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

July 28, 2006

Duke Power Company, LLC d/b/a
Duke Energy Carolinas, LLC (Duke)
ATTN: Mr. B. H.Hamilton
Site Vice President
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION - INTEGRATED INSPECTION REPORT 05000269/2006003, 05000270/2006003, 05000287/2006003, AND 07200004/2006001

Dear Mr. Hamilton:

On June 30, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Oconee Nuclear Station. The enclosed report documents the inspection findings which were discussed on July 10, and July 17, 2006, 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.

Based on the results of this inspection, one self-revealing and three NRC-identified findings of very low safety significance (Green) were identified; all four of which were determined to violations of NRC requirements. However, because of their very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV 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 United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Oconee Nuclear Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and any response will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Docket Nos.: 50-269, 50-270, 50-287, 72-04
License Nos.: DPR-38, DPR-47, DPR-55

Enclosure: NRC Integrated Inspection Report 05000269/2006003,05000270/2006003, 05000287/2006003 and 07200004/2006001 w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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).

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NAME	DRich	MErnstes	GHutto	ERiggs	GKuzo	NGriffis	GLaska
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Letter to B. H.Hamilton from Michael E. Ernstes dated July 28, 2006

SUBJECT: OCONEE NUCLEAR STATION - INTEGRATED INSPECTION REPORT 05000269/2006003, 05000270/2006003, 05000287/2006003, AND 07200004/2006001

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-269, 50-270, 50-287, 72-04
License Nos.:	DPR-38, DPR-47, DPR-55
Report Nos:	05000269/2006003, 05000270/2006003, 05000287/2006003, and 07200004/2006001
Licensee:	Duke Power Company
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	April 1, 2006 - June 30, 2006
Inspectors:	M. Shannon, Senior Resident Inspector D. Rich, Senior Resident Inspector A. Hutto, Resident Inspector E. Riggs, Resident Inspector N. Griffis, Health Physicist (Sections 2PS1, 4OA1 and 4OA5.4) G. Kuzo, Senior Health Physicist (Sections 2PS1and 2PS3) G. Laska, Senior Operations Examiner (Section 1R11.2) J. Rivera-Ortiz, Reactor Inspector (Sections 1R08 and 4OA2.3(2)) R. Chou, Reactor Inspector (Sections 1R08 and 4OA2.3(2)) R. Moore, Senior Reactor Inspector (Section 4OA5.3)
Approved by:	Michael E. Ernstes, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000269/2006003, IR 05000270/2006003, IR 05000287/2006003, 04/01/2006 - 06/30/2006; Oconee Nuclear Station, Units 1, 2, and 3; Inservice Inspection Activities, Radioactive Gaseous and Liquid Effluent Monitoring Systems, Event Followup, and Other Activities.

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

A. <u>NRC Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

- <u>Green</u>. A self-revealing non-cited violation of Technical Specification (TS) 5.4.1 was identified for failure to adequately implement the procedure requirements for protected train equipment, resulting in the lockout of CT3 transformer and subsequent loss of Unit 3 power while in Mode 6.

  The inspectors determined that the licensee’s failure to adequately implement their procedure for protected train equipment was a performance deficiency. The finding was considered to be more than minor because it affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The finding was determined to be of very low safety significance. This was based on the screening criteria found MC 609, Appendix G, Attachment 1, Checklist 4, Pressurized Water Reactor (PWR) Refueling Operation: Reactor Coolant System (RCS) Level > 23' or PWR Shutdown Operation with time to Boil > 2 Hours and Inventory in the Pressurizer. This finding did not meet the criteria in the checklist for requiring a phase 2 or 3 analysis, in that it did not increase the likelihood of a loss of RCS inventory, did not degrade the licensee’s ability to terminate a leak path or add inventory, or degrade the licensee’s ability to recover Decay Heat Removal (DHR) once it is lost.

  This finding has a cross-cutting aspect in the area of human performance because the licensee’s planned work activities did not effectively keep personnel apprised of the operational impact of the work due to the inadequate implementation of their protected train procedure. (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding involving a non-cited violation of 10 CFR Part 50.55a(g)(4)ii for failure to perform a visual (VT-3) examination of the letdown filter housing supports as required by Section XI of the ASME Code. The examinations were performed with a remote camera and the required examination coverage was not obtained as required by Section XI of the ASME Code. The limited remote VT-3 examinations found no indications that the structural integrity of the supports was unacceptable for service. The licensee entered this issue into the Corrective Action Program.

  This finding was of more than minor significance because the incomplete examination of the letdown filter housing supports, if left uncorrected, could become a more significant structural support concern. In addition, a failure to examine the letdown filter supports as required by the ASME Code is related to the "Equipment Performance" attribute of the "Initiating Events" cornerstone and affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. This finding was evaluated using Phase 1 of the NRC IMC 0609, "Significance Determination Process (SDP)." This finding was of very low safety significance because the worst case degradation of the letdown filter supports would result in a detectable and isolable RCS leak that would not impair the mitigating function of the high pressure injection (HPI) system. (Section 1R08)

- Green. The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI for failure to identify a condition adverse to quality in that East and West Penetration Room containment electrical penetration enclosures had not been maintained, such that a number of enclosures allowed the introduction of dirt and debris inconsistent with conditions under which these penetrations were environmentally qualified.

  The finding was considered to be a performance deficiency in that the licensee failed to maintain the containment electrical penetration covers such that debris was allowed to accumulate in a number of enclosures; thereby, jeopardizing the environmental qualification of safety-related circuits. This finding was considered to be more than minor because it affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events in that, the degraded penetration covers, if left uncorrected could allow the environmental qualification of safety-related circuits to degrade such that they would fail following a high energy line break (HELB) in the east penetration rooms. Using the phase 1 screening worksheet of Manual Chapter 0609, Appendix A, the finding was determined to be of very low safety significance, as it did not result in a loss of operability of any equipment needed to mitigate the effects of a HELB.

This finding has a cross-cutting aspect in the area of problem identification and resolution, as the licensee did not appropriately identify the degraded penetration covers consistent with their corrective action program. (Section 4OA5.1)

Cornerstone: Public Radiation Safety

- Green. The inspectors identified a non-cited violation of 10 CFR 20.1501 (b) for failure to ensure that equipment used for effluent monitoring was calibrated for the radiation measured. Specifically, during a 2005 annual calibration of the Unit 1 Gaseous Waste Disposal Monitor (1RIA-37), an incorrect factor was used in calculating the monitor’s expected response to a Chlorine-36 (Cl-36) calibration source. Since the expected response was calculated incorrectly, the detector’s voltage parameters were also modified incorrectly during the calibration, which resulted in non-conservative detector output from June 6, 2005 to April 10, 2006. This finding was entered into the licensee's corrective action program.

  This finding is greater than minor because it is associated with the Public Radiation Safety Cornerstone and affects the cornerstone objective of assuring adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. The failure to maintain appropriate monitor response of 1RIA-37 directly affected the monitor’s isolation function during release of gaseous effluents. The finding was evaluated using the Public Radiation Safety Significance Determination Process (SDP) and was determined to be of very low safety significance because it did not impair the licensee’s ability to assess dose, and offsite doses from gaseous effluents during the time period in question did not exceed limits in 10 CFR 20.1301 and design criteria in Appendix I to10 CFR Part 50.

  This finding has a cross-cutting aspect in the area of human performance because poor work practices resulted in licensee staff not following established calibration procedures during the monitor calibration. (Section 2PS1)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status:

Unit 1 entered the report period at approximately 100 percent rated thermal power (RTP). On June 13, 2006, the Unit was shutdown to mode 5 to perform emergency sump piping inspections. On June 22, 2005, Unit 1 was returned to 100 percent RTP. The unit operated at or near 100 percent RTP for the remainder of the inspection period.

Unit 2 entered the report period at approximately 100 percent RTP. On April 12, 2006, Unit 2 experienced a reactor trip due to a reactor coolant pump trip. On April18, 2006, the Unit was returned to 100 percent RTP. The unit operated at or near 100 percent RTP for the remainder of the inspection period.

Unit 3 entered the report period at approximately 100 percent RTP. On April 17, 2006, unit coast down commenced until April 29, 2006, when Unit 3 was shutdown from approximately 85 percent RTP for the end-of-cycle (EOC) 22 refueling outage (RFO). On June 7, 2006, Unit 3 achieved 100 percent RTP and remained at or near there for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Severe Thunderstorm Warning

a. Inspection Scope

The inspectors verified that the licensee responded appropriately to a severe thunderstorm warning issued for Oconee County, South Carolina on May 20, 2006. The inspectors verified that operations personnel entered abnormal procedure AP/0/A/1700/006, Natural Disaster, and that efforts to restore the Keowee overhead power path to service were underway in accordance with Enclosure 5.4, Severe Weather. The inspectors verified that all control room operations personnel had reviewed Enclosure 5.1, Tornado Information, as required by the AP.

b. Findings

No findings of significance were identified.

.2 Hot Weather Preparations

a. Inspection Scope

The inspectors observed the condition and readiness of the room cooling equipment for the Units 1, 2, and 3 low pressure injection (LPI) pump rooms to ensure that the ability to maintain ambient temperatures in these rooms consistent with post accident design basis assumptions is preserved during hot weather conditions. The inspectors walked down the applicable portions of the low pressure service water (LPSW) system to verify appropriate flows and temperatures were being maintained. The inspectors also observed the material condition of the LPSW piping, coolers and air handling equipment in the LPI pump rooms to ensure that the licensee was maintaining this equipment at the appropriate level. The inspectors took temperature measurements in the rooms during hot weather conditions to verify that the appropriate ambient conditions were being maintained.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

Partial Walkdown

a. Inspection Scope

The inspectors conducted partial equipment alignment walkdowns to evaluate the operability of selected redundant trains or backup systems while the other train or system was inoperable or out-of-service (OOS). The walkdowns included, as appropriate, reviews of plant procedures and other documents to determine correct system lineups, and verification of critical components to identify any discrepancies which could affect operability of the redundant train or backup system. The following three systems were included in this review:

- The Unit 2, A and C high pressure injection (HPI) pumps with B HPI pump OOS for breaker preventive maintenance
- The Unit 1 and 2 B & C LPSW pumps with the A LPSW pump OOS for train maintenance
- Unit 1 LPI system as part of the Operational Decision Making Issue (ODMI) related to the discovery of foreign material in Units 2 and 3 reactor building emergency sump (RBES) suction piping

b. Findings

No findings of significance were identified.

1R05 Fire Protection

Fire Area Walkdowns

a. Inspection Scope

The inspectors conducted tours in 16 areas of the plant to verify that combustibles and ignition sources were properly controlled, and that fire detection and suppression capabilities were intact. The inspectors selected the areas based on a review of the licensee's safe shutdown analysis and the probabilistic risk assessment based sensitivity studies for fire-related core damage sequences. Inspections of the following 16 areas were conducted during this inspection period:

- Standby Shutdown Facility (SSF) Auxiliary Service Water (ASW) pump room, Electrical Switchgear room, Diesel Generator room, Control Room (4)
- Unit 1 and 2 Control Room (1)
- Keowee Hydro Units (KHU) 1 and 2 (2)
- Unit 1, 2 and 3 Auxiliary Shutdown Panels (3)
- Unit 1, 2 and 3 Equipment Rooms (3)
- Unit 1, 2 and 3 East Penetration Rooms (3)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), Problem Investigation Process reports (PIPs), Design Basis Documents, and licensee calculations to determine the areas of the plant that were susceptible to flooding from internal sources. The inspectors toured the SSF to verify that flood protection features were consistent with the licensee's design requirements and risk analysis assumptions. The inspectors examined the following: the sealing of electrical equipment, such as conduits; holes or unsealed penetrations in floors and walls between flood areas; sump pump operation, and sources of internal flooding that were not analyzed or not adequately maintained.

b. Findings

Introduction: An unresolved item (URI) was identified for the failure to identify and correct a significant condition adverse to quality, in that, a substantial portion of the plant drinking water system piping/tubing inside of the SSF was not designed to withstand a seismic event; thereby posing an internal flooding risk to the safety-related equipment located in the SSF. This issue is being identified as an URI pending further inspection and assessment of the affect of the non-seismic piping on SSF equipment.

Description: On January 19, 2006, while performing an extent of condition investigation for the breached SSF flood protection barrier discussed in URI 05000269,270,287/2006002-01, the inspectors identified an SSF internal flooding vulnerability and notified the on-site engineering group of the concern. The licensee's flow diagrams indicated that a substantial portion of PDW piping inside of the SSF was neither seismically qualified nor restrained. The system was constructed primarily of copper tubing with solder connections and pressurized to approximately 85 psig.

On February 8, 2006, the inspectors requested an update on the licensee's review of this issue. Later that same day, the licensee declared the SSF inoperable. The licensee also generated PIP O-06-0740, which stated plant drinking water to the SSF had been isolated until the issue of seismic qualification of the system was resolved.

The inspectors noted three previous opportunities where the licensee could have identified this issue. PIPs O-00-2821 and 00-3098 document and evaluate a PDW system failure in the Unit 2 East Penetration room, which occurred on August 3, 2000, and spilled 12,000 gallons of water into the Auxiliary Building. PIP O-04-6365 documented a verification of the SSF sump system parameters, which determined that the SSF pump room is vulnerable to in-leakage in excess of 1.37 gpm. PIP O-05-5593 documented a Level 1 Assessment of Auxiliary Building Flood Potential from LPSW and PDW.

On September 26, 1972, the Atomic Energy Commission (AEC) requested that the licensee, "... review Oconee Nuclear Station, Units 1, 2 and 3 to determine whether the failure of any non-Category I (seismic) equipment, particularly in the circulating water system and fire protection system, could result in a condition, such as flooding or the release of chemicals, that might potentially adversely affect the performance of safety-related equipment required for safe shutdown of the facilities or to limit the consequences of an accident." The request letter goes on to state that, "The integrity of barriers to protect critical equipment from potentially damaging conditions should be assumed only when the barrier has been specifically designed for such conditions. If your review determines that safety-related equipment could be adversely affected, provide your plans and schedules for corrective action."

In response to the questions contained in the September 26, 1972 AEC letter, on January 29, 1973, the licensee submitted and implemented Final Safety Analysis Report (FSAR) Supplement 13, which committed the licensee to the non-seismic line break requirements. This supplement is referenced Section 3.4, Water Level (Flood) Design, in the December 31, 2005, edition of the UFSAR.

On April 28, 1983, the NRC Safety Evaluation for the SSF stated that, "The licensee has stated that the SSF, the associated mechanical and electrical systems and power supplies meet or exceed the applicable criteria contained in the Oconee FSAR. Additionally, ASME and IEEE codes are utilized as appropriate, in the design of various subsystems and components." In section 4.0, the Safety Evaluation stated that, "The location of the SSF non-Category I piping has been reviewed to determine those areas of proximity to Category I piping or safety-related equipment. Where Category I piping or safety-related equipment is in the proximity area, the non-Category I piping has been seismically qualified and supported or rerouted out of the problem area." UFSAR Section 9.6.4.3 describes the seismic subsystem analysis with respect to SSF non-Category I piping.

Analysis: During a seismic event, the non-seismically qualified PDW piping/tubing located inside of the SSF could have failed; thereby impacting equipment inside of the SSF. This could have impacted the safety function of the SSF during accident scenarios that required the use of SSF equipment to mitigate the consequences of the event, as the internal flood waters could have rendered the SSF equipment inoperable.

Enforcement: This issue remains unresolved pending further inspection and assessment to determine what impact the non-seismically qualified PDW system piping/tubing may have had on SSF equipment during a postulated event requiring the use of the SSF. This finding does not represent an immediate safety concern since PDW to the SSF has been isolated. Accordingly, it will be identified as: URI 05000269,270,287/2006003-01, Failure to Identify and Correct SSF Internal Flooding Risk. This issue is in the licensee's corrective action program as PIP O-06-0740.

1R08 Inservice Inspection (ISI) Activities (Unit 3)

.1 Piping Systems ISI

a. Inspection Scope

On May 8-12, 2006, the inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and the risk significant piping system boundaries in Unit 3. The inspectors selected a sample of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI required examinations and augmented examinations of alloy 600 welds for review.

The inspectors' activities consisted of an on-site review of nondestructive examination (NDE) and welding activities to evaluate compliance with the applicable edition of the ASME Code, Sections V, IX, and XI (Code of Record for the fourth 10-year ISI interval was 1998 Edition with 2000 Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI, IWB-3000 or IWC-3000 acceptance standards. Specifically, the inspectors directly observed the NDE activities described below and reviewed their corresponding procedures, NDE reports, equipment and consumables certification records, and personnel qualification records:

- Ultrasonic test (UT) and liquid penetrant test (PT) examinations of weld 3HP-241-3 (4-inch pipe, High Pressure Injection System, ASME Class 1)
- PT examination of weld 3HP-241-12A (2.5-inch pipe, High Pressure Injection System, ASME Class 1)

The inspectors also reviewed procedures, NDE reports, equipment and consumables certification records, and personnel qualification records for the NDE activities described below:

- PT examination of weld 3-RPV-CRD-58WH9 (Control Rod Drive, housing to body adapter, ASME Class 1)
- PT examination of weld 3-RPV-CRD-58WH60 (Control Rod Drive, base to motor tube, ASME Class 1)

- PT examination of weld 3-RPV-CRD-58 (Control Rod Drive, motor tube to extension, ASME Class 1)
- PT examination of weld 3-RPV-CRD-58W61 (Control Rod Drive, extension to cap, ASME Class 1)

The inspectors reviewed a sample of UT recordable indications from the last Reactor Pressure Vessel (RPV) 10-year ISI report to verify that the evaluation and disposition of indications were in accordance with the applicable edition of ASME Section XI, IWB-3000. Specifically, the inspectors reviewed the disposition of indications for the following welds:

- Weld W2, RPV Upper to Lower Shell (category B-A, ASME Class 1)
- Weld W4, RPV Transition to Lower Shell (category B-A, ASME Class 1)
- Weld W6, RPV Intermediate to Lower Shell (category B-A, ASME Class 1)
- Weld W15, RPV Outlet Nozzle to Shell at 90 deg (category B-D, ASME Class 1)

The inspectors reviewed a sample of welding activities performed since the beginning of the last refueling outage for ASME Class 1 and 2 piping to evaluate compliance with procedures and the ASME Code. Specifically, the inspectors reviewed weld process control reports, welding procedures, procedure qualification records, welder qualification records, and NDE reports for the following welds:

- Weld 3-LPS-0736-2 and 3-LPS-0736-3, 6-inch diameter pipe, Low Pressure Service Water System, ASME Class 2 (Final NDE: PT examination)
- Weld 3-LP-0224-31, 10-inch diameter pipe, Low Pressure Injection System, ASME Class 1 (Final NDE: PT and RT examinations)

In addition to the review of NDEs required by the ASME Code, the inspectors reviewed a sample of UT examinations performed on Main Feedwater (FDW) System piping as part of the licensee's commitments to address the High Energy Line Break project (letter submitted to the NRC dated November 21, 2005, ADAMS Accession No. ML53340283). Specifically, the inspectors reviewed procedures, NDE reports, equipment and consumables certification records, and personnel qualification records for the NDE activities described below.

- UT examination of weld 3-03-31-16A, 24-inch diameter pipe, FDW System
- UT examination of weld 3-03-31-15A, 24-inch diameter pipe, FDW System
- UT examination of weld 3-03-31-5A, 24-inch diameter pipe, FDW System

The inspectors also conducted a Reactor Building (RB) walkdown of multiple elevations and peripheral locations to assess, in general, the material condition of structures, systems, and components. The areas covered through the walkdown were: first through fourth RB floors, west and east side pipe chases, B cavity, and RB basement.

As a follow-up inspection activity related to the licensee's relief request 04-ON-006, dated September 20, 2004 (ADAMS Accession No. ML0427303380) and its withdrawal letter dated May 18, 2006, the inspectors reviewed the visual examination (VT-3) reports pertained to the components for which relief was requested. The subject components were the supports of letdown filter housing 1A (Unit 1), 2A (Unit 2), and 3A (Unit 3). The inspectors reviewed the following NDE reports to verify compliance with the examination requirements of ASME Code, Section XI, Subsection IWF for the third 10-year ISI interval (Code of Record: 1989 Edition).

- VT-3 of 1-LDFTR-1A supports, letdown filter 1A, Unit 1 (Work Order (WO) 98579823)
- VT-3 of 2-LDFTR-2A supports, letdown filter 2A, Unit 2 (WO 98605101 and 98674855)
- VT-3 of 3-LDFTR-3A supports, letdown filter 3A, Unit 3 (WO 98643692/98528904)

The inspectors reviewed the results of the visual examinations performed in 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." The inspectors reviewed the visual examination reports for the following pressurizer locations:

- Welds 3-PSP-1 and 3PZR-WP45 (4-inch spray line)
- Weld 3-50-34-17 (1-inch vent line and annulus area)
- Welds 3PZR-WP91-1, -2, and -3 (2.5-inch safety/relief valve nozzles)
- Welds 3-50-0027-1, -3A, -5, -7, -9, and -11 (installation welds for level transmitter nozzles)
- Welds 3PRZ-WP63-1, -2, -3, -4, -5, and -6 (fabrication welds for level transmitter nozzles)
- Weld 3-RC-0243-5 and 3PRZ-WP63-7 (sampling nozzle)
- Annulus area for pressurizer thermowell connection (drawing OM-2201-0004-001)

The inspectors also reviewed a sample of visual examinations performed on RCS piping to meet the Material Reliability Program examination requirements for Alloy 600 welds. The inspectors reviewed the visual examination reports for the following RCS locations:

- Welds 3RC-287-3, -6, -7, and -63V (3A Hot Leg)
- Welds 3RC-286-11, -14, -15, and -58V (3B Hot Leg)
- Weld 3P1B1-11 (3B1 Lower Cold Leg)
- Weld 3P1B2-9 (3B2 Lower Cold Leg)
- Welds 3-P1A1-10 and 3-50-21-23 (3A1 Cold Leg)
- Welds 3-P1A2-10 and 3-50-21-1 (3A2 Cold Leg)

- Welds 3-P1B1-10 and 3-RC-265-79 (3B1 Cold Leg)
- Welds 3-P1B2-10 and 3-50-20-9 (3B2 Cold Leg)
- Annulus area for RCS temperature nozzle tubes in connections 10 and 11 (drawings OM-2201-003, OM-2201-954, -955, -956, and -957)

b. Findings

Introduction: On May 12, 2006, while performing the NRC baseline procedure 71111.08, the inspectors identified a finding involving a non-cited violation (NCV) of 10 CFR Part 50.55a(g)(4)ii having very low safety significance (Green) for failure to perform a visual examination (VT-3) of the letdown filter housing supports in Units 1, 2, and 3, as required by Section XI of the ASME Code for the third 10-year ISI interval.

Description: On September 20, 2004, the licensee submitted to the NRC a request for relief for the Oconee Nuclear Station Units 1, 2, and 3 (Relief Request No. 04-ON-006). The relief request related to the visual examination requirements (VT-3) of the ASME Code for the letdown filter housing supports. Subsequently, the licensee withdrew the request for relief in a letter dated May 18, 2006, because the licensee could not justify approval of the relief.

The licensee requested relief from the examination coverage required by the ISI Code of Record for the third 10-year ISI interval (Section XI of the ASME Code, 1989 Edition). The licensee performed VT-3 examinations of the letdown filters' 1A, 2A, and 3A supports on October 15, 2003, May 13, 2004, and November 22, 2004, respectively. The examinations were performed by remote camera and the required examination coverage was not obtained as specified by Section XI of the ASME Code. Consequently, these examinations could not be credited to the corresponding 10-year ISI interval. The limited remote VT-3 examinations found no indications that the structural integrity of the supports was unacceptable for service. An additional VT-3 examination of Unit 2 letdown filter 2A supports was performed on November 7, 2005, which found no significant material degradation that could represent a structural integrity concern.

Because the licensee did not perform a complete VT-3 examination of the letdown filter housing supports in Units 1, 2, and 3, as required by the ASME Code, the inspectors determined that the applicable requirements of the ASME Code, Section XI, Subsection IWF, Item F1.40 were not met for the third 10-year ISI interval of each Unit.

Analysis: The inspectors determined that the failure of the licensee to perform complete VT-3 examinations of the supports for the letdown filter housings in Units 1, 2, and 3, as required by the ASME Code was a performance deficiency that warranted a significance evaluation. This finding was of more than minor significance because the incomplete examination of the letdown filter housing supports, if left uncorrected, could become a more significant structural support concern because the full condition of the supports would be unknown and any significant degradation could remain undetected. In addition, a failure of the letdown filter housing supports due to material degradation could result in a challenge of the letdown filter piping pressure boundary and consequently in a potential RCS leak (i.e., Loss of Coolant Accident initiator contributor), since the letdown filters are part of the normal RCS make-up and purification flowpath through the HPI System. Therefore, a failure to examine the letdown filter supports as required by the ASME Code is related to the "Equipment Performance" attribute of the "Initiating Events" cornerstone and affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. This finding was evaluated using Phase 1 of the NRC IMC 0609, "Significance Determination Process (SDP)." This finding was of very low safety significance because the worst case degradation of the letdown filter supports would result in a detectable and isolable RCS leakage that will not impair the mitigating function of the HPI System. The SDP evaluation considered the fact that the limited VT-3 examinations performed in all Units and the additional examination performed in Unit 2 found no indications that the structural integrity of the supports was unacceptable for service.

Enforcement: 10 CFR 50.55a(g)(4)ii requires that inservice examination of components and system pressure tests must comply with the requirements of the ASME Code (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147). ASME Code, Section XI, Subsection IWF, Article IWF-2500, Table IWF-2500-1, Item F1.40 requires a visual examination (VT-3) of 100% of the supports other than piping supports (Class 1, 2, 3, and MC) each inspection interval. Contrary to this requirement, on October 15, 2003, May 13, 2004, and November 22, 2004, the licensee did not perform a complete visual examination of the supports for letdown filter housing 1A, 2A, and 3A, respectively. These examinations were limited in their coverage and could not be credited to the examination requirements for the third 10-year ISI interval. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (PIP O-06-2903), it is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy: NCV 05000269,270,287/2006003-02, Failure to Perform Adequate Examinations of Letdown Filter Supports.

.2 Boric Acid Corrosion Control (BACC) Program

a. Inspection Scope

On May 8-12, 2006, the inspectors reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and the results of the licensee's Mode 3 containment walkdown inspections performed in the Unit 3 Spring 2006 outage (Documents PIP O-06-02408 and PIP O-06-02417). The inspectors also conducted an independent walk-down of the reactor building to evaluate compliance with licensee's BACC program requirements and verify that degraded or non-conforming conditions, such as boric acid leaks identified during the Mode 3 containment walkdown, were properly identified and corrected in accordance with the licensee's corrective action program.

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

- PIP O-05-03376, Evaluation performed on valve 1LP-105 boron accumulation to justify continued service (Low Pressure Injection System, Unit 1)
- PIP O-06-02054, Boron leak on valves 2HP-119, 2HP-120, 2HP-121, and 2HP-123 (High Pressure Injection System, Unit 2)
- PIP O-06-02795, Boric acid residue on check valve 3CF-12 (Core Flood System, Unit 3)

b. Findings

No findings of significance were identified

.3 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

On May 8-12, 2006, the inspectors reviewed activities, plans, pre-outage degradation assessment, condition monitoring and operating assessment, and procedures for the inspection and evaluation of the steam generator Inconel Alloy 690TT tubing for Unit 3 SGs A and B to determine if the activities were being conducted in accordance with Technical Specifications and applicable industry standards. Data gathering, analysis, and evaluation activities were reviewed. The inspectors reviewed corrective action documents PIP O-05-02678, Unexpected Tube Wear Was Identified in the Unit 1 “A” and “B” and PIP O-05-07602, Unit 2 Steam Generator Inspection Results (2EOC21). The inspectors reviewed data results for tubes for Bobbin Probe Inspection of SG A - R145T018 and R076T131 and SG B - R050T123, R049T124, R048T123, R071T003 and R044T108; and for Special Interest Array Probe of SG A - R145T018 and R076T131 and SG B - R035T002, R036T029, R046T007, R071T003, and R067T014 to verify the adequacy of the licensee’s primary, secondary, and resolution analyses. The inspectors reviewed information to determine that the licensee recorded the expected tube wear degradation and was conducting appropriate evaluations in both SGs A and B. The inspectors also reviewed data operators and analysts’ certifications and qualifications, including medical exams.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI related problems, including welding, BACC, and SG ISI that were identified by the licensee and entered into the corrective action program as PIPs. The inspectors reviewed the PIPs to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the report attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Simulator Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on June 6, 2006 to verify that operator performance was adequate and evaluators were identifying and documenting crew performance problems. The scenario began with the simulated unit operating at 100 percent RTP. The Letdown Storage Tank interlock failed, causing 1HP-24 and 25 to open. This resulted in an unintentional RCS boration, and entry into AP-16. The operating crew bypassed the failed interlock and closed 1HP-24 and 25. The training scenario continued with the initiation of small break loss of coolant accident (SBLOCA) in the RCS letdown line, RCS depressurization, automatic actuation of engineered safeguards (ES) channels 1 and 2, and a loss of sub-cooling margin. The scenario concluded when the letdown line was isolated and the unit stabilized in natural circulation. The inspectors observed crew performance in terms of: communications; ability to take timely and proper actions; prioritizing, interpreting, and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; and oversight and direction provided by the shift supervisor, including the ability to identify and implement appropriate Technical Specification (TS) actions.

b. Findings

No findings of significance were identified.

.2 Annual review of Licensee Requalification Examination Results

a. Inspection Scope

On May 8, 2006, the licensee completed the requalification annual operating tests, required to be given to all licensed operators by 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of the individual operating tests, and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

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, scoping, and handling of degraded equipment conditions, as well as common cause failure evaluations. For each item selected, the inspectors performed a detailed review of the problem history and surrounding circumstances, evaluated the extent of condition reviews as required, and reviewed the generic implications of the equipment and/or work practice problem. For those systems, structures, and components (SSCs) 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 reviewed the following items:

- Lee Combustion Turbine (LCT) Maintenance Preventable Functional Failures:
  - PIP O-06-1281, 6C LCT secured due to a clutch fire
  - PIP O-06-1303, 5C LCT tripped unexpectedly with 6C LCT unavailable
- PIP O-06-3455, Large increase in 3FDW-315 and 316 stroke times and PIP O-06-4129, 1FDW-316 valve stroke time out of the Acceptable Range

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluations

a. Inspection Scope

The inspectors evaluated the following attributes for the eight selected SSCs and activities listed below: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (4) that maintenance risk assessments and emergent work problems were adequately identified and resolved.

- PIP O-06-2368, Error in ORAM Sentinel logic for Plant Transient Assessment Tree Loss of LPSW
- PIP O-06-2367, unexpected Orange ORAM risk condition due to 2CCW-21 failed open (Turbine building Flood concern) and SSF ASW pump quarterly test failure
- PIP O-06-3065, Orange ORAM risk condition due to B high pressure service water (HPSW) pump OOS with fire barriers OOS
- Orange ORAM risk condition due to AP-6 entry coincident with Catawba Dual unit trip causing a Red Grid condition and KHU-2 OOS for scheduled maintenance
- PIPs O-06-2468 and 06-2991, ODMI risk associated with continued Unit 1 operation following discovery of foreign material in Unit 2 and 3 RBES suction piping
- PIP O-06-3625, Orange ORAM risk condition caused by entry into AP-6 with 3B FDWP being unavailable
- Unit 3 Midloop with 3T transformer OOS (plant operations review committee (PORC) review of critical action plan)
- Orange ORAM risk condition due to 2LP-21 OOS

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed the operating crew's performance during selected non-routine events and/or transient operations to determine if the response was appropriate to the event. As appropriate, the inspectors: (1) reviewed operator logs, plant computer data, or strip charts to determine what occurred and how the operators responded; (2) determined if operator responses were in accordance with the response required by procedures and training; (3) evaluated the occurrence and subsequent personnel response using the SDP; and (4) confirmed that personnel performance deficiencies were captured in the licensee's corrective action program. The non-routine evolutions reviewed during this inspection period included the following:

- AP/1/A/1700/016, Abnormal RCP Operation, Reactor Coolant Pump (RCP) Abnormal rps–low oil pot level
- PIP O-06-4103, SSF ASW leak of 10 gpm to Unit 3
- Unit 2 Reactor Trip on April 12, 2006

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 Routine Review

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting risk significant systems, to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered; (4) if compensatory measures were involved, whether the compensatory measures were in place, would work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified, the impact on TS limiting condition for operations (LCOs). The inspectors reviewed the following four operability evaluations:

- PIP O-06-1909, 3LP-55 check valve leakage
- PIP O-06-2286, 2B LPI Cooler LPSW flow indication spiking
- PIP O-06-2978, 3HP-404 lifted during PT/3/A/0400/007, SSF reactor coolant makeup (RCMU) pump testing
- PIP O-06-3138, SSF ASW pipe stress calculation

b. Findings

No findings of significance were identified.

.2 Inadequate Foreign Material Exclusion Controls for Unit 3 RBES

a. Inspection Scope

Following the discovery of foreign material in the Unit 2 RBES suction piping (discussed in detail in Inspection Reports 05000269,270,287/2005005 and 2006002), the inspectors reviewed the activities associated with foreign object search and retrieval of the Unit 3 RBES suction piping.

b. Findings

Introduction: An URI was identified for inadequate foreign material exclusion (FME) controls, which resulted in the introduction of foreign material in the Unit 3, A and B RBES suction lines. These lines provide the flowpath for emergency sump recirculation for LPI, reactor building spray (RBS) and piggy back operation of HPI. This issue is designated as an URI pending further inspection and assessment of the affect of the foreign material on downstream emergency core cooling system (ECCS) components.

<u>Description</u>: On April 30, and May 1, 2006, with Oconee Unit 3 in Mode 5, the licensee conducted an as-found, foreign object search and retrieval (FOSAR) of the A and B recirculation lines from the reactor building emergency sump to the containment sump isolation valves (3LP-19 and 20) prior to installing a new, larger RBES.

The licensee discovered a flat washer (2.75 inches outside diameter and 1.125 inches inside diameter), a cotter pin, a piece of wood (approximately 3 inches long), a piece of wire, small pieces of gasket material, and a metal disc (thought to be a penny) inside of the LPI piping leading from the RBES to the 3A LPI and RBS pumps. The licensee also discovered a nail, a piece of wire (approximately 12 inches long and 0.1 inch thick), a 1/4- inch allen wrench (may have been larger based on licensee video), a couple of pieces of small, thin wire, and the head of an adjustable wrench (approximately 3 inches long, 3 inches wide and 1 inch thick) inside of the LPI piping leading from the RBES to the 3B LPI and RBS pumps.

PIP O-06-2468 documents the discovery of the foreign material, its subsequent retrieval, and the final as-left, foreign material free condition of the piping. PIP O-06-2468 also documents the licensee's investigation of this event. The investigation concluded that, "Foreign material was introduced into the above locations [Unit 3, 'A' and 'B' RBES suction lines] due to complacent FME awareness, training and insufficient procedural controls. FME controls were governed by material accountability log sheets and requirements for the cleanliness levels only until March of 2004. This approach was identified as an area for improvement in 2004 due to lack of consistency in FME barrier setup, knowledge of expectations, and specific procedural field controls. Due to this area for improvement, legacy Foreign material has been identified and removed in systems from previous substandard FME work practices. Based on the debris found in the Unit 3 RBES piping, it is possible that either the train B LPI or BS pump could have been damaged, and the train A BS pump could have been damaged. It is not possible to establish when this debris was introduced to the sump lines."

The Maintenance Rule portion of the PIP O-06-2468 documents the licensee's decision to classify this event as a repetitive, maintenance preventable, functional failure for the LPI system. "This event clearly involved a lack of FME control that led to debris introduction in the pipes. To support the purpose of Maintenance Rule this event will be characterized as a MPFF. The failure is related to a lack of FME control which is controlled by procedures and processes."

The Reportability section of PIP O-06-2468 states that, "Based on the conclusions of the Reportability support evaluation, this event appears to be reportable because the debris/FME could have rendered the associated pumps inoperable and the condition existed longer than TS allows one train of LPI or RBS to be inoperable. Since two trains of RBS could have been affected, this is also potentially a condition that could have resulted in a loss of safety function." As such, the licensee is writing a licensee event report (LER) for this event, which is due to the NRC on July 31, 2006.

Analysis: During accident conditions, the foreign material could be transported to the downstream ECCS components. This could impact the safety function of downstream ECCS components during accident scenarios that require sump recirculation, as the foreign material could render the downstream ECCS pumps unable to satisfy their design safety functions.

Enforcement: This issue remains unresolved pending further inspection and assessment to determine what impact the foreign material may have had on downstream ECCS components during a postulated event requiring RBES recirculation. Accordingly, it will be identified as: URI 05000287/2006003-03, Inadequate Foreign Material Exclusion Controls for the Unit 3, A and B Train Reactor Building Emergency Sump Suction Lines. This issue is in the licensee's corrective action program as PIP O-06-2468.

.3 Inadequate Foreign Material Exclusion Controls for Unit 1 RBES

a. Inspection Scope

Following the discovery of foreign material in the Unit 2 RBES suction piping (discussed in detail in Inspection Reports 05000269,270,287/2005005 and 2006002), and the discovery of foreign material in the Unit 3 RBES, the inspectors reviewed the activities associated with foreign object search and retrieval of the Unit 1 RBES suction piping.

b. Findings

Introduction: An URI was identified for inadequate FME controls which resulted in the introduction of foreign material in the Unit 1, A and B RBES suction lines. These lines provide the flowpath for emergency sump recirculation for LPI, RBS and piggy back operation of HPI. This issue is designated as an URI pending further inspection and assessment of the affect of the foreign material on downstream ECCS components.

Description: Following the discovery of foreign material in the 2A, 2B, 3A and 3B RBES recirculation piping, the licensee conducted a forced shutdown of Unit 1 on June 13, 2006, to inspect the 1A and 1B RBES recirculation piping from the existing RBES to the emergency sump containment isolation valves (1LP-19 and -20). On June 14 and 15, 2006, the licensee conducted the FOSAR of the A and B recirculation piping from the RBES to the emergency sump, containment isolation valves.

The licensee discovered a cylindrical object (approximately 3/16-inch in diameter and 3/8-inch in length) and pipe scale/boron inside of the LPI piping leading from the RBES to the 1A LPI and RBS pumps. The licensee also discovered a nut (approximately 1.25 inches across corner to corner and 0.6 inch thick), a nail (approximately 2.5 inches in length and 0.135 inches in diameter, appears to be an 8d nail), a small cylindrical object (approximately 1/8 inch in diameter and 2 inches in length), a few pieces of thin rubber gasket material, a thin piece of wire, several pieces of thin sheet metal approximately ¾-inch in diameter, a 1-inch O-ring and pipe scale/boron inside of the LPI piping leading from the RBES to the 1B LPI and RBS pumps.

PIP O-06-3928 documents the discovery of the foreign material, its subsequent retrieval, and the final as-left, foreign material free condition of the piping. The Maintenance Rule portion of the PIP O-06-3928 documents the licensee's decision to classify this event as a repetitive, maintenance preventable, functional failure for the LPI and RBS systems. The PIP goes on to state that, "...the purpose of Maintenance Rule is to identify and track shortcomings and deficiencies that may lead to a loss of safety function. This event clearly involved a lack of FME control that led to debris introduction in the pipes. To support the purpose of Maintenance Rule this event will be characterized as a MPFF. The failure is related to a loss of FME control which is controlled by procedures and process."

Analysis: During accident conditions, the foreign material could be transported to the downstream ECCS components. This could impact the safety function of downstream ECCS components during accident scenarios that require sump recirculation, as the foreign material could render the downstream ECCS pumps unable to satisfy their design safety functions.

Enforcement: This issue remains unresolved pending further inspection and assessment to determine what impact the foreign material may have had on downstream ECCS components during a postulated event requiring RBES recirculation. Accordingly, it will be identified as: URI 05000269/2006003-04, Inadequate Foreign Material Exclusion Controls for the Unit 1, A and B Train Reactor Building Emergency Sump Suction Lines. This issue is in the licensee's corrective action program as PIP O-06-3928.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed four modification packages related to safety significant systems to verify that the associated systems' design bases, licensing bases, and performance capability would be maintained following the modifications; and that the modifications would not leave the plant in an unsafe condition. The associated 10 CFR 50.59 screenings/evaluations were also reviewed for technical accuracy and to verify license amendments were not required. The inspectors reviewed the following modification packages:

- OE300846, 3A and 3B Letdown Cooler Replacements
- OD100713, Install Main Steam (MS) Pressure Gauges at Unit 1 MS Atmospheric Dump Valves (ADVs)
- OD200714, Install Main Steam Pressure Gauges at Unit 2 MS ADVs
- OD300715, Install Main Steam Pressure Gauges at Unit 3 MS ADVs

b. Findings

No findings of significance were identified

1R19 Post-Maintenance Testing (PMT)

a. Inspection Scope

The inspectors reviewed PMT procedures and/or witnessed test activities, as appropriate, for selected risk significant systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy consistent with the application; (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform its safety function. The inspectors observed testing and/or reviewed the results of the following five tests:

- PT/2/A/0600/012, 2A Motor Driven Emergency Feedwater Pump Test, following breaker work
- PT/3/A/0251/001, 3A Low Pressure Service Water Pump Test, following pump lubrication
- PT/3/A/0600/012, Unit 3 Turbine Driven Emergency Feedwater Pump Test following the installation of oil sample ports, pump lubrication and an uncoupled overspeed test during the Unit 3 refueling outage
- PT/0/A/0400/011, SSF Diesel Generator Test following Governor Oil Change
- PT/0/A/0400/005, SSF ASW Pump Test following packing adjustment

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors conducted reviews and observations for selected outage activities to ensure that: (1) the licensee considered risk in developing the outage plan; (2) the licensee adhered to the outage plan to control plant configuration based on risk; (3) that mitigation strategies were in place for losses of key safety functions; and (4) the licensee adhered to operating license and TS requirements. Between April 29, 2006, and June 3, 2006, the following activities related to the Unit 3 RFO were reviewed for conformance to applicable procedures and selected activities associated with each evaluation were witnessed:

- Outage risk management plan/assessment
- Clearance activities
- Reactor coolant system instrumentation

- Plant cooldown
- Mode changes from Mode 1 (power operation) to No Mode (defueled)
- Shutdown decay heat removal and inventory control
- Containment closure
- Mid Loop activities
- Refueling activities
- Plant heatup/mode changes
- Core physics testing
- Power Escalation

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of the seven risk-significant SSCs listed below, to assess, as appropriate, whether the SSCs met TS, the UFSAR, and licensee procedural requirements. In addition, the inspectors determined if the testing effectively demonstrated that the SSCs were ready and capable of performing their intended safety functions.

- PT/3/A/0151/019, Unit 3 Penetration 19 Leak Rate Test
- PT/3/A/0251/019, Main Steam Atmospheric Dump Valve Functional Test
- MP/0/A/1200/089, Valve - Main Steam Safety - Setpoint Test (Unit 3)
- PT/0/A/0300/001, Unit 3 Control Rod Drive Trip Time Testing
- PT/0/A/0711/001, Unit 3 Zero Power Physics Test
- PT/1/A/0151/019, Unit 1 Penetration 19 Leak Rate Test
- PT/2/A/0600/012, 2B Motor Driven Emergency Feedwater Pump Test (IST)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed documents and observed portions of the installation of three temporary modifications. Among the documents reviewed were system design bases, the UFSAR, TS, system operability/availability evaluations, and the 10 CFR 50.59 screening. The inspectors observed, as appropriate, that the installation was consistent with the modification documents, was in accordance with the configuration control process, adequate procedures and changes were made, and post installation testing was adequate. The following temporary modifications were included in this review:

- TSM OD100792, Install Temporary Plug at 1LPSW-645, Isolation Valve for 1B2 Reactor Building Aux Cooler Outlet Pressure Transmitter
- TSM OD500772, Replace Waste Gas Flow Indicator, 0GWDPE0003A
- TSM OD300854, Leak Repair FDW Transducer installed by OD300577 (temporary instrumentation installed to monitor the 3A SG for evaluation of tube wear)

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Effluent Monitoring and Radwaste Equipment - During inspector walk-downs, accessible sections of the liquid and gaseous radioactive waste (radwaste) processing and effluent systems were assessed for material condition and conformance with system design diagrams. The inspection included the decant and waste monitoring tanks; demineralizer system; liquid waste system pumps, valves, and piping; liquid waste disposal system effluent monitor (RIA-33); turbine building sump effluent monitors (RIA-54 and 3RIA-54); waste gas disposal system effluent monitors (1RIA-37,38 and 3RIA-37,38); condenser air ejector off gas monitors (1/2/3RIA-40); unit vent effluent monitors (1/2/3RIA-43,44,45,46); radwaste facility vent effluent monitor (4RIA-45); and associated airborne effluent sample lines. The inspectors interviewed chemistry supervision and engineering personnel regarding radwaste equipment configuration and effluent monitor operation. The inspectors observed sample collection and analysis of liquid radwaste released from the Decant Monitor Tank, and assessed those activities for procedural adherence.

The inspectors reviewed performance records and calibration results for selected radiation monitors, flowmeters, and air filtration systems. For monitors 1RIA-37, 1RIA-38, and RIA-33, the inspectors reviewed the two most recent calibration records. The inspectors also reviewed the last two functional/flow checks for the U1 process monitors. The inspectors compared local monitor readings to plant computer data in the control room for 1/2RIA-54 and 1/2/3RIA-45. The inspectors reviewed the out-of-service monitors from September 2004 to May 2006, and verified that required compensatory sampling was performed. The most recent surveillances on the Unit 1, 2, and 3 Reactor Building Purge Filter Systems were reviewed. Performance and operations of the systems were reviewed and discussed with cognizant licensee personnel.

Current licensee programs for monitoring, tracking, and documenting the results of both routine and abnormal liquid releases to onsite and offsite surface and ground water environs were reviewed and discussed in detail. Specifically, the inspectors reviewed and discussed the effect of routine effluent liquid releases made in accordance with ODCM requirements on surface water indicator station sample tritium concentrations. In addition, tritium concentration results since October 2004 for ground water monitoring wells associated with onsite chemical treatment/retention ponds were reviewed and discussed in detail. Reports associated with abnormal liquid releases and corrective actions initiated since calendar year (CY) 1973 were reviewed and discussed with responsible licensee representatives to evaluate the potential onsite/offsite environmental impact of significant leakage/spills from onsite systems, structures, and components. Finally, current licensee capabilities and routine surveillances to minimize and rapidly identify any abnormal leaks from liquid radioactive waste tanks, processing lines, and spent fuel pools, were reviewed and discussed in detail.

Installed configuration, material condition, operability, and reliability of selected effluent sampling and monitoring equipment were reviewed against details documented in the following: 10 CFR Part 20; Regulatory Guide (RG) 1.21, Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials In Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants; ONS Technical Specifications (TS), Section 5.0; the Offsite Dose Calculation Manual (ODCM), Rev. 46; Selected Licensee Commitments (SLC), Section 16.11; and Updated Final Safety Analysis Report (UFSAR), Chapter 11. Procedures and records reviewed during the inspection are listed in Section 2PS1 of the report Attachment.

<u>Effluent Release Processing and Quality Control (QC) Activities</u> - The inspectors directly observed the collection of liquid waste samples and discussed the procedures and processes followed by chemistry personnel for obtaining waste gas samples and liquid effluent samples from waste monitor tanks. In addition, the inspectors discussed the process for performing liquid and gaseous releases with chemistry personnel in the radwaste facility control room. HP technician proficiency in processing and counting effluent samples was evaluated.

QC activities associated with gamma spectroscopy were discussed with count room technicians and HP supervision. The inspectors reviewed daily Performance Data Logs from May 1, 2006 to June 21, 2006 for High Purity Germanium (HPGe) detectors No. 2, 3, 4, and 5; and reviewed licensee procedural guidance for count room QC activities. The inspectors also reviewed Performance Data Logs for the liquid scintillation counters. The inspectors reviewed calibration records for HPGe detector No. 2 (select counting geometries) and liquid scintillation counter No. TR-1. In addition, results of the radiochemistry cross-check program for 1st quarter 2005 through 1nd quarter 2006 were reviewed and discussed with cognizant licensee individuals.

Selected portions of procedures for effluent sampling, processing, and release were evaluated for consistency with licensee actions. Three liquid and three gaseous release permits were reviewed against ODCM specifications for pre-release sampling and effluent monitor setpoints. The inspectors discussed performance of pre-release sampling and analysis, release permit generation, and radiation monitor setpoint adjustment with chemistry technicians and control room operators. The inspectors reviewed the 2004 and 2005 Annual Radiological Effluent Release Reports to evaluate reported doses to the public and ODCM changes. Public dose calculations were reviewed and discussed with cognizant licensee personnel.

Observed task evolutions, count room activities, and offsite dose results were evaluated against details and guidance documented in the following: 10 CFR Part 20 and Appendix I to 10 CFR Part 50; ODCM; RG 1.21; RG 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I; RG 1.33, Quality Assurance Program Requirements; and TS Section 5.0. Procedures and records reviewed during the inspection are listed in Section 2PS1 of the report Attachment.

Problem Identification and Resolution - Multiple Problem Investigation Process reports (PIPs) and a Focus Area Evaluation associated with effluent release activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve selected issues in accordance with Nuclear Site Directive (NSD) 208, Problem Investigation Process, Rev. 27. Reviewed documents are listed in Section 2PS1 of the report Attachment.

b. Findings

Introduction: A Green NRC-identified NCV of 10 CFR 20.1501 (b) was identified for failure to ensure that equipment used for effluent monitoring was calibrated for the radiation measured.

Description: The Unit 1 Normal Range Waste Disposal Gas Monitor (1RIA-37) serves to isolate waste gas decay tanks during a release of waste gases from Unit 1 in the event that the effluent flowing past the monitor exceeds a predetermined radiation level. This isolation feature is one barrier that is in place to avoid releasing unexpected concentrations of radioactive gases into the unit vent, and ultimately, into the atmosphere. During a review of 2005 calibration records for 1RIA-37, inspectors noted that the as-found measured response for the detector was greater than 20% out of tolerance with the expected response of the detector. The as-found value was defined as the response of the detector to a NIST traceable Cl-36 calibration source. During the calibration, this as-found value is compared to an expected response for the Cl-36 source that is specifically listed in the calibration procedure. Through further review of the completed surveillance procedure IP/0/B/0360/043, "Sorrento On-Line Dual Range Gas Monitor," performed on June 8, 2005, the inspectors determined that a multiplication factor had been incorrectly applied to the Cl-36 expected response value, which changed the expected value to 37% less than the specific value listed in the procedure. This resulted in the detector's expected response differing from the as-found measured response by 22%. In order to bring the monitor into acceptable tolerance, technicians reduced the detector's voltage until the measured response matched the incorrect expected response. The as left configuration resulted in non-conservative response of the monitor.

The next calibration of 1RIA-37 was performed on April 10, 2006. From review of surveillance records, inspectors determined that the licensee used the correct value for the detector's expected response to the Cl-36 source during this calibration. The licensee identified that the detector was operating out of tolerance, and adjusted the detector's voltage back to the appropriate value. At this time, the monitor had been operating in a non-conservative configuration for multiple waste gas releases performed during the ten month period between calibrations.

Analysis: The inspectors noted that the failure to properly calibrate 1RIA-37 is a performance deficiency. The finding is greater than minor because it is associated with the Public Radiation Safety Cornerstone and affects the cornerstone objective of assuring adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. The finding was evaluated using the Public Radiation Safety Significance Determination Process (SDP). This issue was related to the effluent release program, but did not result in an impaired ability to assess dose, as the licensee had other means to assess doses from gaseous releases. In addition, offsite doses to members of the public did not exceed the limits in 10 CFR 20.1301 or design criteria specified in Appendix I to 10 CFR 50. For these reasons, the SDP evaluation concluded that the issue was of very low safety significance (Green). This finding has a cross-cutting aspect in the area of human performance because poor work practices resulted in licensee staff not following established calibration procedures during the monitor calibration.

Enforcement: 10 CFR 20.1501 (b) requires that the licensee shall ensure that instruments and equipment used for quantitative radiation measurement (e.g. dose rate and effluent monitoring) are calibrated periodically for the radiation measured. Contrary to the above, effluent monitor 1RIA-37 was not appropriately calibrated for the radiation measured from June 8, 2005 to April 10, 2006. The incorrect calibration resulted in operation of the monitor outside of the required operational tolerances.

The failure to maintain the monitor within allowed calibration tolerances was determined to be of very low safety significance (Green) and has been entered into the licensee's corrective action program (PIP O-06-04070). This violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000269/2006003-05, Failure to Calibrate 1RIA-37 Appropriately for the Radiation Measured.

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

a. Inspection Scope

REMP Implementation - The inspectors observed routine sample collection and surveillance activities required by the licensee's REMP. The inspectors noted the material condition and operability of airborne particulate filter and iodine cartridge sample stations at monitoring location number (No.) 060, 074, 077, 078, 079, and 081C. Environmental thermoluminescent dosimeters (TLDs) at site No. 029, 031, 038, and 040 were checked for material condition and appropriate identification. In addition, the automatic water sampler at 062C (Keowee Hydro Intake) was inspected for material condition. The inspectors determined the current location of selected air samplers, TLDs, and water samplers, using NRC global positioning system instrumentation. Land use census results, changes to the ODCM, and sample collection/processing activities were discussed with responsible staff.

The inspectors reviewed the last two calibration records for selected environmental air samplers. The inspectors also reviewed the 2004 and 2005 Radiological Environmental Operating Reports, results of the 2004 and 2005 interlaboratory cross-check program, and selected procedures for environmental sample collection and processing. Selected environmental measurements were reviewed for consistency with licensee effluent data, evaluated for radionuclide concentration trends, and compared with detection level sensitivity requirements.

Procedural guidance, program implementation, and environmental monitoring results were reviewed against: 10 CFR Part 20; Appendix I to 10 CFR Part 50; TS Section 5.0; ODCM, Rev. 46; SLC Section 16.11; RG 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment; and the Branch Technical Position, An Acceptable Radiological Environmental Monitoring Program - 1979. Documents reviewed are listed in Section 2PS3 of the report Attachment.

Meteorological Monitoring Program - The inspectors directly observed the physical condition of primary (Northwest) and backup (Keowee River) meteorological monitoring towers and discussed equipment operability, reliability, and maintenance activities with responsible licensee representatives. The inspectors reviewed calibration records for applicable tower instrumentation, and evaluated measurement data recovery for CY 2003 through CY 2005.

Licensee procedures and activities related to meteorological monitoring were evaluated against: ODCM; UFSAR Section 2.3; ANSI/ANS-2.5-1984, Standard for Determining Meteorological Information at Nuclear Power Sites; and Safety Guide 23, Onsite Meteorological Programs. Documents reviewed are listed in Section 2PS3 of the report Attachment.

Unrestricted Release of Materials from the Radiologically Controlled Area (RCA) - The inspectors observed surveys of material and personnel being released from the RCA using the Small Article Monitor (SAM), Personnel Contamination Monitor (PCM), and Portal Monitor equipment. The inspectors also observed source checks of selected instruments and discussed equipment sensitivity and release program guidance with licensee staff. To evaluate the appropriateness and accuracy of release survey instrumentation, radionuclides identified within recent waste stream analyses were compared with radionuclides used in current calibration sources and performance check sources. The inspectors also reviewed the last two calibration records for selected SAM, PCM and Portal Monitoring equipment.

Licensee programs for monitoring materials and personnel released from the RCA were evaluated against 10 CFR Part 20 and IE Circular 81-07, Control of Radioactively

Contaminated Material. Documents reviewed are listed in Section 2PS3 of the report Attachment.

Problem Identification and Resolution - The inspectors reviewed selected PIPs and audits in the areas of environmental monitoring, meteorological monitoring, and the unrestricted release of materials and personnel. 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. 27. Documents reviewed are listed in section 2PS3 in the Attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

Cornerstone: Public Radiation Safety

The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results for the period of October 1, 2005 through March 31, 2006. For the assessment period, the inspectors reviewed monthly and quarterly dose calculations to the public, out-of-service effluent radiation monitors, selected compensatory sampling data, and selected PIPs related to Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual issues. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. Documents reviewed are listed in Sections 2PS1 and 4OA1 of the report Attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Daily Screening of Corrective Action Reports

As required by Inspection Procedure (IP) 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 daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing copies of PIPs, attending daily screening meetings, and accessing the licensee's computerized database.

.2 Semi-Annual Trend Review

a. Inspection Scope

As required by IP 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's Corrective Action Program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues; but, also considered the results of daily inspectors CAP item screenings discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the six month period of January 2006 through June 2006, 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 focus area reports, system health reports, self-assessment reports, maintenance rule reports, and Safety Review Group (SRG) monthly reports. The inspectors compared and contrasted their results with the results contained in the licensee's latest quarterly trend data. The inspectors also discussed the licensee's activities in monitoring and trending FME related issues with the SRG Manager and trend coordinator and the FME coordinator from maintenance. 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. However, the inspectors noted that a number of PIPs have been generated that involved FME or were coded as relating to FME. Specifically, 20 PIPs were identified by the inspectors that were written by the licensee during the last six-month period (January 2006 - June 2006). The majority of foreign material found in systems during maintenance appeared to be legacy issues not attributable to current performance, with the majority of these coming from outage work in the secondary plant. This has been attributed to the fact that cleanliness standards were at a lower level prior to 2004, at which time they were upgraded in an effort to reduce the number of FME issues in the secondary plant. Additionally, foreign material recently discovered in all three Units' LPI sump suction lines appeared not to have been introduced into the system recently based on the condition and nature of the material. Data from the SRG that trended foreign material introduced into systems from the forth quarter of 2004 to the present, showed peaks in the number of issues corresponding to outages, with the overall trend of the peaks decreasing. The results of the licensee's latest self-assessment performed by the maintenance FME coordinator concluded that since implementing FME maintenance procedure MP/0/A/1800/001, Open System or Component Foreign Material Exclusion, the overall trends in the four FME categories have improved which can be attributable in part to an increased consistency for FME guidance and behaviors. From a review of the data, there appears to be no significant adverse trend in FME controls with respect to current FME work practices.

.3 Annual Sample Review

(1) Standby Shutdown Facility (SSF) Auxiliary Service Water (ASW) leak on Unit 3

a. Inspection Scope

The inspectors performed an in-depth review of an issue entered into the licensee's corrective action program. The sample was within the mitigating systems cornerstone and involved a risk significant system. The inspectors reviewed the actions taken to determine if the licensee had adequately addressed the following attributes:

- Complete, accurate, and timely identification of the problem
- Evaluation and disposition of operability and reportability issues
- Consideration of previous failures, extent of condition, generic or common cause implications
- Prioritization and resolution of the issue commensurate with safety significance
- Identification of the root cause and contributing causes of the problem
- Identification and implementation of corrective actions commensurate with the safety significance of the issue

The following issue and corrective actions were reviewed:

- PIP O-06-4103, SSF ASW leak on Unit 3

b. Findings

No findings of significance were identified.

(2) Main Feedwater Whip Restraints

a. Inspection Scope

On May 8-12, 2006, the inspectors selected samples in the licensee's PIPs in order to review the adequacy of identification, resolution or corrective actions, and extent of condition reviews associated with problems in the Main Feedwater Whip Restraints. The problems for the whip restraints were identified by a previous NRC inspection (Inspection Report 05000269,270,287/ 2004005) and included maintenance of the gap between the plate and the nut for each rod during the normal operation, questions in Revision 1 of Design Calculation OSC-8370, Analysis of Main Feedwater Rupture Restraints with Bounded Rods, and the extent of condition reviews performed by the licensee regarding similar restraints.

The inspectors reviewed the following PIPs and Minor Modifications (MM):

- PIP O-02-06240, To Summarize Field Inspection of Main Feedwater Rupture Restraints Adjacent to Reactor Building Penetrations 25 & 27

- PIP O-03-01612, Documentation of Feedwater Rupture Restraints Hot Inspection Results
- PIP O-04-00663, Results of Post-Hot Setting Inspection of Unit 1 Feedwater Rupture Restraints
- MM ONOE-17575, Modify 2 Pipe Whip Restraints on Unit 1 Main Feedwater Piping
- MM ONOE-17557, Modify 2 Pipe Whip restraints on Unit 2 Main Feedwater Piping

b. Assessment/Findings

The PIPs, MMs, and several Work Orders (WOs) were to replace the clevis and rods, and to adjust and maintain the gaps between plates and nuts during cold and hot conditions. The gaps (0.045" - 0.050") are required during the hot operating condition. The licensee initiated actions in May 2003 on Unit 3 and was scheduled to inspect the gaps for Unit 3 during the EOC22 RFO. The licensee continues efforts to improve a periodic inspection procedure or guideline to maintain the gaps on all 3 units in the future. The inspectors concluded that the licensee has initiated adequate corrective action and evaluations to maintain the gaps.

The inspectors assessed if calculation-related questions identified in Revision 1 of OSC-8370, were appropriately processed in the corrective action program. The licensee did not generate a PIP for the inspectors' questions associated with the Revision 1 calculation. The licensee's explanation was that the information was new and the inspectors' comments and questions were considered in the attachments for Revision 2. However, the inspectors considered that the licensee should have generated a PIP for the calculation questions raised by inspectors because the issues affected the quality of the calculation. The inspectors noted that the licensee did not require an extent of condition review because all six restraints were inspected. No other similar restraints have the gap problems. The licensee's explanation was acceptable. No findings of significance were identified.

4OA3 Event Followup

a. Inspection Scope

The inspectors evaluated the events listed below to assess the overall impact on the plant and mitigating actions. As appropriate, the inspectors: (1) observed plant parameters and status, including mitigating systems/trains; (2) determined alarms/conditions preceding or indicating the event; (3) evaluated performance of mitigating systems and licensee actions; and (4) confirmed that the licensee properly classified, if applicable, the event in accordance with emergency action level procedures and made timely notifications to NRC and state/county governments as required.

- Unit 2 Reactor Trip on April 12, 2006
- PIP O-06-3002, Unit 3 loss of offsite power (LOOP) while in Mode 6

b. Findings

Introduction: A Green self-revealing NCV of TS 5.4.1 was identified for failure to adequately implement the procedure requirements for protected train equipment, resulting in the lockout of CT3 transformer and subsequent loss of Unit 3 power while in Mode 6.

Description: On May 15, 2006, Unit 3 was in day 17 of refueling outage 3EOC22 with the core reload complete and decay heat removal (DHR) in service (Mode 6). AC power was being supplied to the Unit 3 main feeder buses from off-site via the start up transformer (CT3) while the unit auxiliary transformer (3T) was out-of-service for maintenance. At 10:59 am a lockout of CT3 occurred, resulting in a loss of power to the main feeder buses and the Unit. Both Keowee Units auto started and re-energized the main feeder buses via the standby buses (underground path) within approximately 40 seconds. The control room operators entered AP/3/A/1700/026, Loss of RCS Inventory or Decay Heat Removal While on LPI DHR, and AP/3/A/1700/011, Recovery From Loss of Power. LPI flow was restored at 11:13 am and LPSW flow to the LPI cooler was restored at 11:25 am. RCS temperature increased from 83F to 89F. Spent fuel pool cooling was restored at 11:55 am with no appreciable increase in spent fuel pool temperature.

The CT3 lockout was the result of a mechanical shock applied to the 87BU3 (7kV bus differential) when maintenance personnel manipulated the associated reactor coolant pump monitoring panel (3TCPA) cabinet door on which the relay was mounted. The door was opened to view status lights in the panel and was difficult to shut due to a tight latch. The relay tripped when the technician attempted to shut the door. The licensee did not adequately implement their protected train program as specified in procedure OMP 2-25, Operational Risk Assessment and Management. The protected train program is intended to prevent plant events due to simultaneous maintenance, testing, or inadvertent component manipulations when defense-in-depth is reduced such that a single failure or a single inadvertent component manipulation could cause the event. The licensee did not control/prevent work in the 3TCPA cabinet as there were no protected train controls placed on the cabinet at the time and there was no recognition by the outage control center that there was a vulnerability associated with working in the cabinet, even though a similar lockout occurred with CT1 (Unit 1) in 1999 (PIP 99-3676).

Analysis: The inspectors determined that the licensee's failure to adequately implement their procedure for protected train equipment was a performance deficiency. The finding was considered to be more than minor because it affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The finding was determined to be of very low safety significance (GREEN). This was based on the screening criteria found MC 609, Appendix G, Attachment 1, Checklist 4, PWR Refueling Operation: RCS Level > 23' or PWR Shutdown Operation with time to Boil > 2 Hours and Inventory in the Pressurizer. This finding did not meet the criteria in the checklist for requiring a phase 2 or 3 analysis in that it did not increase the likelihood of a loss of RCS inventory, did not degrade the licensee's ability to terminate a leak path or add inventory, or degrade the licensee's ability to recover DHR once it is lost.

This finding has a cross-cutting aspect in the area of human performance because the licensee's planned work activities did not effectively keep personnel apprised of the operational impact of the work due to the inadequate implementation of their protected train procedure.

Enforcement: Technical Specification 5.4.1 requires that procedures shall be established, implemented and maintained covering the applicable procedures recommended in Regulatory Guide 1.33. Regulatory Guide 1.33, Section 1.c, requires administrative procedures for equipment control. Contrary to the above, the licensee failed to adequately implement their procedure for protected train equipment during a reduction in defense-in-depth for offsite power. Because the finding was determined to be of very low safety significance and has been entered into the licensee's corrective action program (PIP O-06-3002), this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000287/2006003-06, Loss of Unit 3 Offsite Power During Mode 6.

4OA5 Other Activities

.1 (Closed) URI 05000269,279,287/2005004-10, Failure to Maintain Containment Electrical Penetration Enclosures.

Introduction: A Green NCV was identified by the inspectors for failure to identify a condition adverse to quality, in that East and West Penetration Room containment electrical penetration enclosures had not been maintained such that a number of enclosures allowed the introduction of dirt and debris inconsistent with conditions under which these penetrations were environmentally qualified.

Description: This issue was discussed in detail in Inspection Report 05000269,270,287/ 2005004 and was left as unresolved pending further inspection and analysis. Subsequently, the licensee issued a letter dated February 15, 2006, identifying specific penetrations and associated circuits whose terminal boards were exposed to dirt and debris as a result of the deficient penetration covers. The licensee determined that only one instrument listed (Unit 1 Pressurizer Temperature, 1RC RD0043C) was required to be environmentally qualified and this instrument had no affect on safe shutdown as pressurizer level density compensation was accomplished by redundant temperature instruments. The licensee also provided to the NRC in a letter dated April 28, 2006, as part of the on-going high energy line break reconstitution project, a strategy for ensuring the appropriate environmental qualification for circuits needed for safe shutdown following a HELB, including the installation of jet shields and equipment upgrades as necessary.

Analysis: The finding was considered to be a performance deficiency in that the licensee failed to maintain the containment electrical penetration covers such that debris was allowed to accumulate in a number of enclosures; thereby, jeopardizing the environmental qualification of safety-related circuits. This finding was considered to be more than minor because it affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events in that, the degraded penetration covers, if left uncorrected could allow the environmental qualification of safety-related circuits to degrade such that they would fail following a HELB in the east penetration rooms. Using the phase 1 screening worksheet of Manual Chapter 0609, Appendix A, the finding was determined to be of very low safety significance (Green), as it did not result in a loss of operability of any equipment needed to mitigate the effects of a HELB. This finding has a cross-cutting aspect in the area of problem identification and resolution, as the licensee did not appropriately identify the degraded penetration covers consistent with their corrective action program.

Enforcement: 10 CFR 50 Appendix B, Criterion XVI, Corrective Action, requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, the license failed to identify and correct penetration covers that had been removed or misadjusted over a number of years of maintenance activities, where if left uncorrected, could allow accumulation of dust, dirt, rust and debris that could invalidate the environmental qualification of safety related circuits. Because the finding was determined to be of very low safety significance and has been entered into the licensee’s corrective action program (PIP O-05-4491), this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000269,270,287/2006003-07, Failure to Maintain Containment Electrical Penetration Enclosures.

.2 (Closed) NRC Temporary Instruction (TI) 2515/165, Operational Readiness of Offsite Power and Impact on Plant Risk

The inspectors reviewed licensee procedures and controls and interviewed operations and maintenance personnel to verify these documents contained specific attributes delineated in the TI to ensure the operational readiness of offsite power systems in accordance with plant Technical Specifications; the design requirements provided in 10 CFR 50, Appendix A, General Design Criterion 17, "Electric Power Systems;" and the impact of maintenance on plant risk in accordance with 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Documents reviewed are listed in the Attachment. Appropriate documentation of the results of this inspection was provided to NRC headquarters staff for further analysis, as required by the TI. This completes the Region II inspection TI requirements for the Oconee Nuclear Station.

.3 (Open) TI 2515/166, Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) - Unit 3

a. Inspection Scope

The inspectors verified Unit 3 implementation of the licensee’s commitments documented in their September 1, 2005, response to Generic Letter 2004-02, Potential

Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactor for Unit 3. The commitments included permanent modifications and program and procedure changes. Permanent modifications included installation of the sump screen assembly, relocation of the sump level instrumentation, and removal of insulation from the Train B Auxiliary Reactor Building Cooling Unit piping. Program and procedure changes were related to plant labeling, the modification process, coatings control, and the Technical Specification surveillance for periodic screen inspection. The inspectors reviewed the sump screen assembly installation procedure, screen assembly modification 10 CFR 50.59 evaluation, missile evaluation, structural (debris) loading calculation, and the vortex analysis. The inspectors also reviewed the foreign materials exclusion controls and the completed Quality Assurance/Quality Control records for the screen assembly installation. The inspectors conducted a visual walkdown to verify the installed screen assembly configuration was consistent with drawings and the tested configuration and verified the design criteria for screen gap. Additionally, the inspectors field verified the removal of insulation from the Train B Auxiliary Reactor Building Cooling Unit piping.

b. Findings and Observations

No findings of significance were identified.

Unit 3 permanent modifications completed at the time of this inspection, which included the sump screen assembly, the instrument relocation and the insulation removal, were implemented in accordance with the licensee's GL 2004-02 response. The downstream effects and chemical effects studies, which could identify the need for additional modifications were not complete. Program and procedure changes, which will apply to all three units were in process.

TI 2515/166 will remain open pending completion and NRC review of the licensee's GL 2004-02 commitments for all station units, including program changes. Currently, the Unit 1 Emergency Sump modifications are scheduled for the Fall of 2006; the Unit 2 permanent modifications are complete. Pending the results of the downstream and chemical effects studies, the Unit 1 and 2 modifications will be reviewed during the 2006 fall outage. Procedure and program change completions, as documented in PIP O-04-07315, are currently scheduled for December 31, 2007, and will be inspected before December 31, 2008.

.4 Independent Spent Fuel Storage Installation (ISFSI) Radiological Controls

a. Inspection Scope

The inspectors reviewed gamma-ray, neutron, and contamination surveys of the ISFSI facility. Inspectors also observed routine gamma-ray surveys and compared the results to previous surveys and TS limits. The inspectors evaluated implementation of radiological controls, including labeling and posting, and discussed controls with an HP Technician and HP supervisory staff. Environmental monitoring for direct radiation from the ISFSI was reviewed, and inspectors observed placement of thermoluminescent dosimeters.

Radiological control activities for ISFSI areas were evaluated against 10 CFR Part 20, 10 CFR Part 72, and Admentment 2 to the Certificate of Compliance (CoC) No. 1014 details. Documents reviewed are listed in Section 4OA5 of the report Attachment.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings (Including Exit Meeting)

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Bruce Hamilton, Site Vice President, and other members of licensee management at the conclusion of the inspection on July 10, 2006. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
N. Alchaar, Civil Engineering
L. Azzarello, Modification Engineering Manager
S. Batson, Superintendent of Operations
D. Baxter, Station Manager
R. Brown, Emergency Preparedness Manager
T. Bryant, Engineering Support
A. Burns, Civil Engineer, Reactor & Electrical Systems
B. Carney, Repair and Replacement Program
S. Capps, Mechanical/Civil Engineering Manager
T. Coleman, ISI Coordinator
N. Constance, Operations Training Manager
D. Covar, Training Instructor
C. Curry, Maintenance Manager
G. Davenport, Compliance Manager
K. Davis, Corporate Eddy Current Level III Examiner
P. Downing, Corporate Steam Generator Manager
C. Eflin, Requalification Supervisor
P. Earnhardt, Modifications Engineer
P. Fowler, Access Services Manager, Duke Power
T. Gillespie, Reactor and Electrical Systems Manager
J. Gilreath, Site Steam Generator Engineer
M. Glover, Engineering Manager
T. Grant, Engineering Supervisor, Reactor & Electrical Systems
R. Griffith, QA Manager
B. Hamilton, Site Vice President
L. Hekking, BACC Program
R. Hester, Civil Engineer
D. Hubbard, Training Manager
T. King, Security Manager
T. Ledford, Engineering Supervisor, Reactor & Electrical Systems
L. Llibre, Engineering Supervisor
D. Mayes, Corporate Consultant Engineer
R. Murphy, Engineering Support
S. Neuman, Regulatory Compliance Group
L. Nicholson, Safety Assurance Manager
J. Patterson, Engineering Supervisor
W. Pursley, Radiation Protection Supervising Scientist
J. Rowell, Engineer, Reactor & Electrical Systems
J. Smith, Regulatory Affairs
B. Spear, Engineer, Reactor & Electrical Systems
J. Steeley, Training Supervisor
J. Stinson, Engineer, Reactor & Electrical Systems

P. Stovall, SRG Manager
F. Suchar, QC Supervisor
S. Townsend, Keowee Operations
J. Twiggs, Radiation Protection Manager
J. Weast, Regulatory Compliance
A. Well, Civil Engineer

NRC
M. Ernstes, Chief of Reactor Projects Branch 1
L. Olshan, Project Manager, NRR

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000269,270,287/2006003-01	URI	Failure to Identify and Correct SSF Internal Flooding Risk (Section 1R06)
05000287/2006003-03	URI	Inadequate Foreign Material Exclusion Controls for the Unit 3, A and B Train Reactor Building Emergency Sump Suction Lines. (Section 1R15.2)
05000269/2006003-04	URI	Inadequate Foreign Material Exclusion Controls for the Unit 1, A and B Train Reactor Building Emergency Sump Suction Lines. (Section 1R15.3)

Opened and Closed

05000269,270,287/2006003-02	NCV	Failure to Perform Adequate Examinations of Letdown Filter Supports (Section 1R08)
05000269/2006003-05	NCV	Failure to calibrate 1RIA-37 appropriately for the radiation measured (Section 2PS1)
05000287/2006003-06	NCV	Loss of Unit 3 Offsite Power During Mode 6 (Section 4OA3)
05000269,270,287/2006003-07	NCV	Failure to Maintain Containment Electrical Penetration Enclosures (Section 4OA5.1)

Closed

05000269,270,287/2005004-10	URI	Failure to Maintain Containment Electrical Penetration Enclosures (Section 4OA5.1)

TI 2515/165	TI	Operational Readiness of Offsite Power and Impact on Plant Risk (Section 4OA5.2)

Items Discussed

TI 2515/166	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) - Unit 3. (Section 4OA5.3)

DOCUMENTS REVIEWED

Section 1R08: Inservice Inspection Activities

Procedures

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NDE-35, "Liquid Penetrant Examination," Revision 21
OP/0/A/1102/028, "Reactor Building Tour," Revision 24
MP/0/A/1800/132, "Evaluation of Boric Acid Leakage on Mechanical, Structural, and Electrical Components," Revision 1
GTSM0808-1, "Welding Procedure Specification," Revision 6
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PIP Documents

PIP O-06-02408
PIP O-06-02417
PIP O-05-02916
PIP O-04-05109
PIP O-04-01970
PIP O-06-02564
PIP O-06-02462
PIP O-04-05685
PIP O-06-02529
PIP O-05-02320
PIP O-05-07602
PIP O-05-02678
PIP O-06-02903
PIP O-05-07602
PIP G-06-00215*

* Document generated as a result of this inspection

Other Records

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UT Examination Report: UT-06-213
UT Examination Reports: BOP-UT-06-058, -059, -057, -060, -056, -061, -044, -045, -051, and -052
RT Examination Report and Films for weld 3-LP-0224-31
Weld Process Control Records for Work Orders 98629418 and 98538274
UT Calibration Reports: CAL-06-241, -242, -243, -221, and -218
Certification for PT Examination Consumables: Batch No. 04G05K, 03E04K, 05K11K, 04D02K, 01M07K, and 02L02K
Certification for UT Examination Couplant: Batch No. 3125
Certification for UT Transducers: 2295-93004, 26269, 26261, 00L5CF, H26794, and 57462-08724
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Summary of ONS Steam Generator Tube Wear Inspection
SGMEP 105, “ROTSG Specific Assessment of Potential degradation Mechanisms for Oconee Unit 3 EOC 22,” Revision 4
Areva Eddy Current Examination Plan for Oconee Unit 3 EOC22
Eddy Current Data Analyses for tubes for SG A - R145T018 and R076T131 and SG B - R050T123, R049T124, R048T123, R035T002, R035T004, R036T029, R036T011, R046T007, R067T014, and R071T003
Areva Document Identifier 51-9009275-000, “A Condition Monitoring and Operational Assessment (CMOA) Evaluation of Wear Scars for Oconee Unit 2, EOC 21,” Rev. 00
Data Acquisition and Analysis Personnel Qualification for Level II Data Operators, Level II A and Level III Analysts
Steam Generator Health Report for 2005T3
Areva FTP Site for Data Analysis Personnel Qualification and Certification
Oconee Plant Operations Review Committee Unit 3 Steam Generator Inspection Results, May 11, 2006 Meeting
Oconee Unit 3 EOC 22 Steam Generator Condition Monitoring and Repair Limits

Section 2PS1: Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

Procedures, Guidance Documents, and Reports

ONS Selected Licensee Commitments (SLC) Section 16.11, Radiological Effluents Control
Offsite Dose Calculation Manual, Revision (Rev.) 46
CP/0/B/5200/045, Liquid Waste Release from the Radwaste Facility (RWF), Rev. 70
CP/0/B/5200/048, Resin Recovery System Operation, Rev. 85
HP/0/B/1001/009, Count Room Instrument Performance Check Procedures, Rev. 15

IP/0/B/0360/040, Sorrento Gas Monitor Sample Flow and Flow Control Tests, Rev. 15
IP/0/B/0398/019, RWF Liquid Radiation Monitor - (RIA-33), Rev. 16,
IP/0/B/0360/039, Sorrento Liquid Monitor Calibration, Rev. 27
IP/0/B/0360/043, Sorrento On-Line Dual Range Gas Monitor, Rev. 17
RPSM 8.7, Quality Assurance for Count Room Instrumentation, Rev. 03
Site Directive 2.4.2, Instrumentation Out of Tolerance Program, Rev. 01
2004 ONS Annual Radioactive Effluent Release Report
2005 ONS Annual Radioactive Effluent Release Report

Records, Data, and Drawings

RWF Liquid Radiation Monitor (RIA-33) calibration data, WO # 98655239-01, 07/6/04
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Dual Range Gas Monitor (1RIA-37/38) calibration data, WO # 98705345-01, 06/06/05
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U1 Function/Flow Check of Process RIAs, WO # 98762066-01, 02/20/06
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Batch Gaseous Waste Release # 2005-011, 04/09/05
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Continuous Gaseous Waste Release # 2006-035, 04/01/06 - 05/01/06
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Groundwater Radiological Data, Gamma Spectroscopy and Tritium Analysis Results. Chemical Treatment Pond (CTP) Locations A-1, A-2, A-8, A-9, BG-4, A-10, A-11, A-12, A-13, A-14, A-17, and A-18, conducted October 2004, January 2005, April 2005, July 2005, October 2005, January 2006, April 2006
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Groundwater Radiological Data, Gamma Spectroscopy and Tritium Analysis Results. Onsite Landfill Monitoring Well (MW)-03, MW-11, MW11D, MW-13, and MW-16 conducted January 2005, January 2006
Decommissioning File Contents Review Data, conducted March 2006
PT/1/A/0110/005 C, Reactor Building Purge Filter Test, 04/09/05 and 06/15/05
PT/2/A/0110/005 C, Reactor Building Purge Filter Test, 11/01/05
PT/3/A/0110/005 C, Reactor Building Purge Filter Test, 12/02/04
OOS effluent monitor data, 9/01/04 - 05/20/06
Inter-laboratory Cross Check Program for 1st Qtr. 05 through 1st Qtr. 2006
Performance Data Log Sheets for Count Room instrumentation, 05/01/06 - 06/21/06
Calibration Data for Packard TriCarb Model 1000 LSC, 05/02/05
HPGe No. 2 calibration data for select geometries: particulate filter (04/15/03), charcoal cartridge (02/26/04), rheodyne vial (03/3/01), 50 ml bottle (03/02/01), 100 cc gas bomb (2/23/01), one liter bottle (03/05/01), four liter gas marinelli (03/05/01)
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PIP O-04-06216, Unexplained increase in Unit 2 Auxiliary Building vent stack radiation levels above the alarm setpoints
PIP O-05-02327, 3A GWD Tank release automatically terminated on 3RIA-37 reaching high setpoint, 04/09/05
PIP O-05-05009, 1RIA-40 indication has been trending up over time, 08/30/05
PIP O-05-05654, Possible emerging trend on RIA system failures
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PIP O-05-06114, 3RIA-37 failed source check, 09/23/05
PIP O-05-06883, 1RIA-32 Alarms during certain evolutions, 10/26/05

Section: 2PS3: REMP and Radioactive Material Control Program

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IP/O/B/1601/011, Meteorological Wind Speed Calibration, Rev. 09
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HP/0/B/1000/067, Quality Assurance for Automated Personnel Monitors, Rev. 20
SH/0/B/2008/001, Calibration and Quality Assurance of Canberra Argos-4AB Contamination Monitors, Rev. 01
Duke Energy Corporation Environmental Radiation (EnRad) Laboratories Procedure 700, Preparation of Environmental Sampling Supply Kits, Rev. 02
Duke Energy Corporation EnRad Laboratories Procedure 702, Airborne Radioiodine and Airborne Particulate Sampling at Oconee Nuclear Station, Rev. 04
Duke Energy Corporation EnRad Laboratories Procedure 703, Water Sampling at Oconee Nuclear Station, Rev. 03
Duke Energy Corporation EnRad Laboratories Procedure 705, Broadleaf Vegetation Sampling at Oconee Nuclear Station, Rev. 03
Duke Energy Corporation EnRad Laboratories Procedure 708, Direct Radiation Measurement (TLDs) at Oconee Nuclear Station, Rev. 03

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2005 Annual Radiological Environmental Operating Report
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Delta Temperature Channel Equipment Calibration Data, North West Tower 10 and 60 Meter Equipment, and Keowee River Site Tower, 10 Meter Equipment; 04/17/06
Oconee Nuclear Station: Meteorological Data Recovery Reports for CY 2005, CY 2004, and CY 2003
Annual Calibration of ISCO Composite Sampler Serial Number 00286, 01/17/05
Field Calibration Data Sheet, ISCO Composite Liquid Sampler Serial Number 00286, 1/16/2006
Low Volume Air Sampler Calibration Data Sheet, Air Sampler Serial Number 00355, 07/09/04, and 02/27/06
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Efficiency Verification Worksheet, EnRad 01803, 03/28/06
SAM-9/SAM-11 Calibration Worksheet, EnRad No. 01802, 03/28/06
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PIP O-05-04082, Radioactive Material In An Uncontrolled Area, 06/16/2005
PIP O-05-08310, Radioactive Material Found Outside of RCA, 12/08/2005
PIP O-04-05855, Keowee Meteorological Tower is Approximately 1 Foot Underwater, 09/08/04
PIP O-04-05861, Environmental Water Sampler Flooded Due to Opening of Keowee Hydro Spill Gates, 09/08/2004
PIP O-04-01437, Applicability of NRC Information Notice 04-05, Spent Fuel Pool Leakage to Onsite Ground Water, 03/18/04
PIP G-05-00331, Cross-Check Gamma Q053GWS [Gamma in Water] Failure to Identify Hg-203, 09/26//2005
PIP G-06-00038, Q054GWR Anomalous Hg-203 in Cross Check, 01/19/2006
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ONS Effluents Dose Summary Memo, 5/20/06
Standard Radiation Protection Management Procedure (SRPMP) 8-2, Investigation of Unusual Radiological Occurrences, Rev. 00

Section 4OA2.3(2): Identification and Resolution of Problems

Procedures

NSD 208, "Problem Investigation Process (PIP)," Revision 27
MP/0/A/3019/011, "Hanger - Minor Corrective Maintenance," Revision 12
MP/0/A/3019/004, "Hangers - QA Condition 1 and 4 - Removal, Installation, or Modification," Revision 57

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PIP O-03-01612, Documentation of Feedwater Rupture Restraints Hot Inspection Results
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PIP O-03-04461, The Senior Resident Discovered an Error in Calculation OSC-8370, Rev. 0, Analysis of Main Feedwater Rupture Restraint with Bounded Rods
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Section 4OA5.3: TI2515/166

MP/0/A/1800/105A, Final Sump Inspection and Cover Installation, Enclosure 13.7, RBES - LPI Suction Line Flange Installation and Removal Drain Line and Sump Inspection, Rev. 04
3-SA-096.009, Proof of Absence of Vortex and Air Intake, Oconee Units 1,2 &3, Rev. 0
S-004, Structural Evaluation of the Emergency Sump Strainer Support Loads, Unit 3, Rev. 0
Duke Power Response to NRC GL 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs, dated September 1, 2005
PIP 0-06-02468, Video Inspection of Emergency Sump 3LP-19 and 3 LP-20 Intake Lines Complete, dated 5/1/2006
WO 98722972, Remove Insulation U3 B Train Aux RBCU, dated 5/15/06
S&W Calculation S-005, Missile Evaluation for the Emergency Sump Strainer, Rev. 1
ALION-REP-DUKE 2736-02, GSI-191 Baseline Analysis for Oconee Units 1,2 &3, dated 7/14/05
TN/3/A/OD300050/01C, Replace the Unit 3 Reactor Building Emergency Sump Strainer, Rev. 1

PIP O-04-07315, Documentation and Tracking of Activities Related to GL 2004-02, dated 10/28/04

4OA5.4: ISFSI Radiological Controls

Procedures, Manuals, and Guidance Documents

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Certificate of Compliance (CoC) No. 1004, Amendment No. 8, 1/23/95
ONS Independent Spent Fuel Storage Installation (ISFSI) Annual Effluent Release Report, 2/23/06

Surveys, Data, and Records Reviewed

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Perimeter TLD Exposure Report spreadsheet data

LIST OF ACRONYMS

ACB	-	Air Circuit Breaker
ADAMS	-	Agency wide Documents Access and Management System
ADV	-	Atmospheric Dump Valve
AEC	-	Atomic Energy Commission
ANS	-	American Nuclear Society
ANSI	-	American National Standards Institute
ARM	-	Area Radiation Monitor
ATWS	-	Anticipated Transient Without SCRAM
AMSAC	-	ATWS Mitigation System Actuation Circuitry
AP	-	Abnormal Procedure
ASME	-	American Society of Mechanical Engineers
ASTM	-	American Society for Testing and Materials
ASW	-	Auxiliary Service Water
BAAC	-	Boric Acid Corrosion Control
BMV	-	Bare Metal Visual
CAM	-	Continuous Airborne Monitor
CAP	-	Corrective Action Program
CC	-	Component Cooling
CCW	-	Condenser Circulating Water
CFR	-	Code of Federal Regulations
Cl-36	-	Chlorine-36
CoC	-	Certificate of Compliance
CRD	-	Control Rod Drive
CROABF	-	Control Room Outside Air Booster Fan
CTPD	-	Core Thermal Power Demand
DBD	-	Design Basis Document
DEC	-	Duke Energy Corporation

DG	-	Diesel Generator
DRS	-	Division of Reactor Safety
ECCS	-	Emergency Core Cooling
EDG	-	Emergency Diesel Generator
EHC	-	Electro-Hydraulic Control
EnRad	-	Environmental Radiation
EOC	-	End-of-Cycle
ES	-	Engineered Safeguards
FDW	-	Feedwater
FME	-	Foreign Material Exclusion
FSAR	-	Final Safety Analysis Report
GPM	-	Gallons per Minute
HPI	-	High Pressure Injection
HPSW	-	High Pressure Service Water
HX	-	Heat Exchanger
ICS	-	Integrated Control
IP	-	Inspection Procedure
IR	-	Inspection Report
ISFSI	-	Independent Spent Fuel Storage Installation
ISI	-	Inservice Inspection
IST	-	Inservice Testing
KHU	-	Keowee Hydroelectric Unit
kV	-	Kilo Volt
LCO	-	Limiting Condition for Operation
LCT	-	Lee Combustion Turbine
LER	-	Licensee Event Report
LHRA	-	Locked-High Radiation Area
LOCA	-	Loss of Coolant Accident
LOOP	-	Loss of Offsite Power
LPI	-	Low Pressure Injection
LPSW	-	Low Pressure Service Water
MDEFW	-	Motor Driven Emergency Feedwater
MR	-	Maintenance Rule
MS	-	Main Steam
MSRV	-	Main Steam Relief Valve
MT	-	Magnetic Particle
MW	-	Monitoring Well
NCV	-	Non-Cited Violation
NDE	-	Non-Destructive Examination
NEI	-	Nuclear Energy Institute
NIST	-	National Institute of Standards and Technology
NRC	-	Nuclear Regulatory Commission
NRMCA	-	National Ready Mixed Concrete Association
NRR	-	Nuclear Reactor Regulation
NSD	-	Nuclear Site Directive
ODCM	-	Offsite Dose Calculation Manual
ODMI	-	Operational Decision Making Issue

ONS	-	Oconee Nuclear Station
OOS	-	Out-Of-Service
ORAM	-	Operational Risk Assessment Monitor
OTSG	-	Once-Through Steam Generator
PARS	-	Publicly Available Records
PASS	-	Post Accident Sampling System
PCM	-	Personnel Contamination Monitor
PDW	-	Plant Drinking Water
PI	-	Performance Indicator
PIP	-	Problem Investigation Process report
PM	-	Preventive Maintenance
PMT	-	Post-Maintenance Testing
PRA	-	Probabilistic Risk Assessment
PT	-	Liquid Penetrant Test
PWHT	-	Post Weld Heat Treatment
QC	-	Quality Control
RB	-	Reactor Building
RBCU	-	Reactor Building Cooling Unit
RBES	-	Reactor Building Emergency Sump
RBS	-	Reactor Building Spray
RCMUP	-	Reactor Coolant Makeup Pump
RCA	-	Radiologically Controlled Area
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
REMP	-	Radiological Environmental Monitoring Program
RETS	-	Radiological Environmental Technical Specifications
RFO	-	Refueling Outage
RG	-	Regulatory Guide
RII	-	Region II
RP	-	Radiation Protection
RPV	-	Reactor Pressure Vessel
RPVH	-	Reactor Pressure Vessel Head
RWF	-	Radwaste Facility
RTP	-	Rated Thermal Power
RV	-	Reactor Vessel
SAM	-	Small Article Monitor
SBLOCA	-	Small Break Loss of Coolant Accident
SCBA	-	Self-Contained Breathing Apparatus
SDP	-	Significance Determination Process
SG	-	Steam Generator
SGRP	-	Steam Generator Replacement Project
SLC	-	Selected Licensee Commitments
SRA	-	Senior Reactor Analyst
SRG	-	Safety Review Group
SRPMP	-	Standard Radiation Protection Management Procedure
SSC	-	Structure, System, and Component
SSF	-	Standby Shutdown Facility

TDEFW	-	Turbine Driven Emergency Feedwater
TI	-	Temporary Instruction
TLD	-	Thermoluminescent Dosimetry
TS	-	Technical Specification
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
UT	-	Ultrasonic Test
VDC	-	Volts-Direct Current
VAC	-	Volts-Alternating Current
WO	-	Work Order
WR	-	Work Request