1 Performance Test

Section 1EP6: Drill Evaluation

Emergency notification forms generated during the June 21, 2005 drill
Fire drill mini scenario 05-4 for the June 21, 2005 drill
Catawba Nuclear Station drill 05-04 scenario guide

Section 4OA2: Problem Identification and Resolution

Safety Review Group Monthly Report, April 2005
Safety Review Group Monthly Report, May 2005
PIP C-05-3956, WC 1st Quarter HU Assessment
PIP C-05-3437, RP Self Assessment of Procedure Use and Adherence Second Quarter 2005

Section 4OA5: Other

MP/0/A/7150/042D; Reactor Vessel Head Penetration Visual Inspection
PIP C-02-141; Review of Catawba reactor vessel head inspection practices as a result of PIP M-02-1511 at McGuire
SLC 16.5.8; Reactor Pressure Vessel Head Inspection
Digital photographs of the Unit 1 reactor vessel head prior to and following cleaning during refueling outage 1EOC15
Videotape of the Unit 1 reactor vessel head inspection conducted during refueling outage 1EOC13
PIP C-05-3187; Documentation of the Unit 1 reactor vessel head inspection conducted on May 19, 2005
FANP Document Number 51-5051703, Rev. 0, CNS ISFSI Construction Specification
Areva Calculation 32-5051258-01, Rev. 1, Catawba Nuclear Station Structural Analysis of Pads for ISFSI Storage Modules
Areva Document Change Notice Serial Nos. 4170509-003, Rev. 00, 4170509-002, Rev. 00, and 4170509-005, Rev. 00
Areva Document Identifier 51-5057571-00, Rev. 00, CNS ISFSI Construction Checklist
Memo to file for acceptance of subgrade and rolling
Hall Contracting Corp. Transmittal 013 for Rebar Certified Mill Test Report
Concrete Supply Co. Proposed Mix Design for Hall Construction Catawba Nuclear Station
MACTEC Project No. 6234-05-2778 , Report of Laboratory Testing, Catawba Nuclear Station, ISFSI Project
Areva Contract Variation Approval Request (CVAR) No. 87-5065683-00

Areva Document Comment Form for FANP Document Nos. 38-5066683-00, Report of Field Density Tests and 38-5067353-00, Hall Transmittal o16
NAC International Drawings 061, Sheets 1 to 4, Project 795, Rev. 0, Weldment and Structure, Vertical Concrete Cask (VCC), NAC-UMS
NAC International Drawings 063, Project 795, Rev. 0, Lid, Vertical Concrete Cask (VCC), NAC-UMS
NAC International Drawings 064, Project 795, Rev. 0, Shield Plug, Vertical Concrete Cask (VCC), NAC-UMS
Framatone Drawing No. 5047949E, Sheets 1 to 3, Rev. 2, Duke Power Company Catawba Nuclear Station ISFSI General Site Plan
Framatone Drawing No. 5047950E, Sheets 1 to 2, Rev. 1, Duke Power Company Catawba Nuclear Station ISFSI Finished Grading Plan
Framatone Drawing No. 5047953E, Sheets 1 to 2, Rev. 2, Duke Power Company Catawba Nuclear Station ISFSI Pad Nos. 1 & 2 Plan & Details
Framatone Drawing No. 5049178E, Rev. 1, Duke Power Company Catawba Nuclear Station ISFSI Conduit & Grounding Plan
PIP C-04-06877; Potential Seisonic accelerations associated with the spent fuel, Dry Storage pads may exceed the valve in the general license application.
PIP C-05-03918; NRC exit comments associated with the ISFSI inspection Nonconformance Condition Report 2005-2598

LIST OF ACRONYMS USED

ASME	-	American Society of Mechanical Engineers
BACC	-	Boric Acid Corrosion Control
CAP	-	Corrective Action Program
CFR	-	Code of Federal Regulations
CIV	-	Combined Intermediate Valve
CNS	-	Catawba Nuclear Station
DG	-	Diesel Generator
ECCS	-	Emergency Core Cooling System
EOC	-	End of Cycle
ET	-	Eddy Current Testing
FWST	-	Refueling Water Storage Tank
GLCP	-	Generic License Change Package
ISFSI	-	Independent Spent Fuel Storage Installation
ISI	-	Inservice Inspection
IST	-	In-Service Testing
LOCA	-	Loss of Coolant Accident
KC	-	Component Cooling
MOV	-	Motor Operated Valve
NCV	-	Non-Cited Violation
ND	-	Residual Heat Removal
NI	-	Safety Injection
NRC	-	Nuclear Regulatory Commission
NRMCA	-	National Ready Mixed Concrete Association
NS	-	Containment Spray
NSD	-	Nuclear System Directive

NV	-	Chemical and Volume Control
PI	-	Performance Indicator
PIP	-	Problem Investigation Process (report)
PSID	-	Pounds Per Square Inch Differential
QC	-	Quality Control
RHR	-	Residual Heat Removal
RN	-	Nuclear Service Water
RTP	-	Rated Thermal Power
SDP	-	Significance Determination Procedure
SG	-	Steam Generator
TI	-	Temporary Instruction
TS	-	Technical Specification
VCT	-	Volume Control Tank
WO	-	Work Order
YC	-	Control Area Chilled Water