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

January 27, 2005

EA-05-019
EA-05-020

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 05000413/2004006 AND 05000414/2004006

Dear Mr. Jamil:

On December 31, 2004, 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 January 06, 2005, with you and 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 one NRC-identified finding and one self-revealing finding of very low safety significance (Green). Both findings were determined to involve violations of NRC requirements; however, because of the 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. While two other violations of NRC requirements were also identified, we have concluded that Catawba's actions did not contribute to the degraded conditions; therefore, no performance deficiencies were identified. Based on these facts, I have been authorized, after consultation with the Office of Enforcement, to exercise enforcement discretion in accordance with section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for these two violations. An evaluation was performed and we have determined that both issues were of very low safety significance. 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, DC, 20555-0001; with copies the Regional Administrator Region II; Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC, 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/

Victor M. McCree, Director
Division of Reactor Projects

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

Enclosure: Integrated Inspection Report 05000413/2004006 and 05000414/2004006
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-413, 50-414
License Nos:	NPF-35, NPF-52
Report No:	05000413/2004006, 05000414/2004006
Licensee:	Duke Energy Corporation
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	4800 Concord Road York, SC 29745
Dates:	September 19, 2004 - December 31, 2004
Inspectors:	E. Guthrie, Senior Resident Inspector A. Sabisch, Resident Inspector D. Jones, Resident Inspector - Robinson (Sections 1R13, 1R19, and 1R20) R. Chou, Reactor Inspector (Section 1R08) J. Lenahan, Reactor Inspector (Sections 1R08 and 4OA5.6) F. Wright, Senior Health Physics Inspector (Section 2OS2) R. Hamilton, Health Physics Inspector (Section 2PS2) J. Kreh, Health Physics Inspector (Sections 2OS1 and 4OA1)
Approved by:	Michael E. Ernstes, Chief Reactor Projects Branch 1 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000413/2004006, IR 05000414/2004006; 9/19/2004 - 12/31/2004; Catawba Nuclear Station, Units 1 and 2; Event Followup and Other Activities.

The report covered a three month period of inspection by three resident inspectors (one visiting) and five region based inspectors; three health physics inspectors and two reactor inspectors. Two Green non-cited violations (NCV) 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. A self-revealing non-cited violation was identified for gas intrusion that resulted in a failure to maintain the 1A and 1B centrifugal charging pumps and 1A safety injection pump in an operable condition, in accordance with Technical Specification 3.5.2, Emergency Core Cooling Systems (ECCS). The licensee had several opportunities to evaluate industry events (some having elements identical to this Catawba gas intrusion event) to address the pressurizer as a gas source and evaluate system integration that could lead to inoperability of ECCS equipment.

  This finding was greater than minor because it affected an objective and attribute of the Reactor Safety Mitigating Systems Cornerstone, in that gas accumulation in the centrifugal charging pump suction piping rendered ECCS systems unavailable and unreliable. Due to the short exposure time and the assumption that the 1A safety injection pump was only affected during high pressure recirculation, the finding was determined to be of very low safety significance. (Section 4OA3.1)

- Green. The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion X, Inspection, because inadequate quality control (QC) inspections were performed in Unit 2 on the 2A containment sump. Specifically, containment sump screen gaps, which were intended to be closed via repair activities, were not discovered by QC inspection following the repairs. The gap would allow a containment sump bypass flow path for debris to affect downstream emergency core cooling system (ECCS) components during containment recirculation.

  This finding was greater than minor because it affected an objective and attribute of the Reactor Safety Mitigating Systems Cornerstone, in that inadequate QC inspection failed to identify containment sump bypass flow paths for debris to affect the availability and reliability of ECCS components during containment recirculation. The finding was evaluated using the phase 1 SDP analysis and was determined to be of very low safety significance based on the small size of the gaps and the low probability that material could bypass the sump screen in that area. (Section 4OA5.1)

B. Licensee-identified Violations

None

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the inspection period operating at 100 percent rated thermal power (RTP). Power was reduced to 8 percent and the turbine generator removed from service on November 12, 2004, to conduct troubleshooting and repair on the electro-hydraulic control (EHC) system. The unit returned to 100 percent power on November 14, 2004. On December 5, 2004, a turbine trip/reactor trip occurred when the 1B moisture separator reheater (MSR) level switch logic actuated on a sensed high water level caused by out-of-adjustment microswitches in the level switch circuitry. The unit returned to 100 percent RTP on December 7, 2004, and remained there through the end of the inspection period.

Unit 2 began the inspection period off-line in its end-of-cycle 13 (2EOC13) refueling outage. The unit was returned to service on October 24, 2004. On October 28, 2004, the reactor was manually tripped when control rod shutdown bank D dropped into the core due to a component failure in the control rod logic cabinet. The unit returned to 100 percent RTP on October 31, 2004. On November 9, 2004, an EHC leak was identified on the number 1 combined intermediate valve, which required the turbine be taken off-line in order to facilitate repairs. The reactor was maintained at approximately 10 percent RTP. The unit returned to 100 percent RTP on November 10, 2004, and remained there through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Cold Weather Preparation

a. Inspection Scope

The inspectors reviewed the licensee's preparations for adverse weather associated with cold ambient temperatures. 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. Risk significant systems reviewed included the standby shutdown facility and the refueling water storage tanks. 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.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

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, and 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. The inspectors verified the following three partial system alignments and reviewed the associated listed documents:

- Unit 1 / 2 ‘B’ train of nuclear service water (RN) with the Unit 1 / 2 ‘A’ train of RN out of service for planned maintenance (Technical Specification Action Item Log (TSAIL) entries C0-04-02420 and C0-04-02443 and work orders (WOs) 9867328, 98641940, and 98641672)

- 2B diesel generator (DG), 2B component cooling water (KC) pumps, and 2B 4.16 kV switchgear with the 2A DG inoperable for maintenance and modification activities (TSAIL entry C2-04-02404, Tagout Removal and Restoration 04-00507, 04-00936, 04-02450, 04-02480)

- 2B train of residual heat removal (ND) with the 2A pump out of service for unplanned maintenance and repair activities (CNS 3.1.30, Section 4.8, WOs 98694637 and 98695770, TSAIL entry C2-04-02384, Tagout Removal and Restoration 04-02547, Problem Investigation Process report (PIP) C-04-5361)

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, based on 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:

- main control room
- Unit 1 mechanical penetration room, 543 foot elevation
- Unit 1 mechanical penetration room, 560 foot elevation
- RN pump structure
- Unit 1 electrical penetration room, 560 foot elevation
- Unit 1 electrical penetration room, 577 foot elevation
- Unit 2, train B auxiliary shutdown panel
- Unit 2, B diesel generator room

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

.1 Inservice Inspection (ISI) - Unit 2

a. Inspection Scope

The inspectors observed in-process ISI work activities, reviewed ISI procedures, and reviewed selected ISI records associated with risk significant structures, systems, and components. The observations and records were compared to the requirements specified in the Technical Specifications (TS) and the ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition with No Addenda to verify compliance. Documents reviewed during this inspection are listed in the Attachment of this report. Portions of the following Unit 2 ISI examinations were observed:

- Liquid Penetrant (LPT) examination of weld numbers 2NV257-1, -5, -10, -11, -16, and 2NV275-1; the LPT examination for retest of weld number 2NV275-1 after removal of surface flaw (non-relevant indication - potential flaw) identified on 9/28/04; and corresponding ultrasonic (UT) thickness exams performed during preparation of the weld for the retest. These welds were socket welds on the chemical and volume control system piping.

- Magnetic Particle (MT) examination of weld numbers 2-SM-6D-A, 2SM-46-01 and 2SM-46-02, and welded attachments at support numbers 2-R-SM-1584 and -1585, on the 34 inch diameter main steam piping inboard of the main steam isolation valves.

The records were compared to the TS, License Amendments, and applicable industry established performance criteria to verify compliance. Qualification and certification records for examiners, equipment and procedures for the above examination activities were reviewed. In addition, the inspectors examined snubbers, spring cans and pipe supports during a walkdown of the Unit 2 containment.

The inspectors also reviewed UT data, procedures, and examiner qualifications for the reactor vessel shell welds. The evaluation of data was compared to the acceptance criteria stated in Westinghouse Procedure PDI-ISI-254, Remote Inservice Examination of Reactor Vessel Shell Welds, Revision 5.

The inspectors reviewed activities, plans, and procedures for the inspection and evaluation of the steam generator Inconel Alloy 600TT tubing. Data gathering and evaluation activities were reviewed, with special emphasis on evaluation of the eddy current data for indications in tube R4C61 and expanded samples in tubes inside of the tube sheet area and tube ends near or above seal weld in steam generator 2B.

The inspectors reviewed information and data taken by eddy current examination from the inside of the reactor bottom head to the bottom mounted instrument penetration nozzles. The data was evaluated and compared to the acceptance criteria stated in Westinghouse Procedure WDI-STD-146, Eddy Current Examination of Reactor Vessel Pipe Weld Inside Surface, Revision 3.

The inspectors reviewed implementation of the licensee’s boric acid corrosion control program to determine if commitments made in response to Generic Letter 88-05 and Bulletin 2002-01 were being effectively implemented. The inspectors reviewed licensee procedures that are performed before and after outages to identify boric acid leakage onto various components, evaluate the cause of the leakage, and evaluate the effects of leakage on the components. The inspectors also reviewed records documenting boric acid leaks, work orders and PIPs specifying corrective actions, and examined completed corrective actions during a walkdown of the Unit 2 containment building.

The inspectors reviewed the licensee’s evaluation and acceptance for a one inch long linear indication in the reactor coolant loop hot leg to steam generator C inlet nozzle connection identified in the current outage (2EOC13), documented in PIP C-04-05421. Documents reviewed included the licensee’s calculation and the Westinghouse flaw evaluation handbook for the Catawba Unit 2 steam generator nozzle weld regions. The licensee’s inservice inspection report for the previous Unit 2 refueling outage (2EOC12) was also reviewed for recordable indications.

The inspectors reviewed welding repair records for this outage on Elective Minor Modification CNCE-61943 (Work Order 98504670 and PIP C-01-04283) involving the welding of a Code Class 2 component, Steam Generator 2A Bowl Drain Plug.

A sample of ISI issues in the licensee’s corrective action program were reviewed to confirm that problems were being identified and placed in the corrective action program, and appropriate corrective actions were being initiated. The inspectors reviewed the licensee’s evaluation and acceptance for a non-relevant indication on Code Class 2 piping as documented in PIP C-04-05051. PIP C-04-05051 was the resolution of a potential flaw (i.e., linear surface indication) on piping weld 2NV-275-1, that was identified by a LPT examination, removed by the grinding, and retested. The inspectors also reviewed PIP C-04-05257, Resolution of Linear Surface Indication Discovered by a MT Examination on Main Steam Piping, and PIP C-04-04941, Evaluation, Correction, and Expanded Samples for a Tube Defect Found on the Steam Generator Tube Inspection.

b. Findings

No findings of significance were identified.

.2 Containment Vessel Inspection - Unit 2

a. Inspection Scope

The inspectors examined interior portions of the Unit 2 steel containment vessel (SCV) and reviewed selected records. The observations and records were compared to the TS, ASME Boiler and Pressure Vessel Code, Article IWE of Section XI, 1992 Edition and 1992 Addenda, and 10 CFR 50.55a. The inspectors examined the accessible interior surfaces of the SCV in the pipe chase area and in accumulator rooms A, B, and D.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed a simulator exam conducted on November 16, 2004, to assess the performance of licensed operators. The scenario, Active Simulator Exam, Operating Procedure (OP)-CN-ASE-33, involved the failure of an automatic reactor trip to occur, a small steam generator tube leak, and a steam generator tube rupture. 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. The inspectors reviewed one year of past crew evaluation summaries in an effort to compare and evaluate current operator performance. 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 three items listed below. Documents reviewed are listed in the Attachment to this report.

- 2A ND pump leak through the cover closure stud threads
- Adjustment of 2CA-048 (Auxiliary Feedwater (CA) pump turbine number 2 discharge flow control valve to 2C steam generator) to meet flow requirements at the end of 2EOC13
- Adjustment of limit switches on 2CA-060 (2A CA motor driven pump discharge flow control valve to 2A steam generator) to meet flow requirements at the end of 2EOC13

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 five 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).

- Unit 1 pressurizer level control system card replacement activities
- Planned maintenance on the 'A' train of RN
- Troubleshooting activities on Unit 1 EHC system cabinet
- Station maintenance and testing activities scheduled for the period when the grid status was in an elevated risk condition (Orange) due to three (3) nuclear units being off-line following the Catawba Unit 2 reactor trip of October 28, 2004.
- Station maintenance and testing activities scheduled for the period when the grid status was Orange due to three (3) nuclear units being off-line following the removal of Catawba Unit 2 from service due to an EHC leak on November 9, 2004

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope

On October 1, 2004, the inspectors observed operator performance following the opening of the Unit 2, 2TA electrical tie breaker, which occurred when the trip pushbutton for the breaker was inadvertently depressed during main control board overlay installation. The inspectors observed licensed operators' use of procedures, equipment manipulations during the transient, and subsequent restoration of the normal electrical alignment. Documents reviewed are listed in the Attachment to this report.

On October 24, 2004, the inspectors observed operator performance during the start-up of Unit 2 following the 2EOC13 refueling outage. The inspectors observed licensed operators' use of procedures, control room pre-evolution briefings, and plant equipment manipulations during the approach to criticality, performance of portions of zero power and startup physics testing and portions of power escalation. Documents reviewed are listed in the Attachment to this report.

On October 28, 2004, the inspectors verified adequate operator performance in response to a manual reactor trip on Unit 2 following the unexpected drop of control rod shutdown bank D from 100 percent RTP power. 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 trip to verify the plant responded as expected.

On October 31, 2004, the inspectors observed operator performance during the start-up of Unit 2 following the repair of the control rod drive system that had resulted in a reactor trip on October 28, 2004. The inspectors observed and verified licensed operators' use of procedures, control room pre-evolution briefings, and plant equipment manipulations during the approach to criticality. Documents reviewed are listed in the Attachment to this report.

On November 9-10, 2004, the inspectors reviewed the power reduction to 46 percent RTP (after an EHC leak was identified on Unit 2) followed by a manual turbine trip, and observed stabilization of the reactor at approximately 10 percent RTP while repairs were facilitated. The inspectors verified operator actions and use of procedures in responding to the EHC leak. In addition, the inspectors reviewed selected trend graphs for parameters following the turbine trip to verify the plant responded as expected.

On November 12, 2004, the inspectors observed operator performance during portions of the power reduction from 100 percent to 8 percent RTP on Unit 1 to repair problems with the EHC control system. The inspectors observed licensed operators' use of procedures, control room pre-evolution briefings, and plant equipment manipulations during the down power evolution. Documents reviewed are listed in the Attachment to this report.

On December 5, 2004, the inspectors verified adequate operator performance in response to an automatic reactor trip on Unit 1 from 100 percent RTP power following a turbine trip resulting from a MSR high water level turbine trip signal caused by out-of-adjustment microswitches in the level switch circuitry. The inspectors observed operator actions and use of procedures in stabilizing the unit. In addition, the inspectors reviewed selected trend graphs for parameters following the trip to verify the plant responded as expected.

On December 7, 2004, the inspectors observed operator performance during portions of the start-up of Unit 1 following repair and adjustment of the microswitches associated with the MSR limit switches. The inspectors observed licensed operators' use of procedures, control room pre-evolution briefings, and plant equipment manipulations during the approach to criticality and portions of the power escalation. 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 ten issues listed below. Documents reviewed are listed in the Attachment to this report.

- Testing of containment penetration valve injection (NW) solenoid valves not performed as part of NW system surveillance (PIP C-04-5206)
- Unit 2 CA flow control valves experienced internal binding due to worn piston seals (PIP C-04-5372)
- Control Room differential pressure decreased below the TS minimum value of 1/8-inch water gauge during A train engineered safeguards feature (ESF) testing (PIP C-04-5549)
- Steam generator secondary side inspection scope expansion due to the discovery of potential feedwater waterbox modification made during the initial manufacturing process (PIP C-04-5224)
- Unit 1 A diesel generator starting air system pressure regulator found to be leaking (PIP C-04-5870)
- Back plate glastic barriers, used to maintain electrical separation, found unsecured (PIP C-04-5486)
- 2B safety injection (NI) pump outboard seal leak (PIP C-04-05410)
- Unit 2 ice condenser blast shield bolting was torqued, which was not in accordance with the Westinghouse design and installation specifications (PIP C-04-5978)
- Potential diversion flow paths to the incore instrumentation rooms (PIP C-04-05738)
- Back leakage through check valves in the 1A CA pump header may have resulted in pressure/temperature conditions where steam formation and voiding could have occurred (PIP C-04-6249)

b. Findings

No findings of significance were identified.

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.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the following permanent plant modification to verify the adequacy of the modification package, and to verify that the design change and subsequent post-modification testing ensured continued reliability and satisfactory performance of the affected emergency diesel generator.

The following plant modification and associated attribute was reviewed:

- CNCE-61427, Diesel Generator 2A Turbocharger Lube Oil Tubing (Mitigating Systems)

  Documents Reviewed:
  - MP/0/A/7400/013, Diesel engine break-in after major maintenance
  - PT/2/A/4350/02A, Diesel generator 2A operability test
  - Drawing CN-2609-2.0, Diesel generator lube oil system flow diagram
  - CNS-1609.LD-00-0001, Diesel generator lube oil system design basis specification

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. The eight tests reviewed are listed below. Documents reviewed are listed in the Attachment to this report.

- Post maintenance testing on 2RN-351 (KC heat exchanger 2B Outlet Throttle Valve)
- 2A RN pump performance test following pump replacement
- Replacement of original construction-vintage circuit board in Unit 1 protection cabinet for pressurizer pressure channel 3
- Replacement of Unit 1 Train A CA pump “loss of suction pressure” pressure transmitter
- Repair of 2CA-56 (Unit 2 CA pump 2A discharge flow control valve to the 2B steam generator)
- Adjustment of 2CA-048 (CA pump turbine number 2 discharge flow control valve to 2C steam generator) to meet flow requirements at the end of 2EOC13
- Adjustment of limit switches on 2CA-060 (2A CA motor driven pump discharge flow control valve to 2A steam generator) to meet flow requirements at the end of 2EOC13
- B control room area chiller performance test following divider plate replacement and chiller cleaning.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors evaluated Unit 2 outage activities to ensure that the licensee: considered risk in developing 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 TS requirements that ensure defense-in-depth. The following specific areas were reviewed:

- Outage Plan - Prior to the outage, the inspectors reviewed the licensee’s outage risk control plan, attended risk briefings, 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 portion of the outage in this inspection period. The inspectors reviewed the Unit 2 outage risk assessment CN-04-049, 2EOC-13-IRT (Independent Review Team) 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 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-04-049, 2EOC-13-IRT Pre-Outage Review, Shutdown Risk Assessment. The inspectors also reviewed the implementation of PT/2/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. Specifically, the inspectors reviewed the tagout for isolating and draining several components in the Unit 2 ND system (Tagout ID 04-01826).

- 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/2/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 the conduct of 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, 2EOC-13-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/2/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 OP/2/A/6150/001, Filling and Venting the Reactor Coolant System, Enclosure 4.16, Reactor Coolant System Vacuum Refill Without Solid Operation, and 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 10 CFR 50.65(a)(4). Potential reactivity changes were identified in the outage risk plan CN-04-049 (2EOC-13-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 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) per10 CFR 50.65(a)(4)
  - PT/2/A/4200/002C, Containment Closure Verification (Part I)
  - PT/2/A/4200/002I, Containment Closure Verification (Part II)
  - PT/2/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 fuel handling procedures. Specifically, the inspectors observed 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/2/A/6550/006, Transferring Fuel with the Spent Fuel Manipulator Crane
  - OP/2/A/6550/007, Reactor Building Manipulator Crane Operation

- OP/2/A/6550/008, Fuel Transfer System Operation
- MP/0/B/7150/012, Refueling Canal Cleanliness
- PT/2/A/4550/001C, Refueling Communications Test
- PT/0/A/4150/022, Total Core Reloading (including 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 Foreign Material Exclusion (FME) Inspection
  - PT/0/A/4150/001J, Zero Power Physics Testing
  - PT/0/A/4150/001, Controlling Procedure for Startup Physics Testing (including tailgate briefing)
  - OP/2/A/6100/001, Controlling Procedure for Unit Startup
  - OP/2/A/6100/003, Controlling Procedure for Unit Operations
  - OP/2/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 seven activities were reviewed:

Surveillance Tests:

- IP/2/A/3200/008B, Unit 2 Train B Reactor Trip Breaker Trip Actuating Device Functional and Operational Test
- IP/2/A/3200/002B, Unit 2 Solid State Protection System Train B Periodic Testing

In-Service Tests:

- PT/2/A/4200/010A, Residual Heat Removal Pump 2A Performance Test

Containment Isolation Valve Test:

- PT/2/A/4200/001 I, Enclosure 3.18, Unit 2 Penetration Number M327 As Found Type C Leak Rate Test

Ice Condenser Surveillance Tests:

- SM/0/A/8510/001, Inspection of Ice Condenser Flow Passages, Unit 2 - Bay 5 and Bay 11
- MP/0/A/7150/006, Ice Condenser Lower Inlet Doors Inspection and Testing, As-Found Test
- MP/0/A/7150/006, Ice Condenser Lower Inlet Doors Inspection and Testing, As-Left Test

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 a drill conducted on December 7, 2004. 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 accuracy of classification, the notification and protective action recommendations (PAR) development activities. The performance of the emergency response organization was evaluated against applicable licensee procedures and regulatory requirements. The inspectors attended the post-exercise critique to evaluate the licensee's self-assessment process for identifying deficiencies relating to failures in classification and notification, as well as PAR development activities. The inspectors assessed the drill for weaknesses and deficiencies in performance of classification and notification requirements. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety (OS) and Public Radiation Safety (PS)

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

Access Control: The inspectors evaluated licensee activities for monitoring and controlling worker access to radiologically significant areas during the Unit 2 refueling outage. The inspection included direct observation of administrative and physical controls, appraisal of the knowledge and proficiency of radiation workers and health physics technicians (HPTs) in implementing radiological controls, and review of the adequacy of procedural guidance and its implementation.

The inspectors observed implementation of radiological controls for selected Radiation Areas (RAs), High Radiation Areas (HRAs), Extra High Radiation Areas (EHRAs), and Very High Radiation Areas (VHRAs) within the Radiologically Controlled Area (RCA) of the plant. Postings and labels (including those applicable to any radioactive materials) at these locations were evaluated for consistency with procedural guidance and compliance with regulations. The inspectors directly observed the posting and locking status of selected HRAs (approximately ten) in the Units 1 and 2 Auxiliary Buildings. Independent dose-rate measurements were taken in five rooms in the Auxiliary Buildings, and the results of those measurements were compared to current licensee surveys to determine whether licensee surveys were complete and accurate.

The inspectors attended a pre-job briefing for an EHRA entry (Room 308A Pipe Trench in Unit 2 Auxiliary Building), evaluated the use of radiological controls, observed the performance of HPTs and radiation workers in several work activities (through direct field or closed-circuit-video observations of work activities), and evaluated Radiation Work Permit (RWP) requirements and electronic dosimeter (ED) alarm setpoints. During general observations of outage work, the inspectors queried radiation workers on RWP requirements associated with their tasks in progress.

The inspectors reviewed administrative guidance documents and procedures for control of material stored in spent fuel pools, posting of areas, access controls to EHRAs and VHRAs, surveys of areas, and RWP use. The inspectors reviewed selected RWPs and surveys to evaluate the adequacy of radiological controls for RAs, HRAs, and EHRAs. Records of internal dose assessments performed during the past year were reviewed and discussed with cognizant personnel. Health Physics management and supervisory personnel were interviewed regarding administrative control of EHRA and VHRA keys, as well as any changes in the past year to procedural guidance for access control. Particular focus was placed on review of the licensee’s controls for the In-Core Sump Rooms, which have the potential to become VHRAs during certain outage operations. The licensee’s use of MURUROA Model V4 FI and V4 MTH2 equipment (disposable supplied-air bubble suits with a self-rescue feature which obviated the need for dedicated standby rescue personnel) was observed in the plant and discussed with Health Physics management, including training and storage considerations. The inspectors evaluated the licensee’s fulfillment of commitments regarding the use of these suits as discussed in the NRC’s Safety Evaluation.

Radiation protection program activities and their implementation were evaluated against 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; Updated Final Safety Analysis Report (UFSAR) Section 12, Radiation Protection Program; and licensee commitments and approved procedures. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment to this report.

Problem Identification and Resolution: Licensee PIPs related to access control, HPT performance, and radiation worker performance were reviewed and discussed with cognizant licensee personnel. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with their procedures. Specific PIPs and other documents that were reviewed and evaluated in detail for this program area are identified in the Attachment to this report.

b. Findings

No findings of significance were identified.

2OS2 As Low As Reasonably Achievable (ALARA) Planning and Controls

a. Inspection Scope

ALARA: The inspectors evaluated ALARA program guidance and its implementation for ongoing Refueling Outage tasks. The inspectors reviewed and discussed with the licensee's staff, ALARA work plan documents (including dose estimates and prescribed ALARA controls) for selected outage work activities expected to incur significant collective doses. 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 closed-circuit-video observations of Unit 2 work activities involving: the core barrel lift; preparations of radioactive shipments; steam generator maintenance including installation of robot and eddy current testing equipment; modifications and work in both containment spray heat exchanger rooms and lower containment; mockup training for installing steam generator nozzle radiography source centering equipment; and on-going work in auxiliary building. The inspectors interviewed radiation workers and HPT 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. 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. Cobalt reduction initiatives for reactor coolant system (RCS) valve replacement activities were reviewed and discussed in detail.

The plant collective exposure histories for calendar years (CY) 2001, 2002 and 2003, taken from data reported to the NRC pursuant to 10 CFR 20.2206(c), 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 2002 and 2003 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 (RG) 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable, and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in the Attachment to this report.

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 listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization: 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 2002 and 2003 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. The licensee's 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 (PCP) and UFSAR details.

Licensee radionuclide characterizations of each major waste stream were evaluated. For dry active waste (DAW), primary resin, secondary resin, and filters, the inspectors evaluated PCP and licensee procedural guidance against 10 CFR 61.55 and the Branch Technical Position (BTP) 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, 2003, through September 24, 2004, period. The licensee's analyses and current scaling factors for quantifying hard-to-detect nuclides were assessed. The inspectors discussed the potential for changes in plant operating conditions and reviewed selected DAW waste stream radionuclide data to determine if known plant changes were assessed and radionuclide composition remained consistent for the period reviewed.

Transportation: 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 September 21, 2004, the inspectors observed the loading, bracing, and placarding; and independently reviewed radiation and contamination survey results applicable to transport of surface-contaminated equipment. The inspectors directly observed radiation surveys of the boxes and the transport vehicle being prepared for shipment. On September 23, 2004, the inspectors observed the shipment receipt and survey of two empty sea land containers and the initial package surveys and loading of two sea land containers containing dry radioactive waste. 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.

Problem Identification and Resolution: PIPs 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 listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period from October 2003 through September 2004. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2, were used to verify the basis in reporting for each data element.

- High Pressure Injection System Safety System Unavailability, Unit 1
- High Pressure Injection System Safety System Unavailability, Unit 2
- Residual Heat Removal System Safety System Unavailability, Unit 1
- Residual Heat Removal System Safety System Unavailability, Unit 2

The inspectors reviewed a selection of Licensee Event Reports (LERs), portions of Unit 1 and Unit 2 operator log entries, TSAIL entries, PIP descriptions, monthly operating reports, and PI data sheets to verify that the licensee had adequately identified the number of unavailability hours and safety system functional failures. These numbers were compared to the numbers reported for the PIs.

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety and Public Radiation Safety Cornerstones

a. Inspection Scope

The inspectors sampled licensee submittals relative to the Performance Indicators (PIs) listed below for the period January 1, 2003 through June 30, 2004. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, Revision 2, were used to confirm the reporting basis for each data element.

Occupational Radiation Safety Cornerstone PI

- Occupational Exposure Control Effectiveness

Public Radiation Safety Cornerstone PI

- RETS/ODCM Radiological Effluent Occurrences

For the specified review period, the inspectors evaluated data reported to the NRC, and sampled and assessed applicable corrective action program issues and selected Radiation Protection program records. The inspectors examined in detail the documentation of the licensee's monthly review for PI occurrences as performed for selected months in accordance with procedure SRPMP 10-1. The inspectors also interviewed the licensee personnel who were responsible for collecting and evaluating the PI data. Licensee procedures and records reviewed for PI verification are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

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

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 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 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 June 2004 through December 2004, 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. The specific 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.

b. 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. However, the inspectors continued to observe a trend associated with inadequate procedure use and adherence that the licensee had not previously fully recognized. This trend has continued to be identified based on actual inspector observations of several activities performed by various organizations of licensee personnel. The observations included individuals not performing procedural steps as written, missing notes in procedures which affected the results, not performing inspections as stated in the procedure, and not performing corrective maintenance in accordance with the procedure. The inspectors made the observations throughout the past six months. The observations were discussed with the licensee and several observations were dispositioned as minor violations. For most of the examples noted above, the licensee generated PIPs as a result of the observations and those are listed in the Attachment to this report.

The inspectors performed a review of the PIP documents generated as a result of each of the inspectors' observations, including the description, classification, corrective actions, and issue coding. The inspectors noted that most of the PIP classification coding identified procedure use and adherence as the underlying cause of each event.

The inspectors discussed the basis for their observation, that a trend continued to exist, with the licensee. The inspectors noted that the licensee had reinforced management expectations with the operations organization, for procedure use and adherence, following the previous semi-annual trend review. Reinforcement of expectations was performed via management discussions and counseling. Following continued discussion of observations involving various site organizations during this inspection period, the licensee generated PIP C-04-06590, which identified this issue as a site-wide emerging trend associated with procedure use and adherence. This PIP cited 63 examples of PIPs that were coded as procedure use and adherence issues. The licensee acknowledged that creating a site-wide trend on this issue will require more management involvement to address corrective actions.

.3 Annual Sample Review

a. Inspection Scope

The inspectors selected PIP C-04-0891 for detailed review. It identified a hydraulic actuator failure on the 1B steam generator main feedwater isolation valve (1CF-42), which caused the valve to stroke closed. The document involved a root cause investigation that was reviewed to ensure that 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 CAP, Nuclear System Directive 208, Problem Investigation Process, and 10 CFR 50 Appendix B.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

.1 (Closed) LER 05000413/2004001-01, Gas Accumulation In Centrifugal Charging Pump Suction Piping

a. Inspection Scope

The inspectors reviewed the LER and associated PIP C-04-0008, which documented this event in the CAP. This review verified that the cause of the January 14, 2004, event involving gas accumulation in the suction piping to the Unit 1 centrifugal charging pumps, was identified and that corrective actions were reasonable. Gas originating from the pressurizer gas space, accumulated in the centrifugal charging pump suction piping through a pressurizer sample purge line interface with a centrifugal charging system relief valve located on the emergency boration line. The gas leaked through the charging system relief valve in a reverse direction to the normal flow path of the valve. The inspectors reviewed plant parameters and gas accumulation quantities, as well as verified that notifications were made in accordance with 10 CFR 50.72. The inspectors performed a corrective action review at the time of the event and documented the inspection in NRC Integrated Inspection Report 0500413,414/2004003. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

Introduction: A Green self-revealing non-cited violation (NCV) was identified for gas accumulation that resulted in a failure to maintain the 1A and 1B centrifugal charging pumps and 1A safety injection pump in an operable condition, in accordance with TS 3.5.2, Emergency Core Cooling Systems (ECCS).

Description: The licensee's investigation determined that the cause of the gas accumulation was reverse leakage of gas in the volume control tank relief header through a centrifugal charging system relief valve (1NV-235) located on the emergency boration line. The configuration of the piping for the pressurizer sample purge line connection to the volume control tank relief valve header contributed to the cause of the event. The licensee determined that both Unit 1 trains of centrifugal charging pumps and the 1A safety injection train were inoperable for 149 hours and 47 minutes.

Analysis: The inspectors determined that the finding involved a performance deficiency, because the licensee did not adequately evaluate external plant operating experience involving several different gas intrusion events, elements of which were identical to the Catawba gas intrusion event. The licensee had several opportunities to evaluate industry events to address the pressurizer as a gas source and evaluate system integration that could lead to inoperability of ECCS equipment. This finding was greater than minor because it affected an objective and attribute of the Reactor Safety Mitigating Systems Cornerstone, in that gas accumulation in the centrifugal charging pump suction piping rendered ECCS systems unavailable and unreliable. The finding was assessed using the SDP for Reactor Inspection Findings for At-Power Situations. Based on information available in LER 05000413/2004001-01, the finding was evaluated using the SDP Phase 2 plant notebook. For this evaluation, the following assumptions were made: (1) the charging pumps were not available for those events that required ECCS flow rates; (2) one safety injection pump was not available for the high pressure recirculation function; and (3) the finding existed for a duration of between three and thirty days. The evaluation determined that the finding exceeded the threshold that required evaluation under Phase 3 of the SDP. The Phase 3 SDP analysis was performed using an exposure time of 150 hours. The NRC's risk model was used to evaluate the finding based on the assumed conditions. Due to the short exposure time and the assumption that the 1A safety injection pump was only affected during high pressure recirculation, the finding was determined to be of very low safety significance (Green).

Enforcement: TS 3.5.2, ECCS, requires that two ECCS trains shall be operable while operating in Modes 1-3. One or more ECCS trains inoperable requires that the train(s) be restored within seventy two hours. Contrary to the above, starting on January 7, 2004, for a period of 149 hours and 47 minutes, two ECCS trains were not maintained in an operable condition within the allowed TS restoration time of 72 hours. Because this failure to maintain two ECCS trains operable was of very low safety significance and has been entered into the CAP as PIP C-04-0008, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000413/2004006-01, Failure to Maintain Two ECCS Trains Operable Due to Gas Accumulation In the Charging Pump Suction Piping.

.2 (Closed) LER 05000413/2004002-00, Manual Reactor Trip Initiated Due to the Closure of a Main Feedwater Isolation Valve

On February 22, 2004, the hydraulic actuator on the 1B steam generator main feedwater isolation valve (1CF-42) failed, causing the valve to stroke closed. Operators diagnosed the event and manually tripped the reactor prior to reaching the lo-lo steam generator automatic reactor trip setpoint. The unit was stabilized at no-load conditions in accordance with plant operating procedures. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the failure of main feedwater isolation valve 1CF-42 and associated corrective actions in PIP C-04-0891.

.3 (Closed) LER 05000414/2004001-00, Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks Found in Steam Generator Channel Head Bowl Drain Line on 2C and 2D Steam Generators

a. Inspection Scope

The inspectors reviewed the LER and associated safety analysis to verify that the cause was identified, that the corrective actions were reasonable, and whether a licensee performance deficiency was associated with the cause of the RCS pressure boundary leakage. The LER documented RCS pressure boundary leakage on the Unit 2 C and D steam generator bowl drain weld areas, which had occurred during the operating cycle between April 2003 and September 2004. The licensee implemented a repair modification to the 2A, 2C, and 2D steam generators prior to returning the unit to service. The licensee determined that the probable cause of the leakage was primary water stress corrosion cracking in the Alloy-600 region of the weld area.

b. Findings

Introduction: A violation of TS 3.4.13.a, RCS Operational Leakage, was identified for RCS pressure boundary leakage occurring while in modes 1, 2, 3 or 4. Enforcement discretion was exercised for this violation. This issue was determined not to be a finding because a performance deficiency was not identified.

Description: The licensee determined the probable cause of the pressure boundary leakage was primary water stress corrosion cracking in the Alloy-600 region of the bowl drain weld area on the 2C and 2D steam generators. Technical Specification 3.4.13.a, RCS Operational Leakage, requires that there shall be no reactor coolant system pressure boundary leakage while in operational modes 1, 2, 3 and 4. The licensee determined, based on boron deposits found during visual inspections of the Unit 2 steam generator bowl drain areas conducted during the 2EOC13 refueling outage, that pressure boundary leakage had occurred during the previous operating cycle.

Analysis: The inspectors determined that a violation of TS 3.4.13.a occurred since a through-wall leak of the RCS pressure boundary existed while the reactor was in operational modes 1 through 4 during the period of April 2003 to September 2004. The inspectors determined that this violation was greater than minor because the RCS pressure boundary leakage was associated with the reactor safety cornerstone objective for initiating events to limit the likelihood of events that could affect RCS integrity during power and shutdown operations.

The inspectors determined that the RCS pressure boundary leakage was not a performance deficiency because the licensee had implemented an enhanced visual and surface inspection program for Alloy-600 weld areas with special emphasis on the steam generator bowl drain area following a similar leak identified and repaired on the 2B steam generator bowl drain in 2001. The decision to implement the enhanced inspection program on the 2A, 2C, and 2D steam generators rather than perform the modification implemented on the 2B steam generator was based on the results of the calculations performed in 2001 which showed that postulated leak rates would be very low if cracking occurred on the remaining bowl drains and that the axial cracks would not result in a catastrophic failure of the weld area. Because a performance deficiency was not associated with this issue, it was not subject to evaluation under the SDP. However, to understand the significance of the violation, Manual Chapter 0609, Appendix A, Determining the Significance of Reactor Inspection Findings for At-Power Situations, was used to evaluate the risk significance of the violation. Using the SDP Phase 1 Screening Worksheet, this assessment concluded that the issue had very low safety significance because the RCS pressure boundary leakage was extremely small and the type of cracking present made a catastrophic failure of the weld area highly unlikely. In addition, a steady increase in leakage through the Alloy-600 weld area would have been detected both in routine RCS leak rate calculations and by area radiation monitors located inside of the containment building. This issue was entered into the licensee's corrective action program as PIP C-04-04663.

Enforcement: The NRC concluded that a violation of TS 3.4.13.a, RCS Operational Leakage, occurred; however, the violation was not attributable to an equipment failure that was avoidable by reasonable licensee quality assurance measures or management controls. Because the applicable criteria specified in the NRC's Enforcement Policy was satisfied, the NRC is exercising enforcement discretion (EA-05-019) in accordance with Section VII.B.6 of the Enforcement Policy and is refraining from issuing enforcement action for this violation.

4OA5 Other Activities

.1 (Closed) Inspection of Reactor Containment Sump Blockage, Temporary Instruction (TI) 2515/153 - (Unit 1 and Unit 2)

a. Inspection Scope

The licensee's response to Bulletin 2003-01, Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors, described interim compensatory measures. The inspectors verified that the interim compensatory measures identified have been implemented, planned, and scheduled. This review included interviews with operators, a review of training records, procedures, documentation of containment inspections and foreign material control activities. During the Unit 2 refueling outage, the inspectors verified that the licensee performed containment walkdowns to quantify potential debris sources. The walkdowns included a structural inspection of the containment sump for gaps in the sump screening. The inspectors conducted a structural integrity inspection of the Unit 2 containment sump on October 13, 2004, following licensee modification and repair activities on the sump. The inspectors inspected the sump screens and structural fit of the containment sump for tears, gaps, and voids greater than one-eighth inch. The inspectors inspected inside the sump for foreign material. Documents reviewed during this inspection are listed in the Attachment to this report. This Unit 2 inspection completes the review for TI 2515/153; therefore, this TI is closed for Unit 1 and Unit 2.

a. Findings

Introduction: A Green NCV was identified by the inspectors for failure to perform adequate quality control (QC) inspections in Unit 2 of the 2A containment sump. Specifically, containment sump screen gaps, which were intended to be closed via repair activities, were not discovered by QC inspection following the repairs. The gap would allow a containment sump bypass flow path for debris to affect downstream emergency core cooling system (ECCS) components during containment recirculation.

Description: The inspectors identified a gap on the 2A containment sump tower area of the enclosure that was approximately three-eighths inch wide by eight inches long, located above the sump screen door, on the top left side. This location had been repaired by welding flat metal bars in place to cover a gap that previously existed. Following the repairs the licensee planned a sump inspection using MP/0/A/7650/100, Opening and Closing of Containment Sump Screens. The procedure identified multiple QC inspection points that were to be inspected for gaps, tears, or voids greater than one-eighth inch. Additionally, the inspectors determined that two work orders had initially been developed with work instructions that specified QC inspections of the gap repairs. These repairs were directly associated with the area that the inspectors found the excessive gap to still exist. The two work orders were not used in their entirety by the licensee due to changes that were made during transfer of the work orders to a modification work package. The specific QC inspections defined in the two original work orders did not get incorporated during the transfer of the work instructions. Since the specific QC work instructions were removed, the only remaining QC inspection that was to be performed was a general inspection using the procedure discussed previously, MP/0/A/7650/100.

Analysis: The inspectors determined that the finding is a performance deficiency because the licensee did not perform adequate inspections following repair of the containment sump. This finding was greater than minor because it affected an objective and attribute of the Reactor Safety Mitigating Systems Cornerstone in that inadequate inspections failed to identify containment sump bypass flow paths for debris to affect downstream ECCS components during containment recirculation, which could have affected the availability and reliability of ECCS systems. The finding was assessed using the SDP for Reactor Inspection Findings for At-Power Situations. The finding was evaluated using the phase 1 SDP analysis and was determined to be of very low safety significance (Green), based on the small size of the gaps and the low probability that material could bypass the sump screen in that area.

Enforcement: 10 CFR 50 Appendix B, Criterion X, Inspection, requires in part, a program for inspection of activities affecting quality shall be established and executed to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Contrary to the above, on October 11, 2004, the licensee failed to perform adequate QC inspections of the 2A containment sump, in that gaps, which were intended to be closed via repair activities, were not discovered by QC inspection following the repairs. Because this failure to perform adequate inspections was of very low safety significance and has been entered into PIP C-04-5511, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000414/2004006-02, Failure to Perform Adequate Inspections of the 2A Containment Sump Following Repairs.

.2 Failure of the 2A Reactor Coolant Pump Oil Collection System to Collect Lube Oil from a Leaking Enclosure Panel Gasket During Power Operation

a. Inspection Scope

The inspectors observed approximately 5 gallons of lube oil on the floor and pump support structure under the 2A reactor coolant pump motor during a Mode 3 walkdown following power operation. The inspectors reviewed PIPs documenting the lube oil leak on the 2A reactor coolant pump motor and the associated root cause investigation to verify that the cause was identified, that both short-term and long-term corrective actions were reasonable, and whether a licensee performance deficiency was associated with the failure of the oil collection system to collect the lube oil leak in the motor oil lift system. The PIPs documented a small lube oil leak on the 2A reactor coolant pump motor oil lift block valve that existed for the entire operating cycle (April 2003 to September 2004) and the corrective actions taken in response to that condition. During the cycle, the licensee refilled the upper motor bearing oil reservoir in November 2003. Licensee observation of the area found no oil leakage on the floor during that evolution. On September 2, 2004, an upper oil reservoir low level alarm was received. The licensee continued power operations and replaced the 2A reactor coolant pump motor during the 2EOC13 refueling outage, which included the lube oil collection system. All other reactor coolant pump motor lube oil collection systems on Unit 2 were inspected during the outage. Sealant was applied to the lube oil collection system enclosure panel gasketed areas on the remaining three reactor coolant pump motors to prevent oil leakage outside of the motor enclosure panels. The 2C reactor coolant pump motor was scheduled to be replaced during the 2EOC14 refueling outage.

b. Findings

Introduction: A violation of 10 CFR 50, Appendix R, Section III.O was identified for the lube oil collection system on the 2A reactor coolant pump allowing approximately 3 to 5 gallons of lube oil to leak from the system onto the pump support and adjacent floor during power operation. Enforcement discretion was exercised for this violation. This issue was determined not to be a finding because a performance deficiency was not identified.

Description: The licensee determined the cause of the lube oil leaking from the oil collection system was an aged gasket located between two bolted enclosure surfaces. The gasket had been in-service since initial plant startup (approximately 18 years). 10 CFR 50, Appendix R, Section III.O, Oil Collection System for Reactor Coolant Pump, requires that reactor coolant pumps in non-inerted containments shall be equipped with an oil collection system capable of collecting lube oil from all potential pressurized and non-pressurized leakage sites. The licensee determined that the oil collection system on the 2A reactor coolant pump motor failed to collect lube oil that had leaked from an oil lift block valve internal to the lube oil recovery system enclosure during power operation.

Analysis: The inspectors determined that a violation of 10 CFR 50, Appendix R, Section III.O occurred since the oil collection system on the 2A reactor coolant pump motor did not capture oil that had leaked from the motor lube oil lift system during power operation. The inspectors determined that this violation was greater than minor because the failure of the oil collection system to capture and contain lube oil leaks was associated with the protection against external factors (fire) attribute and degraded the reactor safety initiating events cornerstone objective to limit the likelihood of those events that challenge critical safety functions during power operations and shutdown.

The inspectors determined that the failure of the oil collection system to capture the lube oil leaking from the enclosure panel was not a performance deficiency because the cause of the oil leak was not reasonably within the licensee's ability to foresee and correct. This conclusion was based on the fact that the motor manufacturer had not identified gasket aging as a potential failure mechanism, no industry operating experience describing gasket leaks was identified, and no leakage of oil was observed during inspections in November 2003. The oil collection system enclosure with the associated gaskets is replaced in its entirety as part of the licensee's reactor coolant pump motor replacement program. Because a performance deficiency was not associated with this issue, it was not subject to evaluation under the Significance Determination Process (SDP). However, to understand the significance of the violation, Manual Chapter 0609, Appendix F, Fire Protection Significance Determination Process, was used to estimate the risk significance of the violation. This assessment concluded that the violation had very low safety significance because review and analysis could not identify credible or likely fire scenarios in the area surrounding the 2A reactor coolant pump that would lead to loss or degradation of safety-related equipment located in the postulated area of the fire. This was based on the small quantity of oil that leaked from the lube oil system, the distance to the nearest cable tray, and the fire suppression sprinkler system associated with the reactor coolant pump motor that was operable during the period of unit operation with the known oil leak. This issue was entered into the licensee's corrective action program as PIP C-04-4562.

Enforcement: The NRC concluded that a violation of 10 CFR 50, Appendix R, Section III.O occurred; however, the violation was not attributable to an equipment failure that was avoidable by reasonable licensee quality assurance measures or management controls. Because the applicable criteria specified in the NRC's Enforcement Policy was satisfied, the NRC is exercising enforcement discretion (EA-05-020) in accordance with Section VII.B.6 of the Enforcement Policy and is refraining from issuing enforcement action for this violation.

.3 (Closed) Unresolved Item (URI) 05000413,414/2003005-02, Inclusion of No-Mode Hours in the "Hours Train Required" Portion of the Residual Heat Removal (RHR) System Performance Indicator Calculation

During the October 2003 review of the data the licensee used to generate the RHR System Safety System Unavailability (SSU) PI, the inspectors found that the licensee was including accumulated defueled (No-Mode) hours as part of the system required availability time used to compute the performance indicator value. The equation used to determine the PI is the ratio of the sum of planned unavailable hours, unplanned unavailable hours and fault exposure hours divided by the hours the system trains were required to be available during the previous 12 quarters. The inspectors noted that excluding the No-Mode hours from the denominator of the equation can cause the PI value to increase. The inspectors reviewed the NEI guidance and clarifying notes for this performance indicator, as well as the applicable frequently asked questions (FAQs). The inspectors found that the response to FAQ #183 stated, "During periods and conditions where Technical Specifications allow both shutdown cooling trains to be removed from service, the shutdown cooling system is, in effect, not required and required hours and unavailable hours would not be counted." Captured under PIP C-03-7216, the licensee indicated that they did not interpret this FAQ to be directly applicable to the issue of excluding No-Mode hours from the RHR SSU PI calculation.

Following the NRC's response to the questions raised in URI 05000413,414/2003005-02, the licensee has changed the methodology used to compute the RHR SSU performance indicator to exclude No-Mode hours from the denominator of the equation for all data reported after July 2004. Recalculation of previously reported RHR SSU performance indicator data with no-mode hours excluded from the denominator did not result in a significant change to the performance indicator value and the 1.5 percent performance indicator threshold required to change the PI from Green to White was not exceeded. As a result, this URI is closed.

.4 (Closed) Spent Fuel Material Control and Accounting at Nuclear Power Plants, TI 2515/154, Unit 1 & Unit 2

This inspection was completed and discussed in NRC Inspection Report 05000413,414/2004003, Section 4OA5; therefore, it is considered closed for Unit 1 and Unit 2.

.5 (Closed) Offsite Power System Operational Readiness - TI 2515/156, Unit 1 and Unit 2

This inspection was completed and discussed in NRC Inspection Report 05000413,414/ 2004004, Section 4OA5; therefore, it is considered closed for Unit 1 and Unit 2.

.6 (Closed) Pressurizer Penetration Nozzles and Steam Space Piping Connections in U. S. Pressurized Water Reactors (NRC Bulletin 2004-01), TI 2515/160, Unit 2 - [Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water Reactors]

a. Inspection Scope

The inspectors reviewed procedures and records documenting activities relative to inspection of the Unit 2 pressurizer penetrations to verify that the licensee complied with commitments made in their July 27, 2004, response to NRC Bulletin 2004-01. The inspectors also independently performed a bare metal visual examination of the following pressurizer penetrations: three six inch diameter safety nozzles, six inch diameter relief nozzle, manway diaphragm, and four inch diameter spray nozzle on pressurizer head; and four instrument nozzle penetrations in the pressurizer vessel wall. The guidelines for the inspection were provided in NRC TI 2515/160, Pressurizer Penetration Nozzles and Steam Space Piping Connections in U. S. Pressurized Water Reactors (NRC Bulletin 2004-01).

b. Findings

There were no indications of boron leakage or penetration degradation in any pressurizer connections examined by the licensee during their inspections.

TI 2515/160 Reporting Requirements:

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

1. Performed by qualified and knowledgeable personnel? [The “bare-metal” visual examinations of the pressurizer penetrations were conducted by NDE inspection personnel who had been trained and qualified in accordance with applicable visual inspection procedures, and were certified in accordance with ASME Code requirements.]

2. Performed in accordance with demonstrated procedures? [The visual inspections were conducted in accordance with Duke Power Procedure QAL - 15, ISI Visual Examination, VT-2, Pressure test, Revision 21. The inspectors reviewed the inspection procedure and verified that it had been reviewed and approved in accordance with the licensee’s procedure review process and NRC requirements. The inspectors verified that the procedure specified inspection prerequisites, inspection requirements, included minimum lighting requirements, adequate instructions for performing the visual examination of the pressurizer penetrations, and inspection documentation requirements.]

3. Able to identify, disposition, and resolve deficiencies? [The inspectors reviewed the licensee's procedures controlling the visual examination and determined that the procedure provided adequate guidance to identify, disposition and resolve identified deficiencies in the pressurizer head penetrations.]

4. Capable of identifying the leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01? [The visual examination method was capable of identifying leakage through and around areas adjacent to the pressurizer penetrations.]

(b) What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system? [Prior to the visual inspections, insulation was removed from the pressurizer head and penetrations. The areas were free of debris, dirt, and boron from other sources. The physical layout of the area was congested, however, with the insulation removed, NDE inspection personnel were able to perform visual inspections around 360E of the circumference of each penetration. There were no viewing obstructions.]

(c) How was the visual inspection conducted? [Inspections were conducted by direct visual by NDE inspection personnel.]

(d) How complete was the coverage? [360E around the circumference of all the nozzles.]

(e) Could small boron deposits, as described in the Bulletin 2004-01, be identified and characterized? [With the lighting available, boron deposits, as described in the bulletin, could have been readily identified and characterized. No boron deposits were found.]

(f) What material deficiencies were identified that required repair? [No material deficiencies were identified that required repair.]

(g) What, if any, impediments to effective examinations, for each of the applied methods, were identified? [No significant items were encountered that impeded the bare metal examinations of the pressurizer penetrations.]

(h) 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? [No indications were identified. Volumetric examinations (UT) were performed on the three six-inch diameter safety nozzles.]

(I) Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system? [No indications of leakage were identified during the current outage.]

4OA6 Meetings

Exit Meeting Summary

On January 06, 2005, the resident inspectors presented the inspection results to Mr. D. Jamil, Site Vice President, and other members of licensee management, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Adams, Human Performance Manager
E. Beadle, Emergency Planning Manager
W. Byers, Security Manager
C. Cauthen, Steam Generator Inspection Engineer
T. Daniels, Emergency Planning/Fire Protection
B. Dolan, Engineering Manager
J. Foster, Radiation Protection Manager
R. Glover, Station Manager
W. Green, Reactor and Electrical Systems Manager
G. Hamilton, Operations Training Manager
G. Hamrick, Mechanical, Civil Engineering Manager
A. Hogge, ISI Plan Manager
D. Jamil, Catawba Site Vice President
L. Keller, Regulatory Compliance Manager
A. Lindsay, Training Manager
P. McIntyre, Acting Safety Assurance Manager
M. Patrick, Work Control Superintendent
J. Pitesa, Operations Superintendent
T. Ray, Safety Assurance Manager
F. Smith, Chemistry Manager
G. Strickland, Regulatory Compliance Specialist
J. Thrasher, Modifications Manager
C. Trezise, Maintenance Superintendent

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000413/2004006-01	NCV	Failure to Maintain Two ECCS Trains Operable Due to Gas Accumulation In the Charging Pump Suction Piping (Section 4OA3.1)
05000414/2004006-02	NCV	Failure to Perform Adequate Inspections of the 2A Containment Sump Following Repairs (Section 4OA5.1)

Previous Items Closed

05000413/2004001-01	LER	Gas Accumulation In Centrifugal Charging Pump Suction Piping (Section 4OA3.1)
05000413/2004002-00	LER	Manual Reactor Trip Initiated Due to the Closure of a Main Feedwater Isolation Valve (Section 4OA3.2)
05000414/2004001-00	LER	Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks Found in Steam Generator Channel Head Bowl Drain Line on 2C and 2D Steam Generators (Section 4OA3.3)
2515/153	TI	Inspection of Reactor Containment Sump Blockage (Section 4OA5.1)
05000413,414/2003005-02	URI	Inclusion of No-Mode Hours in the "Hours Train Required" Portion of the RHR System Performance Indicator Calculation (Section 4OA5.3)
2515/154	TI	Spent Fuel Material Control and Accounting at Nuclear Power Plants (Section 4OA5.4)
2515/156	TI	Offsite Power System Operational Readiness - Unit 1 and Unit 2 (Section 4OA5.5)
2515/160	TI	Pressurizer Penetration Nozzles and Steam Space Piping Connections in U. S. Pressurized Water Reactors (NRC Bulletin 2004-01) - Unit 2 (Section 4OA5.6)

Items Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Nuclear System Directive 317, Freeze Protection Program
Catawba Nuclear Station Freeze Protection Program Engineering Support Document, Rev. 001
Freeze Protection Action Register Readiness Review - Fall 2004
Catawba Nuclear Station Freeze Protection Work Request / Work Order List, November 2004
PT/0/B/4700/038, Cold Weather Protection
PT/0/B/4350/008, Heat Trace Alignment Verification
OP/1/A/6200/014 and OP/2/A/6200/014, Refueling Water System, Enclosure 4.9, Refueling Water Storage (FWST) Tank Cold Weather Protection
OP/1/A/6200/014 and OP/2/A/6200/014, Refueling Water System, Enclosure 4.12, Cold Weather Increased Surveillance
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
IP/O/B/3560/008, Preventative Maintenance and Operational Check of Freeze Protection Heat Trace and Instrument Box Heaters (EHT/EIB) Systems (Fall PM)
IP/0/B/3560/011, Spring Preventive Maintenance and Operational Check of Self Regulated and Constant Wattage Freeze Protection Heat Trace and miscellaneous heated instrument box Heaters (EHT/EIB) Systems
PIP C-04-0096, Not enough heating water flowing to Unit 2 TB heaters TB-11 and TB-32
PIP C-04-0097, Water intrusion into heat trace controller boxes located on Unit 1 and Unit 2 FWST
PIP C-04-0244, EHT drawings/documents need to be improved
PIP C-04-2567, Evaluate rescheduling model work orders for verifying freeze protection is operating properly to cooler ambient conditions
PIP C-04-4064, Outstanding procedural enhancements for the freeze protection Periodic Test (PT) PT/0/B/4700/038.
PIP C-04-6222, PT/0/B/4700/038, Cold Weather Protection, does not properly address verification of heaters in miscellaneous plant areas
PIP C-04-6288, Halogen light under 1FWLT5010 not functioning

<u>PIPs generated as a result of this inspection</u>

PIP C-04-6369, Operations Cold Weather Protection Periodic Test does not properly address verification of heaters in miscellaneous plant areas

Section 1R05: Fire Protection

Pre-Fire Plan for Area 21, Main Control Room, Room 573
Pre-Fire Plan for Area 4, Unit 1 Auxiliary Building 543 foot elevation, Rooms 200 through 248
Pre-Fire Plan for Area 4, Unit 1 Auxiliary Building 560 foot elevation
Pre-Fire Plan for Area 29 / 30, RN Pump Structure
Pre-Fire Plan for Area 6, Unit 1 Electrical Penetration Room 560 foot elevation
Pre-Fire Plan for Area 13, Unit 1 Electrical Penetration Room 577 foot elevation
Pre-Fire Plan for Area 33, Unit 2 B Train Auxiliary Shutdown Panel, Room 263
Pre-Fire Plan for Area 28, Unit 2 B Diesel Generator Room

Section 1RO8: ISI Activities

Procedures

Procedure QAL-13, Inservice Inspection (ISI) Visual Examination, VT-1 and VT-1C, Rev.18, dated 9/11/02
Procedure QAL-14, Inservice Inspection (ISI) Visual Examination, VT-3 and VT-3C, Rev. 24, dated 9/11/02
Procedure QAL-15, Inservice Inspection (ISI) Visual Examination, VT-2, Pressure Test, Rev. 21, dated 2/2/04
Procedure NDE-25, Magnetic Particle Examination, Rev. 21, dated 2/19/03, through FC 03-23
Procedure NDE-35 Liquid Penetrant Examination, Rev 19, dated 1/31/02, through FC 03-22
Duke, Eddy Current Analysis Guidelines for Duke Power Company's D5 Steam Generators, Rev. 3
Duke QA-516, Evaluation of ISI Indications, Rev. 4
Westinghouse PDI-ISI-254, Remote Inservice Examination of Reactor Vessel Shell Welds (Ultrasonic Examination), Rev. 5
Westinghouse MRS-SSP-1673-DDP, Eddy Current Inspection Guidelines for D5 Steam Generators at Catawba Unit 2, Rev. 0
Duke Steam Generator Management Program SGMEP 105, Model D5 Specific Assessment of Potential Degradation Mechanisms for Catawba Unit 2 Steam Generators (Pre-outage Planning Report for 2EOC13), Rev. 5
Westinghouse MRS-GEN-1127, Guidelines for Steam Generator Eddy Current Data Quality Requirements, Rev. 3
Westinghouse WDI-STD-146, Eddy Current Examination of Reactor Vessel Pipe Weld Inside Surface, Rev. 3
Westinghouse WDI-STD-088, Underwater Remote Visual Examination of Reactor Vessel Internals, Rev. 2
Westinghouse WDI-STD-142, Paragon Eddy Current Analysis Guidelines for Inspection of Reactor Vessel Bottom Mounted Instrumentation Tube Penetrations, Rev. 1

Other Documents

Drawing number CN-2NV-257, Reactor Building - Unit 2, Chemical and Volume Control System to Reactor Coolant Pump 2A, Rev. 3
Drawing number CN-2NV-275, Reactor Building - Unit 2, Chemical and Volume Control System to Reactor Coolant Pump 2C, Rev. 11
Drawing number CN-2SM-0046, Reactor/Doghouse, Main Steam System from Steam Generator 2C, Rev. 9
PIP G-03-00294, Invalid UT Exam
PIP C-03-03466, Invalid UT Exam on Dis-similar metal welds
PIP C-04-04663, Boric acid residue on 2C and 2D Steam Generator Bowl Drains
PIP C-04-05051, Linear Indication Discovered during LPT of Weld 2NV-275-1
PIP C-04-054251, Linear Indication Found in SG “C” on Hot Leg during RT (Radiography)
PIP C-04-05257, Linear Surface Indication and Resolution for Weld Attachment on 2-R-SM-1582
Wesdyne Calibration and Scan Data Sheets for Reactor Vessel Welds
Steam Generator Eddy Current Data Analyses for Primary, Secondary, and Resolution Especially Including Over Expansion and Seal Weld Indications
Bottom Mounted Instrumentation UT & ET Examination Data for 58 Tubes

Work Order 98504670 package for Elective Minor Modification CNCE-61943 including liquid penetrant examinations and 10 CFR 50.59 Screen
Liquid penetrant examination reports for weld numbers 2NV257-1, -5, -10, -11, -16, and 2NV275-1
Liquid penetrant examination reports for retest of weld numbers 2NV275-1 after removal of indication identified 9/28/04, and UT thickness exams performed on 9/29/04
Magnetic particle examination report for weld number 2-SM-6D-A, 2SM-46-01 and 2SM-46-02, and welded attachments at Support numbers 2-R-SM-1584 and -1585
Duke letter dated July 27, 2004, Response to NRC Bulletin 2004-01: Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water reactors
Summary of results of visual inspections performed on Unit 2 pressurizer penetrations
Material certifications for Magnaflux cleaner, lot numbers 01G12K and 030F02K, penetrant, lot numbers 97J01K and 01M07K and developer, lot numbers 03D04K and 03A03K; and Magnaflux Ferromor ND8 - yellow powder, lot number 010911-002.
Duke letter dated October 19, 2004, Second Ten Year Inservice Inspection Interval Steam Generator C Hot Leg Nozzle Welds
Duke Calculation CNC-2201.01-00-0006, Evaluation of 2EOC13 ISI Flaw in SG2C to Hot Leg Weld
Westinghouse WCAP-15658-P, Flaw Evaluation for Catawba Unit 2 Steam Generator Primary Nozzle Weld Regions, September, 2004

Section 1R12: Maintenance Effectiveness

PIP C-04-05106, ND pump 2A has a leak through the cover closure stud threads
TSAIL Entry C2-04-02655, Unit 2 CA Pump Turbine, Valve 2CA-048 (CA pump #2 discharge flow control valve to 2C steam generator) passing too much flow
TSAIL Entry C2-04-02649, Unit 2 CA Train “A”, Valve 2CA-060 (2A CA motor driven pump discharge flow control valve to 2A steam generator) failed to pass flow
PIP C-04-5770, Valve 2CA-048 was passing to much flow during CA flow balance testing prior to entering Mode 2 following 2EOC13
PIP C-04-5768, Need to resolve operability / mode change requirements for performing valve inservice tests for CA control valves following flow balance
PIP C-04-5760, When performing flow balance testing on the 2A motor driven CA system, valve 2CA-056 failed to pass flow
PIP C-04-5756, Unplanned Tech Spec entry due to valve 2CA-056 not passing flow to the 2A steam generator
PIP C-04-5757, Valve 2CA-056 would not pass flow while performing PT/2/A/4200/003E, CA Flow Balancing

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Unit 1 Pressurizer Level Control Complex Evolution Plan
PIP M-98-03794, Spike Occurred During Reactor Protection System Testing
PIP C-04-02829, Received annunciator 1AD-6 E/9 Pressurizer Low Level Deviation
PIP C-04-04496, Switching Pressurizer Level Control from 3-2 to 1-2 Caused A Slight Level Transient
CNS-1553.NC-00-0001, Design Basis Specification for the Reactor Coolant System
CNS-011.01EIC-0001, Design Basis Specification for the Protection System
TSAIL entries C0-04-02420 and C0-04-02443

WOs 9867328, 98641940 and 98641672
Unit 1 Critical Evolution Plan, Unit 1 EHC System Fault
Unit 1 Critical Evolution Plan, Unit 1 EHC System Fault, Part 2

Section 1R14: Personnel Performance During Nonroutine Plant Evolutions

Emergency Notification System Notification Report #41246, Unit 1reactor trip caused by a turbine trip on high MSR water level
PIP C-04-6580, Unit 1 reactor trip due to turbine trip on 1B MSR high level indication
PIP C-04-5164, Non-licensed operator inadvertently opened 2TA tie breaker while installing main control board overlay
PT/0/A/4150/001J, Zero Power Physics Testing
PT/0/A/4150/001, Controlling Procedure for Startup Physics Testing
PT/0/A/4150/001, Controlling Procedure for Startup Physics Testing tailgate briefing
OP/2/A/6100/001, Controlling Procedure for Unit Startup
Emergency Notification System Notification Report #41154, Unit 2 manual reactor trip and auxiliary feedwater auto start of steam generator lo-lo level
AP/2/A/5500/014, Control Rod Misalignment
AP/2/A/5500/015, Rod Control
PIP C-04-5878, Catawba Unit 2 Manual Reactor Trip
PIP C-04-6580, Catawba Unit 1 automatic reactor trip
OP/1/A/6100/003, Controlling Procedure for Unit Operation, Enclosures 4.2 and 4.3
OP/1/B/6300/001, Turbine Generator
OP/1/A/6100/005, Unit Fast Recovery
PT/0/A/4150/019, 1/M Approach to Criticality
PT/0/A/4150/002A, Transient Investigation, Enclosure 13.1, Reactor Trip Evaluation (Unit 1 reactor trip of 12/05/04)
Unit 1 Plant Computer and Operator Aid Computer alarm log reports (Unit 1 reactor trip of 12/05/04)
Transient Investigation Report for Unit 1 Reactor Trip on 12/5/04
Troubleshooting plan to determine the cause of the Unit 1 reactor trip of 12/05/04
IP/1/A/3200/011, Procedure for In-Situ response time testing of Resistance Thermal Detectors
Catawba Unit Threat Status update reports (12/06/04 through 12/07/04)

PIPs Generated as a result of this inspection

PIP C-04-5967, Differences noted in the Estimated Critical Rod Position calculated by Operations and Reactor Engineering personnel during the Unit 2 reactor startup
PIP C-04-5976, During portions of troubleshooting the rod control system, a written troubleshooting plan was not used.

Section 1R15: Operability Evaluations

TSAIL Entry C0-04-02524, Control room filtration system
NSD 203, Appendix E for Unit 2 ice condenser blast shield bolting being torqued rather than being hand tight as specified in the Westinghouse installation procedure
WO 98697140, Unit 2 ice condenser blast shield bolting
PIP C-04-5608, Loose bolted connections identified on Unit 2 blast shield bars during FME closeout inspection
Minor Modification Package CD200098, Equivalent substitution of bolts for the Unit 2 ice

condenser blast shield
Westinghouse Electric ice condenser installation procedure
Auxiliary Feedwater Health Report, 2nd trimester 2004
PIP C-04-0742, Unexpected entry into TS due to high temperatures on valve 1CA61 following the run of 1A CA pump for Pump Inservice Test
OP/1/A/6250/002, Auxiliary Feedwater System, Enclosure 4.13.1, Steam Generator (SG) Check Valve Leakby Temperature Data for CA Pump 1A
OP/1/A/6250/002, Auxiliary Feedwater System, Enclosure 4.9, Cooldown of Motor Driven CA Pumps Piping
OP/1/A/6250/002, Auxiliary Feedwater System, Enclosure 4.12, Checking of CA Pipe Surface Temperatures
Calculation CNC-1223.42-00-0045, Determination of Allowable CA Pump S/G Header Check Valve Backleakage Temperatures
Operator Aid Computer Alarm Response for points C1A1369 and C1A1381
Instrument calibration data sheets for 1CAPG5770 (9/24/02 and 11/24/04)
Minor Design Change No. CD100149, 1CA-57 and 1CA-61 Piping Design Temperature Change
NRC Bulletin 85-01, Steam Binding of Auxiliary Feedwater Pumps
Work Order (WO) 98697140, Attach bolts on the Unit 2 ice condenser blast shield

<u>PIPs generated during these inspections:</u>

PIP C-04-6350, Pressure gauge 2CAPG5790, Unit 2 Turbine Driven CA Pump header pressure, was reading 1300 pounds/square inch gauge (psig) when SG pressure was 1000 psig
PIP C-04-6376, An erroneous value was recorded for SG header pressure **in OP/1/A/6250/002, Enclosure 4.13.1, which was completed on 11/21/04**
PIP C-04-6035, The 10CFR50.59 screening for the ice condenser jet impingement shield bolting material design change (CD200098) **did not identify the need for a UFSAR change**

Section 1R16: Operator Workarounds

Nuclear System Directive 506, Operator Workarounds
Catawba Nuclear Station Operator Workaround Book
PIP C-04-2668, Part 21 Notification issued by Rotork Controls related to a deficiency with the primary switch mechanism not operating properly in elevated temperature conditions
PIP C-02-3326, Part 21 Notification for Rotork Add-on Packs
PIP C-04-0150, All reach rods currently installed in the plant can not be relied upon for positive configuration control
PIP C-02-0080, Volume Control Tank Makeup system is not reliable at low flow rates or for short duration usage
PIP C-04-0010, 1NS18A has seat leakage which can cause containment spray to pressurize when residual heat removal is placed in service
PIP C-03-0220, 1/2NF228A may not close during an accident condition due to the actuator spring size
PIP C-04-0120, Background leakage on 2NV172A must be performed weekly due to leakby to reactor holdup tank

Section 1R19: Post-Maintenance Testing

PT/0/A/4400/022 A, RN Pump A Performance Test
Procedure PT/0/A/4400/008 B, RN Flow Balance Train B
Drawings No. CN-2574-2.0, 2.1, 2.2, 2.3, 2.4, 2.5, 2.6 and 2.7, Flow Diagram of Nuclear Service Water System (RN)
IP/1/A/3222/059C, Pressurizer Pressure, Protection channel 3, Loop 1NCPT5170 (PT-457) Calibration
WO 98694201, Calibrate Pressurizer Channel 3
WO 98675986, Replace Circuit Board 1C3-521
WO 98598647, Replace 1CAPS5221
PT/2/A/4250/003E, CA System Discharge Control Valve Throttling Procedure
TSAIL Entry C2-04-02655, Unit 2 CA Pump Turbine, Valve 2CA-048 (CA pump #2 discharge flow control valve to 2C steam generator) passing too much flow
TSAIL Entry C2-04-02649, Unit 2 CA Train “A”, Valve 2CA-060 (2A CA motor driven pump discharge flow control valve to 2A steam generator) failed to pass flow
PIP C-04-5770, Valve 2CA-048 was passing to much flow during CA flow balance testing prior to entering Mode 2 following 2EOC13
PIP C-04-5768, Need to resolve operability / mode change requirements for performing IWV’s for CA control valves following flow balance
PIP C-04-5760, When performing flow balance testing on the 2A motor driven CA system, valve 2CA-056 failed to pass flow
PIP C-04-5756, Unplanned Tech Spec entry due to valve 2CA-056 not passing flow to the 2A steam generator
PIP C-04-5757, Valve 2CA-056 would not pass flow while performing PT/2/A/4200/003E, CA Flow Balancing
PT/0/A/4450/008 E, Control Room Area Chillers Performance Test

PIP generated during these inspections:

PIP C-04-5314, Tubing connection to RN pump 2A was found not to be connected after the pump was reassembled and being returned to service

Section 1R20: Refueling and Outage Activities

PIPs generated during these inspections:

PIP C-04-4562, The as-found condition of the 2A reactor coolant pump oil leak was not communicated to engineering during the 2EOC13 mode walkdown.
PIP C-04-5048, Presence of foreign material noted on the horizontal surfaces of the Unit 2 polar crane hook during the movement of reactor internals
PIP C-04-5707, QC inspection of the Unit 2 ECCS sump screen work did not include an inspection or verification of gap spacing following repairs and modifications
PIP C-04-5511, A gap that exceeded the 1/8" criteria on the ‘A’ ECCS sump screen was identified by the NRC conducting a closeout inspection
PIP C-04-5608, Lower ice condenser blast shield bolting found to be loose during foreign material inspection
PIP C-04-5619, Prior to vacuum refill, Operations had to refill the reactor coolant system above 7.25% to calibrate the ultrasonic level detectors
PIP C-04-5730, Results of the upper and lower containment walkdown conducted by the NRC

PIP C-04-5612, The Unit 2 spent fuel pool (rooms 614, 540 and 541) radiation protection radiological survey plan views were not posted
PIP C-04-5978, Operability impact assessment on having all ice condenser blast shield bolts torqued rather than only hand-tightened as specified in the Westinghouse design document.

Section 1R22: Surveillance Testing

PT/2/A/4400/001, ECCS Flow Balance, dates October 9, 2001 and April 8, 1997
PT/2/A/4200/001 I, As Found Containment Isolation Valve Leak Rate Test, Enclosure 13.18, Penetration Number M327 As Found Type C Leak Rate Test
SM/A/8510/001, Inspection of Ice Condenser Flow Passages
MP/0/A/7150/141, Ice Condenser Lower Inlet Doors Corrective Maintenance
PIP C-04-5305, Lower inlet door testing in the Unit 2 ice condenser was stopped after 4 of 18 doors failed due to condensation in the lower containment freezing the doors shut
PIP C-04-5633, While performing Ice Condenser Lower Inlet Door testing, two doors failed to meet the acceptance criteria
PIP C-04-5635, While performing Ice Condenser Lower Inlet Door testing, six doors failed to meet the acceptance criteria

Section 1EP6: Drill Evaluation

Catawba Nuclear Station Emergency Response Organization Drill 04-06, December 7, 2004
Nuclear Power Plant Emergency Notification Forms for December 7 drill
Emergency Response Organization Drill 04-06 Critique summary

Section 2OS1: Access Control to Radiologically Significant Areas

Procedures, Instructions, Guidance Documents, and Operating Manuals:

SH/0/B/2000/012, Access Controls for High, Extra High, and Very High Radiation Areas, Rev. 3
SH/0/B/2000/005, Posting of Radiation Control Zones, Rev. 1
SH/0/B/2000/007, Placement of Personnel Dosimetry for Non-Uniform Radiation Fields, Rev. 1
RA/0/1100/001, Radiation Protection Routines, Rev. 3
RA/2/1100/002, Unit 2 Refueling Outage Lower Containment Controls and Surveillance, Rev. 7
Radiation Protection Management Procedure (RPMP) 2.2, Radiation Control Zones, Rev. 3
RPMP 2.4, EHRA and VHRA Documentation and Locking Hardware Control Guidelines, Rev. 3
RPMP 7.7, Radiation Work Permits, Rev. 0
Nuclear System Directive (NSD) 305, Special Nuclear Material Safeguards and Accountability, Rev. 1
NSD 501, Temporary Storage of Radioactive Material in the Spent Fuel Pool, Rev. 4
Lesson Plan TTC083, Delta Air Supplied Suit (MURUROA Model V4F1 or MTH2) – User & Rescuer, Rev. 0

RWPs:

RWP 11, Routine Spent Fuel Pool Area Activities (excluding refueling), Rev. 18
RWP 18, Miscellaneous Valve Maintenance, Rev. 14
RWP 2603, U-2 Operations Rounds, Miscellaneous Tagouts, and Fill and Venting in the Aux. Bldg. During 2EOC13, Rev. 9

RWP 2808, U-2 RB S/G Install and Remove Nozzle Dams in A/D Steam Generators During 2EOC13, Rev. 18
RWP 2840, U-2 A, C, and D S/G Rathole Modification During 2EOC13, Rev. 4
RWP 2954, Extra High Radiation Area Entry into U-2 Auxiliary Building Room 308A Pipe Trench, Rev. 6

Records, Data, and Drawings:

Initial "B" S/G Bowl Survey (RWP 2811)
Survey No. O-092104-9, 09/20/2004
Survey No. O-092104-10, 09/20/2004
Survey No. O-092104-13, 10/05/2004
Catawba Nuclear Station Internal Dose Assessments, 01/01/2003 - 10/06/2004
Summary of Spent Fuel Pool Inventory, 07/06/2004
Plan of the Day (POD) for 2EOC13, 10/05/2004
EHRA, VHRA Key Logs
2004 CNS Radiation Worker Practices Observation Tracking System

Audits and Self-Assessments:

NPA Assessment GO-04-007(NPA)(RP)(ALL), Radiation Protection Functional Area Evaluation, 04/2004
Assessment No. RPS-16-04, Assessment of Radiation Protection Controls for Access to Radlogically Significant Areas, conducted 08/11-26/2004
ED Dose Alarms Station Goal Report for 2003

PIPs:

PIP C-03-00167, Worker's electronic dosimeter was turned on but not logged into EDC system while working in the RCA, 01/14/2003
PIP C-03-03143, Radiological data insert on a shielded storage area (posted as contaminated) in the primary Chemistry Hot Lab has not been updated since 10/31/2002, 05/21/2003
PIP C-03-03752, Unposted Radiation Area was discovered on elevation 594 at the Unit 2 VE filters, 06/25/2003
PIP C-03-03918, Door to elevation 522 Pipe Chase Room 113 was found open and radiological posting was obstructed, 07/08/2003
PIP C-04-00652, McGuire PIP M-04-0611 identified a concern with an opening in the door of room 641 being large enough to gain access to the EHRA within the room, 02/10/2004
PIP C-04-02412, Self-assessment of implementation status for assessment of unplanned radiation exposures at Catawba, 05/17/2004
PIP C-04-03621, RP additional perspective for improving performance from Assessment Team Manager's outline, 07/27/2004
PIP C-04-04903, Unit 2 Reactor headstand manway access cover had nuts on only 2 of 6 studs, 09/23/2004
PIP C-04-05217, Self-assessment of RP procedure use documentation, 10/03/2004

Section 2OS2: ALARA Planning and Controls

Procedures, Instructions, Guidance Documents, and Operating Manuals:

Duke Power Nuclear System ALARA Manual, Rev. 16
NSD 208, Problem Investigation Process (PIP)
SH/0B/2000/007, Placement of Personnel Dosimetry for Non-Uniform Radiation Fields, Rev. 1

Records, Data, and Drawings:

Catawba Nuclear Station ALARA Committee Action Item Register, no date.
Catawba Nuclear Station Radiation Protection Surveillance and Control, 2EOC13, Radiation Protection Steam Generator Outage Plan.
Catawba Nuclear Station Dose Reduction Initiative Planning Process, ALARA Planning for the Containment Core Spray Heat Exchanger Replacement Project, no date.
Catawba Nuclear Station Dose Reduction Initiative Planning Process, ALARA Planning for the Reactor Head Work, no date.
List of 2004 RWPs and accumulated dose, 08/20/2004
Radiation Protection Source Term Data Annual review, Catawba Nuclear Station, Report Time Period: 2002-2003
Comparison of Catawba and McGuire Containment Entries at Power for 2003, no date
ALARA Committee, Agenda, dated 04/07/04.
ALARA Committee Meeting Minutes for meeting on 01/07/04
ALARA Committee, Agenda, dated 07/07/04

Assessment and Corrective Actions:

Duke Power Company Report, Assessment of Radiation Protection ALARA Planning and Controls, Assessment Dates 08/11/04 - 08/26/04, ROP Self Assessment
PIP C-03-02821, Dose estimate for U-1ECCS valve line up was exceeded, 05/052003
PIP C-03-03587, Innage exposure to date through May 31st is 2.867 rem over estimate, 06/17/2003
PIP C-C-04604, Additional exposure was received while performing labeling and re-lamping in U-1 NV Valve Gallery, Room 315, 08/13/2003
PIP C-C-05682, Dose received during execution of Unit 1 NM leakrate PT was more than expected, 10/21/2003
PIP C-03-05526, Two work orders for manipulator bridge PMs and replacing bridge crane parts were estimated at 90 mrem and 360 mrem, respectively. Actual exposures for both jobs was 77 mrem, 10/10/2003
PIP C-03-05479, Dose estimate for replacing 1RN40AB AOP frame was 84 mrem. Job performed for 15 mrem, 10/07/2003
PIP C-03-07011, Total exposure for snaking 1FW009 was considerably higher than recent outage, 12/10/2003
PIP C-04-00731, Air Actuator Valve Work during 1EOC14 exceeded it's exposure estimate by >25%, 02/12/2004
PIP C-04-02812, Total dose estimate for the removal of tri-nuke vacuum filter from the Unit 1 Spent Fuel Pool was 4 mrem. Actual total dose received for the job however was 15 mrem. The previous total dose received performing this evolution was 1 mrem. 06/09/2004

PIP C-04-03521, ...dose to repair valve 2KF-73...estimated at 56 mrem actual dose was 5 mrem, 07/21/2004

2PS2 Radioactive Material Processing and Transportation

Procedures, Instructions, Guidance Documents, and Operating Manuals:

EWP (Environmental Work Practice) 2.11, Landfill, Rev. 2
EWP 9.1, Shipping Hazardous Materials, Rev. 7
EWP 9.5, Shipping Dangerous Goods By Aircraft, Rev. 2
EWP 9.9, DOT Security Plan, Rev. 0
US DOT Hazardous Materials Certificate of Registration for Registration Year 2004-2005, Issued 6/29/04
HP/0/B/1000/036, Low Level Waste Landfill Operation Monitoring, Rev. 84
HP/0/B/1004/036, Radioactive Sources, Rev. 0
HP/0/B/1006/002, Collection and Processing of Radioactive Trash and Filters and Use of Radioactive Container Storage Areas, Rev. 11
HP/0/B/1006/003, Receipt and Opening of Radioactive Material Packages, Rev. 9
RA 0 1500 001, 10 CFR 61 Radioactive Waste Classification Program and Determination of Waste Classification, Rev. 0
SH/0/B/2000/006, Removal of Items From RCA/RCZs and Use of Release/Radioactive Material Tags, Rev. 1
SH/0/B/2004/001, Preparation and Shipment of Radioactive Material, Rev. 3
SH/0/B/2004/002, Preparation and Shipment of Radioactive Waste, Rev. 3
MP/0/A/7550/011, Duratek 8-120B Cask Handling, Loading, & Unloading, Rev. 18

Records, Data, and Drawings:

CNS Laboratory Data for 2002 and 2003 Samples (12/17/2002 -10/31/2003) (10 CFR 61 waste stream analysis)
Quality Assurance (QA) Program Approval for Radioactive Material Packages N0. 0266, Rev. 7
Certificate of Compliance 9168, CNS 8-120B Package, Rev.12
EQSS(Employee Qualifications and Skills System) Matrix Report, CNS Radioactive Material Control Group Qualifications, 8/16/2004
Catawba Nuclear Station Annual Solid Radioactive Waste Disposal Report for 2003, ½9/2004
Shipping Papers: 03-0005, Steam Generator Eddy Current Equipment (15 boxes)
Shipping Papers: 03-0011, 8-120 HIC with High Rad Filters individually characterized
Shipping Papers: 03-0013, Fuel Handling Equipment
Shipping Papers: 03-0023, 20 ft Sealand Container-DAW
Shipping Papers: 04-0005, Containment Spray Heat Exchanger
Shipping Papers: 04-0013, 8-120 HIC-Resin

Assessment and Corrective Actions:

PIP C-04-00727, This PIP is written to document and address 7 AFIs (recommendations) identified during the annual RP FAE at CNS. 2/12/2004
PIP C-04-01409, RP Self Assessment RPS-05-04 of 2003 Normal DAW generation performance towards goal, goal development process, and potential reductions of DAW, 3/24/2004

PIP C-04-03678, RP Self Assessment RPS-15-04 for NRC Prep Audit of Radioactive Material Processing and Transportation Using NRC Inspection Plan 71122.02., 7/18/04-7/28/04.
PIP C-04-04187, Review of Maintenance Procedures for opening and closing radioactive shipping cask identified several issues associated with procedure completion. (Documentation Problems)

Section 4OA1: Performance Indicator Verification

Procedures, Instructions, Guidance Documents, and Operating Manuals:

SRPMP 10-1, NRC Performance Indicator Data Collection, Validation, Review, and Approval, Rev. 1

Records, Data, and Drawings:

Memorandum to File - Subject: NRC Performance Indicator Data Review for 04/03, 05/12/2003
Memorandum to File - Subject: NRC Performance Indicator Data Review for 08/03, 09/11/2003
Memorandum to File - Subject: NRC Performance Indicator Data Review for 11/03, 12/08/2003
Memorandum to File - Subject: NRC Performance Indicator Data Review for 01/04, 02/09/2004
Memorandum to File - Subject: NRC Performance Indicator Data Review for 05/04, 06/09/2004

Section 4OA2.2: Problem Identification and Resolution Semi Annual Trend Review

Safety Review Group Monthly Reports: December 2003 through November 2004
PIP C-04-6369, Operations Cold Weather Protection Periodic Test does not properly address verification of heaters in miscellaneous plant areas
PIP C-04-5967, Differences noted in the Estimated Critical Rod Position calculated by Operations and Reactor Engineering personnel during the Unit 2 reactor startup
PIP C-04-5976, During portions of troubleshooting the rod control system, a written troubleshooting plan was not used
PIP C-04-5612, The Unit 2 spent fuel pool (rooms 614, 540 and 541) radiation protection radiological survey plan views were not posted
PIP C-04-5305, Lower inlet door testing in the Unit 2 ice condenser was stopped after 4 of 18 doors failed due to condensation in the lower containment freezing the doors shut
PIP C-04-5633, While performing Ice Condenser Lower Inlet Door testing, two doors failed to meet the acceptance criteria

Section 4OA3: Event Followup

NRC Information Notice 88-23, Supplement 5, Potential for gas binding of high pressure safety injection pumps during a loss-of-coolant accident
PIP C-99-1219, Review the associated industry operating experience concerns for site applicability, including operability/reportability concerns
OP/1/A/6200/001M, Fill and vent of the chemical and volume control system
EP/1/A/5000/ES-1.3, Transfer to cold leg recirculation
EP/1/A/5000/ES-1.4, Transfer to hot leg recirculation
WO98481880-01, 1NV-235 I/R boron test and repair
MP/0/A/7650/088, Controlling procedure for systems pressure testing of American Society of Mechanical Engineers, section XI Duke class A, B and C systems and components
PIP C-04-0270, Ultrasonic testings and venting at 1NV-858 and 860

PIP C-98-0045, Stations response to recommendations associated with Significant Operating Experience Report 97-01
PIP C-04-3597, Identified a performance deficiency in the mechanical/civil engineering area.
PIP C-04-0088, Vented air 1NV858 for approximately 1 minute until a solid stream of water was observed
Event Notification Report 40451, Potential for gas binding centrifugal charging pumps

Section 4OA5.1: Other Activities

PIP C-04-4300, Several alarms received in the control room for 2A reactor coolant pump motor for upper oil reservoir low level
PIP C-04-4374, Potential personnel safety issues related to possible oily surfaces around the 2A reactor coolant pump
PIP C-04-4562, The as-found condition of the 2A reactor coolant pump oil leak when the unit was shutdown for the 2EOC13 refueling outage
OP/2/A/6150/002B, Reactor coolant pump motor oil fill and drain system
Drawing CN-1041-13, General arrangement Containment and Reactor Building Unit 2 Plan, 552 foot elevation

Section 4OA5.2: Other Activities

PIP C-04-05707, During review of work performed on ECCS sump screen, it was noted that the QC inspection did not include verification of not gaps greater than 1/8" in the repair area.
PIP V-03-03457, NRC issued bulletin 2003-01 identifying potential susceptibility of pressurized water reactors recirculation sump screens to debris blockage.
WO 98634756, 2NI-SN-ASMP, inspect NI sump screen gaps
WO 98634757, 2NI-SN-BSMP, inspect NI sump screen gaps
WO 98639796-01, Unit 2 sump screens, open/close as required.
WO 98639796-02, Unit 2 sump 2A screen open/close as required.
WO 98618033, Upgrade Unit 2 Reactor building emergency recirc. sump screen
WO 98696703-01, Repair Unit 2 recirculation sump screens, repair gap on left hand side of 2S
WO 98696703-02, Repair gap on bottom right side 2B sump
Drawing CN-1081-43, Reactor Building Unit 2 Recirculation Sump Screen Assembly
MP/0/A/7650/058, Procedure for fabrication and erection of structural and misc. steel

PIPs generated during these inspections:

PIP C-04-05511, During inspection of the ECCS sump in the pipe chase, the NRC found a gap that exceeded the 1/8" criteria at the left hand corner above the door of the “A” sump

LIST OF ACRONYMS USED

BTP	-	Branch Technical Position
CA	-	Auxiliary Feedwater
CAP	-	Corrective Action Program
CFR	-	Code of Federal Regulations
CNS	-	Catawba Nuclear Station
CY	-	Calendar Year
DAW	-	Dry Active Waste

DG	-	Diesel Generator
ECCS	-	Emergency Core Cooling System
ED	-	Electronic Dosimeter
EHC	-	Electro-Hydraulic Control
EOC	-	End-of-Cycle
ESF	-	Engineered Safety Feature
EHRA	-	Extra High Radiation Area
FAQ	-	Frequently Asked Questions
FME	-	Foreign Materials Exclusion
HPT	-	Health Physics Technician
HRA	-	High Radiation Area
IRT	-	Independent Review Team
ISI	-	Inservice Inspection
KC	-	Component Cooling Water
LER	-	Licensee Event Report
LPT	-	Liquid Penetrant
MSR	-	Moisture Separator Reheater
MT	-	Magnetic Particle
NCV	-	Non-Cited Violation
ND	-	Residual Heat Removal
NEI	-	Nuclear Energy Institute
NI	-	Safety Injection
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
NSD	-	Nuclear Site Directive
NW	-	Containment Penetration Valve Injection
OP	-	Operating Procedure
OS	-	Occupational Radiation Safety
PAR	-	Protective Action Recommendations
PCP	-	Process Control Program
PI	-	Performance Indicator
PIP	-	Problem Investigation Process (report)
PS	-	Public Radiation Safety
QC	-	Quality Control
RA	-	Radiation Area
RCS	-	Reactor Coolant System
RHR	-	Residual Heat Removal
RN	-	Nuclear Service Water
RTP	-	Rated Thermal Power
RWP	-	Radiation Work Permit
SCV	-	Steel Containment Vessel
SDP	-	Significance Determination Process
SSU	-	Safety System Unavailability
TI	-	Temporary Instruction
TS	-	Technical Specification
TSAIL	-	Technical Specification Action Item Log
URI	-	Unresolved Item
UT	-	Ultrasonic
VHRA	-	Very High Radiation Area
WO	-	Work Order