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

January 30, 2006

Duke Energy Corporation (DEC)
ATTN.: Mr. B. H. Hamilton
Site Vice President
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION - INTEGRATED INSPECTION REPORT 05000269/2005005, 05000270/2005005, 05000287/2005005 AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION INSPECTION REPORT 72-04/2005001

Dear Mr. Hamilton:

On December 31, 2005, 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 January 5, 2006, with you and other members of your staff.

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

Based on the results of this inspection, the NRC has identified two self-revealing findings and two NRC identified findings of very low safety significance (Green), which were determined to be violations of NRC requirements. Additionally, licensee-identified violations, which were determined to be of very low safety significance, are listed in the this report. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these six findings as non-cited violations (NCVs) consistent with Section VI.A.1 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 facility.

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

Sincerely,

/RA/

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

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

Enclosure: NRC Integrated Inspection Report 05000269/2005005,05000270/2005005, 05000287/2005005 w/Attachment: Supplemental Information cc w\encl.:

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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 No:	50-269/2005005, 50-270/2005005, 50-287/2005005, 72-04/2005001
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	October 1, 2005 - December 31, 2005
Inspectors:	M. Shannon, Senior Resident Inspector A. Hutto, Resident Inspector E. Riggs, Resident Inspector J. Lenahan, Senior Reactor Inspector (Section 1R08) S. Vias, Senior Reactor Inspector (Section 1R08) P. VanDoorn, Senior Reactor Inspector (Section 4OA2.4) W. Loo, Senior Health Physicist (Sections 2OS1, 2OS2, 2PS2, 4OA1, 4OA5.7, 4OA5.8) H. Gepford, Health Physicist (Sections 2PS1, 2PS2, 2PS3, 4OA5.7, 4OA5.8) J. Kreh, Emergency Preparedness Inspector (Sections 2OS1, 2OS2)
Approved by:	Michael E. Ernstes, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IRs 05000269/2005005, 05000270/2005005, 05000287/2005005, 72-04/2005001; 10/01/2005 - 12/31/2005; Oconee Nuclear Station, Units 1, 2, and 3; Maintenance Effectiveness, Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems, Radiological Environmental Monitoring Program and Radioactive Material Control Program, and Other Ativities.

The report covered a three-month period of inspection by the onsite resident inspectors and announced regional-based inspections conducted by three reactor inspectors, two health physicists, and one emergency preparedness inspector. 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. NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A Green self-revealing non-cited violation was identified for failure to have adequate procedures for testing the Standby Shutdown Facility (SSF) diesel generator as required by Technical Specification (TS) 5.4.1. The licensee’s existing test procedures did not establish the appropriate plant conditions with the Unit 2 condenser cooling water (CCW) system shut down such that the water supply to the SSF auxiliary service water (ASW) and station ASW heated above 90 degrees F, rendering both unavailable for all three units. The licensee entered this finding into their corrective action program under Problem Investigation Process report (PIP) O-05-7479.

  This finding was considered to be of more than minor significance because it affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, as the elevated temperature of the SSF ASW and station ASW supply resulted in the unavailability of these systems. This issue was determined to be of very low safety significance based on the screening criteria found in MC 0609, Appendix A, Phase 1 SDP worksheet. More specifically, the total additional unavailability of the SSF (one day) as result of overheating the supply did not exceed the TS allowed outage time. (Section 1R12)

- Green. A NRC-identified non-cited violation of 10 CFR 50 Appendix B, Criterion XVI was identified for the failure to identify a condition adverse to quality, in that feedwater terminal ends had not been identified; and therefore, actions to mitigate the affects from a terminal end line break had not been implemented.

The licensee entered this finding into their corrective action program under PIP O-06-00138.

This finding was considered to be more than minor because an unprotected terminal end line break would impact the Reactor Safety Cornerstone for Mitigating Systems associated with the availability, reliability and function of systems needed to respond to a high energy line break (HELB). This issue was determined to be of very low safety significance based on a very low initiating event frequency being calculated as a result of the limited number of welds and feet of pipe under consideration. In addition, the large early release frequency impact was below the threshold, because of the size of break required to damage the containment penetration was an even lower probability event. This finding involved the cross-cutting aspect of Problem Identification and Resolution. (Section 4OA5.2)

Cornerstone: Public Radiation Safety

- Green. An NRC-identified NCV of 10 CFR 20.1302(a) was identified for failure to ensure surveys of particulate radioactive materials in effluents released to unrestricted areas by the unit vents were adequate to demonstrate compliance with dose limits for individual members of the public. The failure to conduct appropriate evaluations to assure representative sample collection from the Unit 1, 2, and 3 unit vent exhaust streams when sampled through the tee connections on the sample line to 1,2,3-RIA-43 and the elbow connections on the associated Selected Licensee Commitment required unit vent particulate sampler lines could result in inaccurate measurement of airborne particulate radionuclides in effluent samples, potentially leading to effluent releases exceeding allowed concentrations or dose limits to members of the public. This finding was entered into the licensee's corrective action program as PIPs O-04-7084 and O-05-4874. The licensee has approved and scheduled installation of a design modification for the monitors that will remove the non-conforming bends and replace them with bends of radius greater than or equal to five times the size of the diameter of the sample lines.

  This finding is greater than minor because it is associated with the program and process attribute of 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. This finding which involved radioactive material control was assessed using the Public Radiation Safety SDP. Since the finding did not result in the failure to assess dose, due to the licensee having other means by which dose from particulate releases could be assessed, and because the licensee did not exceed the limits in 10 CFR 50 Appendix I or 10 CFR 20.1301(a), it was determined to be of very low safety significance. This finding involved the cross-cutting aspect of Problem Identification and Resolution. (Section 2PS1)

- Green. A self-revealing non-cited violation of 10 CFR 20.1501(a) and 10 CFR 20.1802 was identified for an inadequate survey of contaminated equipment and failure to control and maintain constant surveillance of licensed material when free-released contaminated equipment was subsequently shipped to two locations as “clean” material without appropriate radiological controls. One of the locations was a non-licensed individual possessing neither the training nor equipment necessary to identify and control the contaminated material. The licensee entered the finding into the corrective action program as PIP O-04-8873. The corrective actions associated with this PIP included sending a radiological response team to one of the locations to identify, contain, and decontaminate any contaminated equipment and performing a detailed root cause analysis of the event.

  The finding is greater than minor because it is associated with the human performance attribute of 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 conduct adequate surveys resulted in the free-release of contaminated equipment, potentially leading to exceeding the dose limits to members of the public through loss of control of licensed material. This finding which involved radioactive material control was assessed using the Public Radiation Safety SDP. Since the finding neither resulted in an exposure to the public in excess of five millirem nor involved greater than five occurrences, it was determined to be of very low safety significance. The cause of this finding is related to the cross-cutting element of Human Performance. (Section 2PS3)

B. Licensee-Identified Violations

Two violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. These violations are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the report period at 100 percent rated thermal power (RTP) and operated at or near 100 percent RTP for the remainder of the inspection period.

Unit 2 began the report period at approximately 93 percent RTP. The unit was shutdown from approximately 73 percent RTP on October 22, 2005, and commenced the End-of-Cycle 21 (2EOC21) refueling outage. The unit was brought on-line on November 30, 2005, and achieved 100 percent RTP on December 2, 2005. The unit operated at or near 100 percent RTP for the remainder of the inspection period.

Unit 3 began the report period at 100 percent RTP. On November 1, 2005, the unit was reduced to approximately 50 percent RTP following a heater drain system transient. Subsequently, the unit was returned to 100 percent RTP on November 2, 2005. The unit operated at or near 100 percent RTP for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee’s preparations for the onset of seasonal cold weather. Specifically, the inspectors reviewed the completed operation’s Cold Weather Checklist contained in operating procedures OP/(1), (2) and (3)/1102/020 for outside temperature conditions expected to be below 35 degrees F. The checklists performed operational checks of heat trace circuits and associated alarm circuits associated with the condenser circulating water pumps (1), standby shutdown facility building heating (1), and borated water storage tank and associated piping system heating (3). The inspectors independently verified alarm circuits were not in alarm and localized heat trace circuits were providing heating. The inspectors verified that the freeze protection circuit checks were performed as necessary before any significant cold weather impacted the plant.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 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. 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 backup instrument air (IA) compressors with primary IA compressors out of service (OOS) for train maintenance

- A control room chiller with the B control room chiller OOS for the Unit 2 emergency power switching logic functional test

- Keowee Hydro Units (KHUs) 1 and 2, CT-4 and 5 with the standby shutdown facility (SSF) and station auxiliary service water (ASW) systems OOS due to dewatering of Unit 2 condenser circulating water (CCW) system to support replacement of 2C and 2D CCW inlet expansion joints and 2CCW-10 and 2CCW-11 valve seats

d. Findings

No findings of significance were identified.

.2 Complete Walkdown of the Unit 1 Emergency Feedwater System

a. Inspection Scope

The inspectors performed a system walkdown on accessible portions of the Unit 1 alternate Emergency Feedwater System (EFW) flowpath and its support system. This included the AC motor control and motor power supply cables. The inspectors focused on verifying proper valve and breaker positioning, power availability, no damage to piping or cable tray structural supports, and material condition. The inspectors also verified that system check valves were being tested by the inservice testing (IST) program.

Documents and drawings reviewed for this semi-annual inspection sample included:

- OP/1/A/1106/006, Emergency Feedwater System,
- OP/1/A/0600/018, Emergency Feedwater Train Operability
- OP/1/A/0251/014, Feedwater Check Valve Functional Test
- Technical Specification 3.7
- Updated Final Safety Analysis Report (UFSAR) Section; 7.4.3
- Selected Licensee Commitment; 16.7.3
- Drawings; OFD 121D-1.1, OFD 121B-1.3, and OFD 133A-2.5

A review of Problem Investigation Process reports (PIPs) and open maintenance work orders was performed to verify that material condition deficiencies did not significantly affect the SSF DG's ability to perform its design functions and appropriate corrective action was being taken by the licensee.

The inspectors conducted a review of the system engineer's trending data and system health reports to verify that appropriate trending parameters were being monitored and that no adverse trends were noted.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

Fire Area Walkdowns

a. Inspection Scope

The inspectors conducted tours in fifteen 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 areas were conducted during this inspection period:

- Unit 1and 2 Spent Fuel Cooler Room (1)
- Unit 1, 2 and 3 Equipment Rooms (3)
- Unit 2 Cable Room (1)
- SSF pump room, diesel room, ventilation room, and control room (4)
- Unit 1, 2 and 3 Control Rooms (3)
- Unit 1, 2 and 3 Turbine Building Operating Level (3)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

Internal Flooding (Turbine Building)

a. Inspection Scope

The inspectors reviewed PIP O-05-2809, which documents the discovery of a bulge in the recently installed, internal sealing expansion joint on 1CCWMJ0010. The expansion joint was installed during the previous refueling outage (RFO), 1EOC-21, and was being inspected for the first time since its installation approximately 18 months earlier as a backup to the rayon reinforced rubber expansion joints that have been inservice for approximately 30 years. Engineering was contacted about the inspection observation, as was the expansion joint manufacturer. The remaining Unit 1 and 2 condenser inlet and outlet water box internal expansion joints were inspected satisfactorily, and the licensee plans to inspect Unit 3 during the Spring 2006 RFO.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

Annual Review

a. Inspection Scope

The inspectors reviewed the Unit 2 reactor building cooling unit (RBCU) Performance Test, PT/0/A/0160/006. The testing was performed to verify that the RBCU cooling capacity met Technical Specification (TS) and design basis requirements following the installation of tube end sleeves on the RBCU heat exchanger due to tube wall thinning concerns. The tube end sleeves were installed in accordance with Heat Exchanger - Tube Sleeving Installation Using Vendor Document Number 6037084A, MP/0/A/100/024. The inspection focused on compliance with procedural requirements, appropriate data collection during the testing, satisfactory performance of the RBCU, and its ability to perform its design function during a postulated event.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

.1 Inservice Inspection (ISI)

a. Inspection Scope

The inspectors reviewed ISI procedures, observed in-process ISI work activities, and reviewed selected ISI records. The observations and records were compared to the TSs and the applicable Code (ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition, 2000 Addenda) to verify compliance and to ensure that examination results were appropriately evaluated and dispositioned. This was the first outage of the first period of the fourth ten-year interval.

The inspectors observed non-destructive examination (NDE) activities. Specifically, liquid penetrant (LPT) examination of pipe to elbow weld numbers 2-53A-8-48 and -50 on the high pressure injection piping, and visual examination (VT) and ultrasonic examination (UT) of reactor vessel studs, numbers 1 through 12, and VT for the nuts and washers. In addition, the inspectors examined snubbers, spring cans and pipe supports during a walkdown of the Unit 2 containment. The inspectors reviewed the licensee's evaluation of an indication identified in 1996 on pressurizer weld 2-PZR-WP76.

The inspectors reviewed records of the above inspections including calibrations, equipment certifications, consumable certifications, and personnel qualifications. The inspectors also reviewed the licensee's evaluation of indications identified in 1996 on pressurizer weld 2-PZR-WP76.

The inspectors also reviewed records documenting welding activities associated with Work Orders (WO) 98603735 and 98705438 for replacement of Class 2 valves on the high pressure injection system. The inspectors also reviewed WO 98670064 and 98711484 for replacement of Class 1 valves in the reactor coolant system. The records were reviewed to determine if the welding process and examinations were performed in accordance with ASME Section XI repair/replacement requirements. The inspectors reviewed drawings, work instructions, weld process sheets, and weld travelers.

The inspectors reviewed implementation of the licensee's Boric Acid Corrosion Control program to determine if commitments made in response to Generic Letter 88-05 and Bulletin 2002-01 were being effectively implemented. The inspectors reviewed a summary of boric acid leakage screening reports, evaluations, work orders, and corrective actions. The inspectors examined various components during walkdowns inside the containment to verify the leaks were properly assessed and corrective actions were implemented.

A sample of ISI issues in the licensee's corrective action program was reviewed to confirm that problems were being identified and placed in the corrective action program, and appropriate corrective actions were being initiated.

b. Findings

No findings of significance were identified.

.2 Steam Generator (SG) Tube ISI

a. Inspection Scope

From October 31 - November 3, 2005, the inspectors reviewed the Unit 2 SG (2EOC21) tube examination activities conducted pursuant to TS and the ASME Code Section XI requirements.

The inspectors reviewed the SG examination scope, eddy current testing (ET) acquisition procedures, ET analysis procedures, the SG Operational Assessment, records and examination reports to confirm that:

- The SG tube ET examination scope was sufficient to identify tube degradation confirming that the ET scope completed was consistent with the licensee's procedures, plant TS requirements and ASME Code Section XI requirements.

Additionally, the inspectors reviewed the SG tube ET examination scope to determine that it was consistent with that recommended in EPRI “Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6" and included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions and support tube plates.

- The licensee adequately followed-up on a new tube degradation mechanism as discussed in the SG tube degradation assessment.

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

- The ET probes and equipment configurations used to acquire ET data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H “Performance Demonstration for Eddy Current Examination” of EPRI “Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6."

- The licensee adequately examined for loose parts indications and performed secondary visual inspections at various locations in the steam generators.

- The licensee adequately evaluated for any contractor deviations from their ET data acquisition or analysis procedures or EPRI “Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6." The inspectors verified and reviewed that any deviations identified in this area were properly documented in the licensee’s corrective action system.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator simulator training on November 29, 2005. The scenario involved a tornado striking the north side of the turbine building, resulting in a loss of main condenser vacuum, a loss of main feedwater, a main turbine trip and a reactor trip. The simulated event was complicated by the failure of the 1A motor driven emergency feedwater (MDEFW) pump and the turbine driven emergency feedwater (TDEFW) pump to start. While attempting to cross-connect emergency feedwater from the simulated Unit 3, the turbine bypass valves were rendered inoperable due to the loss of condenser vacuum; consequently, the unit’s atmospheric dump valves were utilized to steam the 1B SG. The 1B MDEFW pump then failed, causing the operating crew to initiate high pressure injection (HPI) forced cooling of the unit’s core and declare an Alert. 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 TS actions and properly classify the simulated event.

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:

- Unit 2 Reactor Building Coating Remediation Project. During the Unit 2 refueling outage the licensee made extensive repairs to the containment liner between the containment building 4th floor and containment ventilation ducting.

- SSF Extended Maintenance Outage

b. Findings

Introduction: A Green self-revealing non-cited violation (NCV) was identified for failure to have adequate procedures for testing the SSF diesel generator as required by TS 5.4.1. The licensee's existing test procedures did not establish the appropriate plant conditions with the Unit 2 CCW system shut down such that the water supply to the SSF ASW and station ASW heated above 90 degrees F, rendering both unavailable for all three units.

Description: On October 24, 2005, the licensee commenced a SSF maintenance outage to coincide with outage related Unit 2 CCW maintenance that required the draining of the CCW condenser inlet piping. On November 8, 2005, the licensee discovered during operator rounds that the operating SSF heating ventilation and air conditioning (HVAC) service water supply was greater than 300 psig and that the service water supply temperature was 100 degrees F; thereby, exceeding the SSF ASW and station ASW operability limit of 90 degrees F. Earlier on November 8, 2005, the licensee performed post maintenance testing on the SSF emergency diesel (EDG), which included loaded run time totaling approximately four hours. The CCW supply piping had been filled; however, the system had not been restored to the point to allow a CCW pump to be started. Consequently, the diesel runs were performed without any CCW flow. Normal diesel cooling water is taken from the CCW supply header and returned back to the header. AP-25, SSF Operation, requires that diesel cooling return be diverted to the storm drains to prevent overheating the supply, as CCW flow would not be anticipated during conditions that required the SSF. The EDG test procedures did not require cooling flow to be diverted, as they are typically performed when the CCW system is in service. The licensee did not anticipate the need to divert the cooling water return or provide a CCW flow path while performing the EDG test procedures. Therefore, the licensee did not establish adequate instructions or prerequisites in the test procedures, or implement operating procedures to establish and maintain the CCW supply to the SSF ASW and station ASW in an operable status. The SSF ASW and station ASW were returned to available status the following day after venting the CCW supply and reducing temperature below 90 degrees F.

Analysis: This finding was considered a performance deficiency because it is expected that the licensee will implement test procedures that are compatible with existing plant conditions. This finding was considered to be of more than minor significance because it affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, as the elevated temperature of the SSF ASW and station ASW supply resulted in the unavailability of these systems. This issue was determined to be of very low safety significance (Green) based on the screening criteria found in MC 0609, Appendix A, Phase 1 SDP worksheet. More specifically, the total additional unavailability of the SSF (one day) as result of overheating the supply did not exceed the TS allowed outage time.

Enforcement: Technical Specification 5.4.1 requires that written procedures shall be established, implemented and maintained in accordance with Regulatory Guide (RG) 1.33, Revision 2, Appendix A, dated February 1978. RG 1.33 Appendix A specified that procedures are required for operation and testing of safety-related PWR systems including Emergency Power and Auxiliary Feedwater. Contrary to the above, the licensee did not establish adequate instructions or prerequisites in the test procedures, or implement operating procedures to establish and maintain the CCW supply to the SSF ASW and station ASW in an operable status. Because this finding was of very low safety significance and was entered into the licensee's corrective action program (PIP O-05-7479), this issue is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000269,270,287/2005005-01, Inadequate Procedures for Testing the SSF Diesel Generator with the CCW Supply Secured.

1R13 Maintenance Risk Assessment and Emergent Work Evaluations

a. Inspection Scope

The inspectors evaluated the following attributes for the **seven** 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-05-7439, ORAM Risk Condition was Documented as Yellow, but Discovered to be Orange
- PIP O-05-7479, Unable to Restore the SSF Following SSF DG Run which heated Unit 2 CCW inlet piping to greater than the allowable limit, resulting in unexpected SSF unavailability and an orange ORAM risk condition
- PIP O-05-7516, Unable to restore the Station ASW System following SSF DG Run which heated Unit 2 CCW inlet piping to greater than the allowable limit, resulting in unexpected station ASW pump unavailability and a red ORAM risk condition
- Orange ORAM risk condition, high pressure service water supply to HPI cooling flow detector replacement
- Orange ORAM risk condition, Complex Plan for low pressure injection (LPI) suction relief modification
- Yellow ORAM risk condition, Keowee Lake level < 793.7 feet (Complex Plan)
- Unit 2 ‘J’ Test was performed with SSF OOS, creating an orange ORAM risk condition

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:

- PIP O-05-8397, Greater than 25 Percent of the Oconee Nuclear Site Siren System was Inoperable Due to the Ice Storm

- PIP O-05-7164, Loss of 3D1 Heater Drain Pump (HDP) and Power Reduction to 85 Percent RTP, and PIP O-05-7169, Loss of 3D2 HDP and Power Reduction to 50 Percent RTP

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

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 operation (LCOs). The inspectors reviewed the following operability evaluations:

- PIP O-05-5496, Keowee Hydroelectric Unit (KHU) Main Step-up Transformer Cooling Fan Contactor Failed to Realign When KHUs Were Realigned

- PIP O-05-6829, Foreign Material Found in 2LP-19 and 20 Suction Piping from the RBES

- PIP O-05-6995, Retaining Clip Associated with the SSF Diesel Engine Fuel Injector Rocker Assembly was Found with an Ear Missing

- PIP O-05-7106, Water, Ferrous Wear Material, Metal, and Other Material Found in the Unit 2 TDEFW Pump Inboard Bearing Oil

- PIP O-05-7221, Required Number of SSF Controlled Pressurizer Heaters Needed to Offset Pressurizer Ambient Heat Losses Has Changed, Thereby Changing TS 3.10.1 Bases

b. Findings

Introduction: An unresolved item (URI) was identified for inadequate foreign material exclusion (FME) controls which resulted in the introduction of a large, hex nut and a construction nail in the A and B reactor building emergency sump (RBES) suction lines. 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.

Description: On October 24 and 26, 2005, with Oconee Unit 2 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 LPI and RBS pumps prior to installing a new, larger RBES. The licensee discovered a large, rusty nail inside of the LPI piping leading from the RBES to 2LP-19 and a large, rusty, hex nut inside of the LPI piping leading from the RBES to 2LP-20. PIP O-05-6829 documents the discovery of the foreign material. On November 11, 2005, the foreign material was removed during a post-modification FOSAR. The nail was a 16d construction nail, which is 3.5 inches in length and 0.1620 inches in diameter while the hex nut was approximately 2 inches across the corners of the nut and 1 inch thick.

The licensee's investigation of this event is documented in the problem evaluation portion of PIP O-05-6829 and states that, "The inappropriate act was foreign material was introduced into the lines associated with 2LP-19 and 2LP-20. The lines have never been inspected since the time of construction (over 30 years). An equipment problem could have resulted from the foreign material in the lines. The lines provide the recirculation path for LPI to RBS. The apparent cause is foreign material entered the system and was not removed. The cause is indeterminable due to there is no way to determine when the material entered the system." Furthermore, the Maintenance Rule portion of PIP O-05-6829 documents the licensee's decision to classify this event as a functional failure for the LPI and RBS systems. "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 [maintenance preventable functional failure]. The failure is related to a lack of FME control which is controlled by procedures and processes."

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 05000270/2005005-02, Inadequate Foreign

Material Exclusion Controls for the A and B Train Reactor Building Emergency Sump Suction Lines. This issue is in the licensee's corrective action program as PIP O-05-6829.

1R16 Operator Work-Arounds

.1 Semi-Annual Review of the Cumulative Effects of Workarounds

a. Inspection Scope

The inspectors performed a cumulative review of existing operator work-arounds to determine any change from the previous review. The review also considered the effect of the work-arounds on the operators' ability to implement abnormal or emergency operating procedures. The inspectors periodically reviewed PIPs and held discussions with operators to determine if any conditions existed that should have been identified by the licensee as operator work-arounds.

b. Findings

No findings of significance were identified.

.2 Risk Significant Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed one risk significant operator work-around to determine if the functional capability of the respective system or the human reliability in responding to an initiating event were affected. The inspectors specifically evaluated the effect of the operator workarounds on the ability to implement abnormal or emergency operating procedures. The inspectors also assessed what impact it would have on the unit if the work-around could not be properly performed.

- The work-around reviewed was documented in PIP O-05-8049, 2HP-15 Moore Controller Will Not Allow Automatic Feed and Bleed Operations. The malfunction of the 2HP-15 controller in automatic caused the control room operators to make RCS inventory additions with 2HP-15 in manual. Troubleshooting revealed a loose connection on the controller's circuit card which was promptly repaired. While 2HP-15 was inoperable in automatic, reactivity control was maintained by the use of initial and final bleed holdup tank and letdown storage tank levels.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed two 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:

- NSM ON-13103, Unit 1 and 2 Control Room Booster Fan Intake Modification
- NSM ON-23106, Reactor Building Emergency Sump Return Line Modification (Unit 2)

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (PMT)

a. Inspection Scope

The inspectors reviewed PMT procedures and/or 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 six tests:

- WO 98732127 (Remove and Replace the 2A Letdown Cooler) and MP/0/A/1720/017 (System Leakage Test Controlling Procedure)
- PT/2/A/0600/012, Unit 2 TDEFW Pump Test following Unit 2 RFO
- PT/2/A/0600/018, Unit 2 Emergency Feedwater (EFW) Train Operability Test following Unit 2 RFO

- PT/0/A/0400/011, SSF Diesel Generator Test following the refilling of the Unit 2 CCW inlet header after the replacement of 2C and 2D CCW inlet expansion joints and 2CCW-10 and 2CCW-11 valve seats

- PT/3/A/0400/007, SSF RC Makeup Pump Test, following increase in bearing vibration.

- PT/0/A/0711/001, Unit 2 Zero Power Physics Test

b. Findings

No findings of significance were identified.

1R20 Refueling & 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 October 22, 2005 and December 2, 2005, the following activities related to the Unit 2 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

Routine Surveillance

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of the six risk-significant SSCs listed below, to assess, as appropriate, whether the SSCs met TS, UFSAR, and licensee procedure 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/2/A/0610/001 A, EPSL Normal Source Voltage Sensing Circuit
- PT/2/A/0151/019, Penetration 19 Leak Rate Test (Containment Leak Rate)
- PT/2/A/0251/019, Main Steam Atmospheric Dump Valve Functional Test
- IP/0/B/0276/002, ATWS Mitigation System AMSAC/DSS Logic
- Ultrasonic thickness measurement of Unit 2 east penetration room feedwater piping welds 2FDW-225-22B, H03.001.001, and H03.001.002.
- PT/1/A/0600/012, Unit 1 TDEFDW 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 selected 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 two items were reviewed under this inspection procedure:

- Unit 2 RBCU Heat Exchanger - Tube Sleeving Installation Using Vendor Document Number 6037084A, MP/0/A/100/024.

- Unit 1 Control Battery 1CA Single Battery Charger Installation

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

Emergency Preparedness Drill

a. Inspection Scope

The inspectors observed and evaluated the emergency preparedness drill held on March 23, 2005. The scenario began with a tornado warning being issued by the National Weather Service for Oconee County, causing the SSF to be manned. A simulated tornado struck the northwest side of the site, resulting in a tornado-generated missile piercing the simulated unit's borated water storage tank. Consequently, an Alert was declared and a site assembly was initiated. A second tornado passed over the site tearing away a portion of the unit's turbine building roof and exposing the unit's vital, 4160 V switchgear to excessive amounts of rainwater. The unit's TC, TD and TE load centers lost power due to the electrical faults resulting from the water intrusion into the turbine building. A Site Area Emergency was declared, and the SSF reactor coolant makeup pump began providing reactor coolant makeup flow. The SSF diesel generator failed, and a General Emergency was declared. The simulated events of the drill were properly classified, and appropriate notifications were promptly made to the local authorities, the state, and the NRC.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational and Public Radiation Safety

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Control - Licensee program activities for monitoring workers and controlling access to radiologically significant areas and tasks were inspected. The inspectors evaluated procedural guidance; directly observed implementation of administrative and established physical controls; assessed worker exposures to radiation and radioactive material; and appraised radiation worker and technician knowledge of, and proficiency in, the implementation of Radiation Protection (RP) program activities.
During the inspection, radiological controls for ongoing refueling activities for Unit 2 (U2) were observed and discussed. Reviewed tasks included steam generator nozzle dam preparation, installation and testing, “A” letdown cooler removal, “A” and “B” steam generator (SG) eddy current testing, and “A” SG tube pulls. In addition, licensee controls for selected tasks scheduled and ongoing during the refueling outage were assessed. The evaluations included, as applicable, Radiation Work Permit (RWP)

details; use and placement of dosimetry and air sampling equipment; electronic dosimeter set-points; and monitoring and assessment of worker dose from direct radiation and airborne radioactivity source terms. Effectiveness of established controls was assessed against area radiation and contamination survey results and occupational doses received. Physical and administrative controls and their implementation for extra (locked) high radiation areas (EHRA) and very high radiation areas (VHRA) were evaluated through discussions with cognizant licensee representatives, direct field observations, and record reviews.

Occupational workers' adherence to selected RWPs and Health Physics Technician (HPT) proficiency in providing job coverage were evaluated through direct observations of staff performance during job coverage and routine surveillance activities, review of selected exposure records, and interviews with cognizant licensee staff. Radiological postings and physical controls for access to designated high radiation area (HRA) and EHRA locations within the U2 Reactor Building (RB), Units 1, 2, and 3 Auxiliary Building, and Spent Fuel Pool Floor areas were evaluated during facility tours. In addition, the inspectors independently measured radiation dose rates and evaluated established posting and access controls for selected Units 1, 2, and 3 Auxiliary Building locations. Occupational doses associated with direct radiation exposure and potential radioactive material intakes were reviewed and discussed with cognizant licensee representatives.

RP program activities were evaluated against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; Updated Final Safety Analysis Report (UFSAR) details in Section 12, Radiation Protection; and approved licensee procedures. Licensee procedures, guidance documents, records, and data reviewed within this inspection area are listed in Section 2OS1 of the report Attachment.

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

The inspectors completed 21 of the required 21 samples for Inspection Procedure (IP) 71121.01. All samples have now been completed for this IP.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA) - Implementation of the licensee's ALARA program during the Unit 2 End of Cycle 21 outage was observed and evaluated by the inspectors. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for outage work tasks expected to incur the maximum collective exposures. Reviewed activities included SG nozzle dam preparation, installation and testing; SG eddy-current testing; SG channel head wash; and SG tube pull. Incorporation of planning, established work controls, industry lessons learned, and expected dose rates and dose expenditure into the ALARA pre-job briefings and RWPs for these activities were reviewed. The inspectors made direct and/or closed-circuit video observations of these activities (including selected pre-job briefings and post-job evaluations) while evaluating the licensee's use of engineering controls, low-dose waiting areas, and on-the-job supervision. The inspectors reviewed the licensee's exposure tracking system to determine whether it adequately supported control of collective exposures. These elements of the ALARA program were evaluated for consistency with the methods and practices delineated in applicable licensee procedures.

Selected elements of the licensee's source term reduction and control program were examined to evaluate the effectiveness of the program in supporting implementation of the ALARA program goals. Shutdown chemistry program implementation and the resultant effect on containment and auxiliary building dose-rate trending data were reviewed and discussed with ALARA staff.

Trends in individual and collective personnel exposures at the facility were reviewed. The plant collective exposure history for the years 2002-2004, based on the data reported to the NRC pursuant to 10 CFR 20.2206(c), was reviewed and discussed with ALARA staff, as were established goals for reducing collective exposure. Long-term trends in the plant's three-year rolling average collective exposure history, outage, non-outage and total annual doses for selected years were reviewed and discussed with licensee representatives. Records of year-to-date individual radiation exposures sorted by work groups were examined for significant variations of exposures among workers. The inspectors examined the dose records of the eight declared pregnant workers from January 2004 to November 2005 to evaluate total or current gestation dose. The applicable RP procedure was reviewed to assess licensee controls for declared pregnant workers.

The licensee's ALARA program implementation and practices were evaluated for consistency with UFSAR Chapter 12, Sections 12.1 (ALARA) and 12.4 (Radiation Protection Program); 10 CFR Part 20 requirements; Regulatory Guide 8.29, Instruction Concerning Risks from Occupational Radiation Exposure, February 1996; and licensee procedures. Documents reviewed during the inspection of this program area are listed in Section 2OS2 of the report Attachment.

Problem Identification and Resolution - The inspectors reviewed CAP documents listed in Section 2OS2 of the report Attachment that are related to the ALARA program. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with NSD 208, Problem Investigation Process (PIP), Rev. 27.

The inspectors completed 15 of the required 15 samples and 10 of the 14 optional samples for IP 71121.02. All required samples have now been completed for this IP.

b. Findings

No findings of significance were identified.

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Effluent Monitoring and Radwaste Equipment - The inspectors walked-down the unit vent monitor skid on all three units (1,2,3-RIA-43,44,45,46), evaluating the physical condition and design of the skid. The inspectors payed particular attention to the use of elbows and tee connections on the particulate sampling lines. The inspectors reviewed PIP reports, a design modification, and a report associated with the configuration of the sample tubing for the unit vent particulate monitoring and sampling station as they related to the ability to perform representative sampling. These documents and observations were discussed in detail with cognizant licensee representatives.

Effluent Release Processing and Quality Control Activities - The inspectors observed performance of sampling and analysis of the U2 RB purge and subsequent generation of a gaseous waste release permit. In addition, the inspectors reviewed gaseous waste permits, including projected public dose assessments, for select batch and continuous gaseous releases made during Calendar Year 2005.

The inspectors completed one of the required ten samples, resulting in closure of IP 71122.01 that was opened in NRC Inspection Report Number 50-0269, 270, 287/2004-004 with nine of ten samples completed. All samples have now been completed for this IP.

b. Findings

Introduction: An NRC-identified Green NCV of 10 CFR 20.1302(a) was identified for failure to ensure surveys of particulate radioactive materials in effluents released to unrestricted areas by the unit vents were adequate to demonstrate compliance with dose limits for individual members of the public. This issue was initially identified as an Unresolved Item following an onsite inspection in September 2004.

Description: During field observations of the Unit Vent Particulate Effluent Radiation Monitors (1,2,3-RIA-43) and the associated fixed-filter particulate samplers, the inspectors questioned the adequacy of the sample lines to provide representative particulate samples. Specifically, the inspectors noted a tee connection on the sample lines to 1,2,3-RIA-43 and an elbow connection on the associated Selected Licensee Commitment (SLC) required unit vent fixed-filter particulate sampler lines. The unit vents are required to be continuously monitored for particulates by SLC Table 16.11.3-2. The inspectors noted that the observed connections were not in accordance with acceptable industry practices regarding sample lines as outlined in American National Standards Institute (ANSI) N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities, which states that elbows should be avoided and that inlet sample lines should have bends with radii that are five times the diameter of the sample line as a means to ensure that representative particulate samples are obtained from an exhaust stack.

In February 1998, the licensee initiated a PIP report following NRC-identification of a 90-degree bend on the inlet tubing on 2RIA-43,44,45,46. One of the corrective actions was to inspect the inlet piping/tubing for all other radiation monitors which measured particulate and iodine to ensure no 90-degree bends or abrupt changes existed. The inspection failed to identify the tee connections on 1,2,3-RIA-43 or the elbows on the fixed-filter sampler lines.

Review of records and discussions with cognizant licensee personnel determined that the tee connections on 1,2,3-RIA-43 and the auxiliary sampling station were initially identified as not meeting the isokinetic requirements of ANSI N13.1-1969 in April 2003. At that time, it was determined that there were no operability concerns with respect to 1,2,3-RIA-43, there were no design basis requirements or other requirements within the UFSAR with respect to 90-degree bends, and that Oconee was not committed to ANSI N13.1-1969. The conclusion of the licensee's review of the condition was that ANSI N13.1-1969 was good practice but that it would not be cost effective to revise the sample station tubing. The evaluation failed to recognize that Oconee is committed to Regulatory Guide 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, which references ANSI N13.1-1969.

The licensee again reviewed the tee connections on 1,2,3-RIA-43 and the elbow connections on the SLC required particulate sampler lines following the identification of this concern as Unresolved Item 05000269,270,287/2004004-03 in September 2004. At that time, the licensee performed an operability assessment and determined that the unit vent sampling program was operable but non-conforming. The operability assessment dealt primarily with the fixed particulate filter assembly located on the RIA-43 skids, the SLC required sampler. Review of the operability assessment by the inspectors determined that it was based on primarily three sources of data: (1) a particle-size study performed at Pilgrim Nuclear Station; (2) high efficiency particulate air (HEPA) filtration of particles 0.3 microns in diameter and larger for some of the inputs to the unit vent; and (3) Radiological Environmental Monitoring Program (REMP) data. The inspectors determined that the evaluation failed to demonstrate the adequacy of particulate measurements from the unit vent monitors/samplers. Specifically, the Pilgrim particle-size study data cannot be applied directly to Oconee; Pilgrim is a boiling water reactor and Oconee is a pressurized water reactor. The assumption of HEPA filtration of large particles did not address the historical performance of the HEPA system with respect to leakage or failure. Also, the review of annual REMP reports focused on dose calculations from all gaseous effluent sources, not particulate alone, and the numerical values were so small that direct comparison with effluent dose calculations revealed little useful information.

The licensee has approved and scheduled installation of a design modification for the monitors that will remove the non-conforming bends and replace them with bends of radius greater than or equal to five times the size of the diameter of the sample lines.

Analysis: The inspectors noted that a failure to adequately evaluate the effect of the observed tee connections on the sample lines to 1,2,3-RIA-43 and the elbow connections on the associated SLC required unit vent particulate sampler lines on the representativeness of particulate sampling is a performance deficiency. The licensee is expected to make surveys of radioactive materials in effluents released to unrestricted areas. Identifying and assessing the affect of the 90-degree bends was reasonably within the licensee's ability to foresee and correct. The finding is greater than minor because it is associated with the program and process attribute of 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 conduct appropriate evaluations to assure representative sample collection from the Units 1, 2, and 3 unit vent exhaust streams using 1,2,3-RIA-43 and the particulate vent sampler lines could result in inaccurate measurement of airborne particulate radionuclides in effluent samples. The finding was evaluated using the Public Radiation Safety SDP. This issue was related to the effluent release program and potentially resulted in an impaired ability to assess dose but did not result in the failure to assess dose, as the licensee had other means by which dose from particulate releases could be assessed, and the licensee did not exceed the limits in 10 CFR 50 Appendix I or 10 CFR 20.1301(a). For these reasons, the SDP evaluation concluded that the issue is of very low safety significance. This finding was entered into the licensee's corrective action program as PIPs O-04-7084 and O-05-4874. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, in that the licensee had multiple opportunities to identify and address the elbow/tee connections in the particulate sampling lines following installation of the skids around 1990 and subsequent to NRC-identification of 90-degree bends in the inlet tubing upstream of the 2RIA-43, 44, 45, 46 skid in February 1998.

Enforcement: 10 CFR 20.1302(a) requires licensees to make surveys of radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public. Contrary to the above, at the time of the inspection a comprehensive evaluation of the effect of 90-degree bends on the particulate sampling lines had not been performed. The licensee therefore was not assured that the unit vent particulate measurements obtained using 1,2,3-RIA-43 and the associated SLC required sampler were accurate. Because the failure to comply with 10 CFR 20.1302(a) is of very low safety significance and has been entered into the licensee's corrective action program (PIPs O-04-7084 and O-05-4874), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000269,270,287/2005005-03, Failure to Ensure Adequacy of Measurements of Particulate Effluents Released from Unit Vent.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization - The inspectors evaluated licensee methods for processing and characterizing radioactive waste (radwaste). Inspection activities included direct observation of processing equipment for solid and liquid radwaste and evaluation of waste stream characterization data.

Solid and liquid radwaste equipment was inspected for material condition, configuration compliance with the UFSAR, and consistency with Process Control Program (PCP) requirements. The inspectors reviewed the status of non-operational or abandoned in place radwaste equipment. The inspectors reviewed the licensee's administrative and physical controls of non-operational or abandoned in place radwaste equipment to prevent unmonitored releases, impact operating systems or contribute to unnecessary personnel exposure. Inspected equipment included liquid radwaste hold-up tanks, resin transfer piping, filters, dewatering facilities, and elements of the Powdered Resin Recovery System. The inspectors discussed system changes, component function, and equipment operability with licensee staff. In addition, procedural guidance for resin transfer was evaluated and compared with current equipment configuration. The inspectors also observed performance of High Integrity Container dewatering activities, chemical addition and processing of Waste Feed Tank C, and sampling of Waste Monitor Tank A by radwaste chemistry personnel.

Licensee radionuclide characterizations for selected waste streams were reviewed and discussed with radwaste staff. For primary bead resin, demineralizer bead resin, Powdex resin, cartridge filters, and dry active waste (DAW), the inspectors evaluated analyses for hard-to-detect nuclides and appropriate use of scaling factors. Comparison results between licensee waste stream characterization data and outside laboratory data were reviewed for the waste streams using the data maintained in the 10 CFR Part 61 Waste Classification and Waste Form Implementation Program Manual and sample data dated November 10, 2004. For selected shipment records, waste classification calculations were performed and the methodology used for resin waste stream mixing and concentration averaging was evaluated. The inspectors also interviewed cognizant radwaste staff and reviewed procedural guidance to evaluate the licensee's program for monitoring changing operational parameters.

Radwaste processing activities were reviewed for consistency with the licensee's PCP, Rev. 14, and UFSAR Chapter 11. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 61.55 and guidance provided in the Branch Technical Position on Waste Classification and Waste Form, 1983. Reviewed documents are listed in Section 2PS2 of the report Attachment.

Transportation - The inspectors evaluated the licensee's activities related to transportation of radioactive material. The evaluation included direct observation of shipment preparation activities and review of shipping-related documents.

The inspectors directly observed transportation activities including shipment packaging, surveying, blocking and bracing, vehicle placarding, vehicle checks, emergency instructions, preparation of disposal manifest, and the provision of shipping papers and special instructions to drivers. Specifically, the inspectors observed outgoing shipments of contaminated equipment, DAW, and reactor coolant system samples.

As part of the document review, the inspectors evaluated a number of shipping records for consistency with licensee procedures and compliance with NRC and Department of Transportation regulations. The shipments reviewed included Powdex resin, dewatered ion exchange resin, dewatered primary filter media, reactor coolant pump seals, and control rod samples. These shipments included LSA-II, Type A, and Type B packages, as well as an international shipment. In addition, training records for individuals currently qualified to ship radioactive material were checked for completeness and the training curriculum provided to these workers was evaluated. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Parts 20 and 71, 49 CFR Parts 170-189; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H.

Problem Identification and Resolution - The inspectors reviewed the licensee's corrective action documents and self-assessments related to radioactive material processing and transportation areas, to determine if problems were identified and entered in the system for resolution. Specifically, the inspectors reviewed PIP reports and interviewed cognizant licensee personnel to determine if problems were identified, properly characterized, prioritized, evaluated and corrected in accordance with licensee procedure NSD 208, Problem Investigation Process (PIP), Rev. 27. Reviewed documents are listed in Section 2PS2 of the report Attachment.

The inspectors completed six of the required six samples for IP 71122.02. All samples have now been completed for this IP.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors followed up on an event involving the loss of control of radioactive material by the licensee.

b. Findings

Introduction: A self-revealing Green NCV of 10 CFR 20.1501(a) and 10 CFR 20.1802 was identified for inadequately surveying contaminated equipment and failing to control and maintain constant surveillance of licensed material when free-released contaminated equipment was subsequently shipped to two locations as “clean” material without appropriate radiological controls.

Description: In November 2004, Oconee began demobilization of equipment used for the Unit 3 (U3) steam generator replacement (SGR). The process involved removing equipment from containment, performing initial surveys to determine contamination levels and locations, decontaminating the equipment as necessary, re-surveying the equipment, and releasing the equipment if the fixed contamination on the equipment was less than 100 corrected counts per minute (ccpm).

On December 17, 2004, Oconee RP was notified by Arkansas Nuclear One (ANO) that clean material shipped from Oconee to support their outage was surveyed and found to be contaminated. The materials had been at ANO for approximately two weeks before Oconee was notified. Surveys of cavity deck plates performed by ANO indicated levels of fixed contamination ranging from 60 ccpm to 400 ccpm and identified a hot particle reading 4,000 ccpm. The particle was analyzed and determined to consist of 14.1 nanocuries of Cesium-137, Cobalt-60, and Cesium-134. In addition, an auxiliary crane hoist showed fixed activity up to 400 ccpm.

On December 20-21, 2004, Oconee RP personnel conducted follow-up onsite surveys of other SGR equipment that had been released from the protected area. This effort identified cavity decking handrails with fixed contamination and superheat control panels with fixed contamination up to 6,000 ccpm and internal removable contamination.

On December 29, 2004, a team of HPTs was dispatched from Oconee to Mammoet in Rosharon, TX. Mammoet supplied heavy equipment to Oconee for the U3 SGR and did not possess a license for possession or use of radioactive material. Downender/ upender pieces were found to have contamination ranging from 600 to 700 ccpm. Additionally, beams and support stands were identified with fixed contamination levels ranging from 100 ccpm to 2,000 ccpm. All items were decontaminated by the team and the contamination was shipped back to Oconee for disposal. Members of the team indicated that some of the contaminated materials were readily identified by frisking, others had areas that had been circled as contaminated which were still contaminated, and still others had areas where an experienced HPT should have been able to find contamination.

In February 2005, the licensee began implementation of Interim Procedural Guidelines for Release of SGR Equipment and Materials. On February 21, fixed contamination was identified on the steering wheel of a forklift used by the Steam Generator Team (SGT). Over the next two months, contamination was identified on several pieces of SGR equipment that had been released from the protected area, ranging from 100 ccpm to 3,000 ccpm. In addition, a hot particle of 800 ccpm was identified and removed from a wooden deck plate.

A root cause evaluation of this event was performed by the licensee. The investigation reviewed personnel qualifications, procedures and processes, techniques, instruments used by SGT and Oconee RP personnel, and oversight provided in the release of the SGR equipment. The investigation included review of training records, survey instrument calibrations, survey documentation of released equipment, and interviews with HPTs, supervisors, and managers. The licensee determined that worker training and instrumentation were adequate for the task. However, for some material released, there were no records available to indicate a survey had been performed. There was also evidence that inadequate surveys were performed in that contamination was missed or items were released before being completely deconned and resurveyed. The evaluation identified two root causes: 1) inadequate radiological surveys performed by HPT personnel releasing equipment; and 2) management failure to adequately develop, implement, and provide oversight for the SGT Radioactive Material Demobilization Plan.

Analysis: The inspectors noted that the failure to adequately survey contaminated equipment prior to release from the radiologically controlled area and subsequent shipment of the material as “clean” to two locations is a performance deficiency. The licensee is expected to make surveys of radioactive materials sufficient to ensure compliance with the requirements of 10 CFR 20, including 10 CFR 20.1802 which requires the licensee to maintain control of licensed material. The finding is greater than minor because it is associated with the human performance attribute of 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 conduct adequate surveys resulted in the free-release of contaminated equipment which was subsequently shipped to two locations, including an unlicensed individual, potentially leading to exceeding the dose limits to members of the public through loss of control of licensed material. The finding was evaluated using the Public Radiation Safety SDP. This issue was related to the radioactive material control program and neither resulted in an exposure to the public in excess of five millirem nor involved greater than five occurrences. For these reasons, the SDP evaluation concluded that the issue is of very low safety significance. This finding was entered into the licensee’s corrective action program as PIP O-04-8873. A contributing cause of the finding is related to the cross-cutting area of Human Performance, in that the inadequate surveys performed by the radiation protection technicians and inadequate oversight by licensee management directly contributed to the finding.

Enforcement: 10 CFR 20.1501(a) requires licensees to make surveys necessary for the licensee to comply with the regulations in 10 CFR 20 and 10 CFR 20.1802 requires licensees to control and maintain surveillance of licensed material that is in a controlled or unrestricted area. Contrary to the above, in November and December 2004, numerous pieces of contaminated equipment were surveyed and free-released from the radiologically controlled area and shipped to two locations as “clean” material without appropriate radiological controls. In addition, one of the locations was a non-licensed individual possessing neither the training nor equipment necessary to identify and control the contaminated material. Because the failure to comply with 10 CFR 20.1501(a) and 10 CFR 20.1802 is of very low safety significance, has been entered into the licensee’s corrective action program (PIP O-04-8873), and the associated doses to members of the public were negligible, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000269,270,287/ 2005005-04, Failure to Adequately Survey and Maintain Control of Licensed Material.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee data submitted to the NRC for the performance indicators (PIs) listed below for the period from October 1, 2004, through September 30, 2005. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute 99-02, “Regulatory Assessment Indicator Guideline,” Rev. 3, were used to verify the basis in report for each data element.

Occupational Radiation Safety Cornerstone - For the specified period, the inspectors assessed CAP documents to determine whether HRA, VHRA, or unplanned exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred. For the specified period, the inspectors evaluated data reported to the NRC, and subsequently sampled and assessed applicable CAP documents and selected RP Program records. The reviewed records included personnel exposure investigation reports. Reviewed documents relative to this PI are listed in the report Attachment.

Public Radiation Safety Cornerstone - The inspectors reviewed and evaluated selected radiological liquid and gaseous effluent release data, abnormal release results, cumulative and projected doses to the public, and selected PIP records. Documents reviewed are listed in 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 March 2005 through September 2005, although some examples expanded beyond those dates when the scope of the trend warranted. The review also included issues documented outside the normal CAP in major equipment problem lists, plant health team vulnerability lists, focus area reports, system health reports, self-assessment reports, maintenance rule reports, and Safety Review Group monthly reports. The inspectors compared and contrasted their results with the results contained in the licensee's latest quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensees trend report were reviewed for adequacy.

b. Assessment and Observations

No findings of significance were identified. However, the inspectors noted that a number of PIPs have been generated for mispositioning of valves, breakers and other plant equipment. Specifically, 14 PIPs were written during the last six-month period (July 2005 - December 2005) and another 15 were written during the six-month period prior (January 2005 - June 2005) related to mispositioning of plant equipment. Configuration management coming out of outages accounted for two significant mispositionings that affected Unit 3 SSF pressurizer heaters (URI 05000287/2005004-03) and Unit 2 EFW. The inspectors reviewed the licensee's mispositioning index that is tabulated by the operations department and found that these events were captured in the index and reflected by a reduction of the index from 98.33 to 89.33 from October 2005 to November 2005. The observations were discussed with the Safety Review Group (SRG) Manager to determine if any additional trending were performed in this area. The SRG has been trending mispositioning event data and their data suggested a significant increase in events during the last Unit 2 outage, which was in agreement with the inspectors observations. As a result of their trending, PIP O-05-8439 was written to perform a common cause assessment of recent mispositioning events (PIPs O-05-7700, O-05-7849, O-05-7871, O-05-7951, O-05-8043) in an attempt to capture any programmatic weaknesses and to develop appropriate corrective actions. The inspectors found during their review that mispositioning events have been an area of increased plant focus. The licensee has formed a mispositioning review team that is headed by operations and includes members from a variety of plant organizations that periodically meet to review mispositioning event data to ensure it is appropriately categorized and trended. In addition, mispositioning events have increased management attention as the mispositioning index has recently been added to the daily Management Focus Meeting as a regularly scheduled update.

.3 Focused Review

a. Inspection Scope

The inspectors performed an in-depth review of two issues entered into the licensee's corrective action program. The samples were within the mitigating systems cornerstone and involved risk significant systems. 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 issues and corrective actions were reviewed:

- Containment Coatings Remediation Effort
- High Energy Line Break Issue Resolution (still under review)
- Tornado Mitigation Resolution (still under review)

b. Findings

No findings of significance were identified.

.4 Annual Sample Review

a. Inspection Scope

The inspectors selected PIP O-04-00518 for review. This PIP involved the licensee's evaluation for Violation (VIO) 05000269,270,287/2004007-01: Failure to Obtain Prior NRC Approval to a Change to the Facility Involving Unreviewed Safety Questions on

High Energy Line Break Analysis. This PIP was previously reviewed (see NRC Reports 05000269,270,287/2004005 and 2005002) to verify that the licensee had performed an adequate root cause evaluation and initiated appropriate interim and long term corrective actions. Short term actions have included additional rigorous review of Screenings and Evaluations, increased oversight and assessments of the process, and lessons learned training. The inspectors noted previously that long term corrective actions were incomplete and the licensee had not conducted an extent of condition review required by the licensee's process. The inspectors evaluated status of corrective actions and the extent of condition, reviewed related PIPs, reviewed results of ongoing assessments, and held discussions with licensee personnel regarding these issues.

b. Findings

No findings of significance were identified.

To date, the licensee has completed planned program revisions and conducted additional separate training for personnel performing Screens and Evaluations. The number of personnel qualified for Evaluations has been significantly reduced in order to maintain improved proficiency. The licensee plans to provide additional basic training regarding the licensing basis by April 2006; and additional computerization of the process with a tutorial component by May 2006. The inspector concluded that the additional rigor in the process should be sufficient to allow delay of these additional actions.

The licensee had initiated a plan for extent of condition review in June 2005 calling for a review of Oconee evaluations conducted during 1999 and 2000. The licensee assessment team had discovered that this was the period of time when the Oconee reviews had the least rigor. Site personnel were requested to begin the review in July 2005. The licensee subsequently delayed starting the review until December 30, 2005. Since the assessment had not yet begun, further delay will be required. Although Extent of condition timeliness is not a regulatory requirement, the extent of condition completion has been untimely. Further inspection is warranted to confirm completion of licensee corrective actions.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000287/2005-01-00, TS LCO Condition Allowed Outage Time and Required Action Completion Time Exceeded Due to Blockage of LPI/RBS Pump Room Air Flow Path

The inspectors' inspection activities and evaluation of this event are discussed in detail in Section 4OA5.3 of this report. This LER is closed.

.2 (Closed) LER 05000269/2005-01-01, Exceeded TS: Emergency Power Path Auxiliary Power Source Inoperable

This LER revision resulted from the licensee's discovery of an additional single failure vulnerability related to the DC control power supply configuration for the Keowee Hydro Units (KHUs), while implementing corrective actions for the original issue discussed and closed in Inspection Report 05000269,270,287/2005004. The inspectors reviewed the licensee's corrective actions and extent of condition with respect to single failure vulnerabilities with the KHUs and determined that their actions were appropriate. The risk significance was minimal as there was no actual loss of safety function of the KHUs. This LER revision is closed.

4OA5 Other Activities

.1 (Closed) URI 05000287/2005004-03, Failure to Maintain Design Control of the SSF Supply Power Breaker for Unit 3, Bank 2, Group C Pressurizer Heaters

This issue was discussed in detail in Inspection Report 05000269,270,287/2005004, and was left unresolved pending a Phase 3 risk evaluation. Subsequently, a regional Senior Reactor Analyst performed a Phase 3 risk evaluation of the issue utilizing a Phase 3 evaluation that was performed for a similar finding discussed in Inspection Report 05000269,270,287/2003011. The results of the previous analysis were modified to reflect the exposure time of the current finding. It was determined to be of very low risk significance (Green), primarily due to the short exposure time.

The issue was identified in Inspection Report 05000269,270,287/2005004 as a violation of 10 CFR 50, Appendix B, Criterion III. Specifically, adequate design control of the Unit 3, Bank 2 Group C pressurizer heater breaker had not been maintained, in that, the licensee failed to update the startup procedure with regard to the newly installed breaker. Consequently, the supply breaker had been mispositioned for approximately 234 hours prior to being closed by the licensee on January 4, 2005. Because of the very low safety significance of this issue, because it has been entered into the licensee's corrective action program as PIP O-05-0122, and because it was identified by the licensee, this violation is being treated as a licensee identified NCV, documented in Section 4OA7 of this report.

.2 (Closed) URI 05000269,270,287/2005004-11, Failure to Identify Unmitigated/ Unprotected Feedwater Line Terminal Ends

This issue was discussed in detail in Inspection Report 05000269,270,287/2005004, and was left unresolved pending a Phase 3 risk evaluation. Subsequently, a regional Senior Reactor Analyst performed a Phase 3 risk evaluation of the issue and determined it to be of very low risk significance (Green). This result is different from the Phase 2 because of the method for determining the initiating event frequency, and due to the elimination of non-minimal cutsets. The initiating event frequency for the pipe break was determined in the Phase 3 using methods that counted the number of welds, and feet of pipe, for the area under consideration, and multiplied these values times appropriate failure rates. As a result of the limited number of welds and feet of pipe, a very low initiating event frequency was calculated. The large early release frequency (LERF) impact was below the threshold, because of the size of break required to damage the containment penetration was an even lower probability event.

The issue was identified in Inspection Report 05000269,270,287/2005004 as a NRC identified violation of 10 CFR 50, Appendix B, Criterion XVI, as the licensee failed to identify that unprotected feedwater line terminal ends existed that could impact the mitigation systems needed to protect the plant from a high energy line break (HELB). Because of the very low safety significance of this issue and because the issue has been entered into the licensee's corrective action program as PIP O-06-00138, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000287/2005005-05, Failure to Identify Unmitigated/ Unprotected Feedwater Line Terminal Ends.

.3 (Closed) URI 05000287/2005004-06, Inadequate Design Control of LPI/RBS Room Ventilation Pathways

This issue was discussed in detail in Inspection Report 5000269,270,287/2005004, and was left unresolved pending a Phase 3 risk evaluation. Subsequently, after re-evaluating those sequences with containment sump temperatures greater than 200 degrees F, it was concluded that only two Phase 2 loss of coolant accident (LOCA) sequences (medium break LOCA [MBLOCA] and large break LOCA [LBLOCA]) were applicable. These sequences resulted in two E-7 events which required further review by a regional Senior Reactor Analyst, who determined that the additional contribution to risk caused from an external event initiated MBLOCA and/or LBLOCA was not enough to raise the significance of the event to greater than very low (Green). In addition, since the sequences do not result in containment bypass, the risk from LERF will not cause the event to be greater than Green.

The issue was identified in Inspection Report 05000269,270,287/2005004 as a violation of 10 CFR 50, Appendix B, Criterion III, as the licensee failed to obtain approval from the licensee's design organization prior to modifying the airflow pathway to the Unit 3 B Train LPI/RBS pump room. Because of the very low safety significance of this issue and because the issue has been entered into the licensee's corrective action program as PIP O-05-5564, and because this issue was identified by the licensee, this violation is being treated as a licensee identified NCV, documented in Section 4OA7 of this report.

.4 Operation of an Independent Spent Fuel Storage Installation (ISFSI)

a. Inspection Scope

Under the guidance of Inspection Procedure 60855.1, the inspectors reviewed the licensee's procedure for loading spent fuel shipments to the ISFSI (MP/0/A/1500/016). The inspectors reviewed Oconee Nuclear Engineering Instruction (ONEI-400) for Dry Storage Certification (DSC) for ISFSI shipments DSC-083 and DSC-084 and discussed spent fuel documentation with the cognizant reactor engineer to verify that the licensee has identified each fuel assembly, recorded the parameters and characteristics of each fuel assembly, and has maintained a record of each fuel assembly as a controlled document.

The inspectors selected and reviewed completed procedures for physical inspection and inventory of the ISFSI (PT/0/A/0750/003, Physical Inventory of Reportable Special Nuclear Material, Enclosure 13.6, Dry Cask Storage Inspections) and completed ONEI-400 ISFSI inventory sheets to verify that records have been established for all spent fuel in storage in the ISFSI, that duplicate records are maintained by the licensee, and that a physical inventory has been conducted on all spent fuel stored in the ISFSI at least every 12 months.

b. Findings

No findings of significance were identified.

.5 (Closed) Temporary Instruction (TI) 2515/160, Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water Reactors (Unit 2 - NRC Bulletin 2004-01)

a. Inspection Scope

The inspectors reviewed procedures and records documenting activities relative to inspection of the Unit 2 pressurizer penetrations to verify that the licensee complied with commitments made in July 27, 2004, response to NRC Bulletin 2004-01. The inspectors also independently performed a bare metal visual examination of the following pressurizer penetrations: three six-inch diameter safety nozzles, vent line, manway diaphragm, and four-inch diameter spray nozzle on pressurizer head; and three instrument (level transmitter) nozzles penetrations in the pressurizer vessel wall. The guidelines for the inspection were provided in TI2515/160.

b. Findings

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

TI 2515/160 Reporting Requirements:

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

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

2. Performed in accordance with demonstrated procedures? The visual inspections were conducted in accordance with Duke Power Procedure QAL - 15, Inservice Inspection (ISI) Visual Examination, VT-2, Pressure test, Revision 22. The inspectors reviewed the inspection procedure and

verified that it had been reviewed and approved in accordance with the licensee's procedure review process and NRC requirements. The inspectors verified that the procedure specified inspection prerequisites, inspection requirements, included minimum lighting requirements, adequate instructions for performing the visual examination of the pressurizer penetrations, and inspection documentation requirements.

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

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

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

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

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

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

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

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

(h) If volumetric or surface examination techniques were used for the augmented inspections examinations, what process did the licensee use to evaluate and dispose any indications that may have been detected as a result of the examinations? No indications were identified. Licensee is planning to replace instrument nozzle penetrations in a future outage to facilitate performance of UT exams on these nozzles.

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

.6 ISFSI Radiological Controls

a. Inspection Scope

The inspectors conducted independent gamma and neutron surveys of the ISFSI facility and compared the results to previous surveys. The inspectors also observed and evaluated implementation of radiological controls, including RWPs and postings, and discussed the controls with an HPT and Health Physics supervisory staff. Radiological controls for loading the ISFSI casks were also reviewed and discussed.

Radiological control activities for ISFSI areas were evaluated against 10 CFR Part 20, 10 CFR Part 72, and applicable licensee procedures. Documents reviewed are listed in section 4OA5 of the report Attachment.

b. Findings

No findings of significance were identified.

.7 (Closed)TI 2515/161, Transportation of Reactor Control Rod Drives (CRDs) in Type A Packages

a. Inspection Scope

The inspectors reviewed shipping logs and discussed shipment of CRDs in Type A packages with shipping staff. The inspectors reviewed shipments made since January 1, 2002, and noted that no shipments of CRDs in Type A packages were made during this time period.

b. Findings

No findings of significance were identified.

.8 (Closed) URI 05000269,270,287/2004002-04, Potential Failure to Maintain Reactor Building Coatings per GL 98-04 Commitments Resulting in Potential Loss of Reactor Building Emergency Sump (RBES) Recirculation

This issue, which was discussed in detail in Inspection Reports 05000259,270,287/2004002 and 05000259,270,287/2004003, was left unresolved pending determination of the impact the degraded coatings may have on the RBES in each Unit, and to determine if the licensee took adequate action to ensure the amount of coatings susceptible to detachment during a LOCA was minimized.

By letter dated August 7, 2003, the licensee provided a response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," dated June 9, 2003, for Oconee Nuclear Station, Units 1, 2, and 3 (ONS). The NRC staff reviewed and closed the response in a closure letter dated March 30, 2004. Subsequently, concerns regarding degraded containment coatings, remediation efforts, and adequacy of Bulletin 2003-01 interim compensatory measures were raised by the NRC inspection staff. Therefore, the review of Bulletin 2003-01 was reopened and additional information was requested by letter dated March 30, 2005. The licensee responded with additional information in letters dated April 29, 2005, August 16, 2005, and October 13, 2005. NRC staff reviewed the additional information, and documented its evaluation in a letter to the licensee dated January 20, 2006.

In PIP O-04-02591, the licensee documented its paint chip transportability assessment used to conclude that mechanisms necessary to create sump screen blockage within the context of Generic Safety Issue (GSI) -191, “Assessment of Debris Accumulation on PWR Sump Performance,” do not appear to exist. In the above letter, the NRC staff found that the licensee reasonably addressed transport of degraded coatings to the sump. A full review of coating debris transport will be necessary for resolution of GSI-191. However, the current licensing basis for ONS assumes 50 percent sump screen blockage and does not require a full mechanistic evaluation. As described in GL 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” licensees should comply with established regulatory requirements until all plant modifications are completed in accordance with the resolution schedule (i.e., December 31, 2007).

The NRC staff’s evaluation also concluded that the interim compensatory measures put in place, reduce the interim risk associated with potentially degraded or nonconforming emergency core cooling system recirculation functions. However, as part of the overall GSI-191 effort, the staff expects Oconee to address coatings and their impact on sump performance. In a public meeting held on December 20, 2005, the licensee detailed their long range plans for containment coating remediation and compliance with Generic Letter 2004-02.

In addition, the NRC’s January 20, 2006, letter also identified key areas that require further evaluation in order to successfully resolve GSI-191 at ONS. These areas will be reviewed as part of the staff’s audit of the ONS response to Generic Letter 2004-02. Accordingly, based on the above, this URI is closed.

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 January 5, 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.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being disposition as a NCVs.

- The inspectors determined that a licensee identified violation of 10 CFR 50, Appendix B, Criterion III, Design Control occurred, as the licensee failed to maintain adequate design control of the Unit 3, Bank 2 Group C pressurizer heater breaker. Specifically, the licensee failed to update the startup procedure with regard to the newly installed breaker. The risk significance and enforcement aspects of this issue were discussed in detail in Inspection Report 05000269,270,287/2005004, and Section 4OA5.1 of this report.

- The inspectors determined that a licensee identified violation of 10 CFR 50, Appendix B, Criterion III, Design Control occurred as the licensee failed to obtain approval from the licensee's design organization prior to modifying the airflow pathway to the Unit 3 B Train LPI/RBS pump room. The risk significance and enforcement aspects of this issue were discussed in detail in Inspection Report 05000269,270,287/2005004, and Section 4OA5.3 of this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

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
S. Capps, Mechanical/Civil Engineering Manager
C. Cauthen, SG Component Engineer
T. Coleman, ISI Coordinator
N. Constance, Operations Training Manager
D. Covar, Training Instructor
C. Curry, Maintenance Manager
G. Davenport, Compliance Manager
K. Davis, ECT Level III
P. Downing, Manager Steam Generator Maintenance & Engineering
C. Eflin, Requalification Supervisor
P. Fowler, Access Services Manager, Duke Power
T. Gillespie, Reactor and Electrical Systems Manager
J. Gilreath, SG Component Engineer
M. Glover, Engineering Manager
T. Grant, Engineering Supervisor, Reactor & Electrical Systems
R. Griffith, QA Manager
B. Hamilton, Station Manager
L. Hekking, RCS Leakage Engineer
R. Hester, Civil Engineer
D. Hubbard, Training Manager
R. Jones, Site Vice President
G. Kent, Nuclear Regulatory Issues and Industry Affairs Engineer
T. King, Security Manager
T. Ledford, Engineering Supervisor, Reactor & Electrical Systems
L. Llibre, Engineering Supervisor
R. Matheson, Safety Review Group Manager
D. Mayes, Consulting Engineer
R. Murphy, Engineering Support
S. Neuman, Regulatory Compliance Group
L. Nicholson, Safety Assurance Manager
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, Manager, Radiation Protection
J. Weast, Regulatory Compliance

NRC

M. Ernstes, Chief of Reactor Projects Branch 1
R. Haag, Chief of Plant Support Branch 1
L. Olshan, Project Manager, NRR
L. Plisco, Deputy Regional Administrator, RII

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000270/2005005-02	URI	Inadequate Foreign Material Exclusion Controls for the A and B Train Reactor Building Emergency Sump Suction Lines (Section 1R15)

Opened and Closed

05000269,270,287/2005005-01	NCV	Inadequate Procedures for Testing the SSF Diesel Generator With the CCW Supply Secured (Section 1R12)
05000269,270,287/2005005-03	NCV	Failure to Ensure Adequacy of Measurements of Particulate Effluents Released from Unit Vent (Section 2PS1)
05000269,270,287/2005005-04	NCV	Failure to Adequately Survey and Maintain Control of Licensed Material (Section 2PS3)
05000269,270,287/2005005-05	NCV	Failure to Identify Unmitigated/Unprotected Feedwater Line Terminal Ends (Section 4OA5.2)

Closed

05000287/2005-01-00	LER	TS LCO Condition Allowed Outage Time and Required Action Completion Time Exceeded Due to Blockage of LPI/RBS Pump Room Air Flow Path (Section 4OA3.1)
05000269/2005-01-01	LER	Exceed TS: Emergency Power Path Auxiliary Power Source Inoperable (Section 4OA3.2)

05000287/2005004-03	URI	Failure to Maintain Design Control of the SSF Supply Power Breaker for Unit 3, Bank 2, Group C Pressurizer Heaters (Section 4OA5.1)
05000269,270,287/2005004-11	URI	Failure to Identify Unmitigated/Unprotected Feedwater Line Terminal Ends (Section 4OA5.2)
05000287/2005004-06	URI	Inadequate Design Control of LPI/RBS Room Ventilation Pathways (Section 4OA5.3)
2515/160	TI	Pressurizer Penetration Nozzles and Steam Space Piping Connections in U. S. Pressurized Water Reactors - Unit 2 (Section 4OA5.5)
2515/161	TI	Transportation of Control Rod Drives in Type A Packages (Section 4OA5.7)
05000269,270,287/2004002-04	URI	Potential Failure to Maintain Reactor Building Coatings per GL 98-04 Commitments Resulting in Potential Loss of RBES Recirculation (Section 4OA5.8)

Discussed

05000269,270,287/2004007-01	VIO	Failure to Obtain Prior NRC Approval to a Change to the Facility Involving Unreviewed Safety Questions on High Energy Line Break Analysis (Section 4OA2.4)

DOCUMENTS REVIEWED

1R08: Inservice Inspection Activities

[ISI]

QAL-13, Inservice Inspection (ISI) Visual Examination, VT-1, VT-1C and VT-1MC, Rev. 20
QAL-14, Inservice Inspection (ISI) Visual Examination, VT-3, Rev. 26
QAL-15, Inservice Inspection (ISI) Visual Examination, VT-2, Rev. 22
NDE-35, Liquid Penetrant Examination, Rev. 20
NDE-600, Ultrasonic Examination of similar Metal Welds in Ferrtitic and Austentic Piping, Rev. 0, Field Change FC05-08
PDI-UT-5, PDI Generic Procedure for Straight Beam Ultrasonic Examination of Bolts and Studs, Rev. C

MP/0/A/1800/132, Evaluation of Boric Acid Leakage on Mechanical, Structural and Electrical Components, Rev. 0
OP/0/A/1102/028, Reactor Building Tour, Rev. 21

[SG]

SGMEP 105 - "ROTSG Specific Assessment of Potential Degradation Mechanisms"
Rev. 5, 9/21/05.
Document 51-9001693-000, X-Probe High Speed Tests Report, 9/30/05.
Document 51-5064539-000, "A Condition Monitoring and Operational Assessment Evaluation of Wear Scars for Oconee Unit 1, EOC22, 5/25/05.
Eddy Current Analysis Guidelines for Duke Power Company's ROTSG, Revision 1.
Duke Response to Generic Letter 2004-01, Requirements for Steam Generator Tube Inspections, October 28, 2004.
Field Procedure and Operating Instructions for Installation of a Flexible Stabilizer in a Recirculating Steam Generator, Rev. 17.
Document 1274768A, "Secondary Side Visual Inspection and Loose Parts Retrieval Procedure for Heat Exchangers" 1/22/04.

PIPs

[ISI]

O-05-06919	Inadequate Evaluation of Indications Identified in 1996 on Pressurizer Weld 2-PZR-WP76
O-05-06748	Boric Acid Leaks Identified During Unit 2 Walkdown
O-05-07125	Coupling on steam generator blowdown header found to be below min. wall

[SG]

O-05-02678,Unexpected tube wear was identified in the Unit 1 "A" & "B" Steam Generators during the 1EOC22 outage.
O-05-03607, PORC Meeting Minutes (5/10/05) - PORC review of U1 steam generator tube wear - restart authorization & operability of Units 2&3.
O-05-03386, Stress analysis performed for replacement OTSGs may not meet the design specification OSS-0279.00-00-0002.
O-05-06246, An internal assessment for the Steam Generator Program has discovered that no 4th interval inservice inspection plans have been developed nor submitted to the regulatory authorities as required by ASME Section XI.
O-02-01153, Steam Generator tubes plugged pre-service with welded plugs may have been intentionally violated (punctured) prior to plugging and may not meet current tube stabilization requirements.

2OS1: Access Control To Radiologically Significant Areas

Procedures, Instructions, Guidance Documents, and Operating Manuals

Duke Industrial Radiography Safety Manual, Rev. 0
Duke Power Company (DPC), Oconee Nuclear Station (ONS), Independent Spent Fuel Storage Installation Phase III and Phase IV DSC Loading and Storage, Procedure Number (No.) MP/0/A/1500/016, Revision (Rev.) 023
DPC, ONS, Physical Inventory of Reportable Special Nuclear Materials, Procedure No. PT/0/A/0750/003, Rev. 016
DPC, ONS, Radiation Protection Routines, Procedure No. HP/0/B/1000/054, Rev. 38
DPC, ONS, Radiological Protection Requirements for Independent Spent Fuel Storage Installation Phases III, IV, and V (DSCs 41-99), Procedure No. HP/0/B/1000/097
DPC, Standard Procedure for Oconee, McGuire and Catawba Nuclear Stations (SPOMCNS), Access Controls for High, Extra High, and Very High Radiation Areas, Procedure No. SH/0/B/2000/012, Rev. 004
DPC, SPOMCNS, Internal Dose Assessment, Procedure No. SH/0/B/2001/001, Rev. 002
DPC, SPOMCNS, Investigation of Skin and Clothing Contaminations, Procedure No. SH/0/B/2001/003, Rev. 006
DPC, SPOMCNS, Investigation of Unusual Dosimetry Occurrence or Possible Overexposure, Procedure No. SH/0/B/2001/002, Rev. 004
DPC, SPOMCNS, Posting of Radiation Control Zones, Procedure No. SH/0/B/2000/005, Rev. 002
DPC, SPOMCNS, Preparation of a Radiation Work Permit (RWP), Procedure No. SH/0/B/2000/003, Rev. 006
DPC, SPOMCNS, , Removal of Items from RCA/RCZs and Use of Release/Radioactive Material Tags, Procedure No. SH/0/B/2000/006, Rev. 001
DPC, SPOMCNS, Taking, Counting and Recording Surveys, Procedure No. SH/0/B/2000/004, Rev. 006
Duke Power Quality Assurance Program Related (DPQAPR), Nuclear Policy Manual (NPM), Nuclear System Directive (NSD): 208, Problem Investigation Process (PIP), Rev. 27
DPQAPR, NPM, NSD: 210, Corrective Action Program, Rev. 4
DPQAPR, NPM, NSD: 223, Trending Program, Rev. 4

Records and Data Reviewed

Extra High Radiation Area Key Logbook
Non-Fuel Material Stored in the Spent Fuel Pools Inventory List, Dated 08/02/05
ONS, One Liner Survey Report, Survey Nos. 102605-37, RW 240 Opening at Top of Tank; and 110805-27, Alpha Letdown Cooler
ONS, Survey Nos. 022705-3, 021505-10, 030105-15, 030605-1, 030705-9, 032505-2, 043005-3, 080805-13, 100705-16 and 101805-21 for RM254.WMF - Room 254 SFP Demineralizer and Filter Room; 040405-10 and 052605-5 for RW240.WMF - Rad Waste Room 240 Resin Batch Tank; and 060100-10, 061105-12, 081305-11, 100999-3 and 102705-38 for RW236.WMF - Rad Waste Room 236 Corridor
RWP No. 24, Removal and Replacement of Radioactive Filters/Strainers (Including Vacuums/HEPAs)

RWP No. 34, Radiography Operations Inside of the Nuclear Station Owner Control Fence
RWP No. 99, Radiography Operations Outside of the Nuclear Station Owner Control Fence
RWP No. 2216, “A” OTSG Remove/Replace Primary Manway/Handhole Covers
RWP No. 2217, “A” OTSG Mechanical Decon
RWP No. 2218, “A” OTSG Install/Remove Nozzle Dams
RWP No. 2239, “B” OTSG Remove/Replace Primary Manway/Handhole Covers
RWP No. 2241, “B” OTSG Install/Remove Nozzle Dams
RWP No. 5006, Spent Resin Operations
RWP No. 5008, Removal and Replacement of Letdown Filters
RWP No. 5009, Removal and Replacement of Spent Fuel Filters
RWP No. 5014, Hot Spot Reduction/Removal Activities
RWP No. 5029, Surveillance/System Operation and Corrective Maintenance in Extra High Radiation Area Fields

<u>CAP Documents</u>

PIP O-04-00545, PCE 04-003, Distributed skin - Protective clothing did not protect worker assigned to Rx head team working as fuel handler
PIP O-04-00565, Sudden expulsion of water out the vent valve of the ISFSI canister, Load 79, while welders were grinding on the weld
PIP O-04-02671, ONS RT communication posters for radiography contain errors
PIP O-05-0624308439, RT boundary violation during RCS HL RT
PIP O-05-03146, The RMA boundary rope on the ground floor of the Turbine building was prematurely moved prior to material inside the RMA being surveyed for release
PIP O-04-08431, Unauthorized individual inside posted radiography boundary
PIP O-05-03615, RP key box in RMC office was found unlocked
PIP O-05-06243, Storage & transporting of radioactive scaffold boards outside of the RCA
PIP O-05–07553, Gate/entranceway at the ISFSI was open and a clear view of the radiological posting was compromised
PIP O-05–07557, A roped off radioactive material storage area containing radioactive material was discovered without the Radioactive Material insert in the posting

2OS2: ALARA Planning and Controls

<u>Procedures, Instructions, Guidance Documents, and Operating Manuals</u>

System ALARA Manual: Section III, ALARA Program, Rev. 14; Section IV, ALARA Planning, Rev. 16; Section VIII, Station ALARA Committee, Rev. 15; Attachment 6.2 to Section IV, ALARA Planning Worksheet (APW)
DPC, SPOMCNS, Declared Pregnant Worker, Procedure No. SH/O/B/2002/003, Rev. 1

<u>Records and Data</u>

2A LPI Cooler: Comparison of Crud Burst Peak Data - 2EOC21 vs 1EOC22 [graph]
ALARA Items in Modification Arena (updated 09/14/2005)

APW, ALARA Briefing Checklist, and Post-Job ALARA Critique for "A" and "B" SG Upper and Lower Channel Head Wash and Perform RP Channel Head Surveys (with RWPs 2217 and 2240)
APW, ALARA Briefing Checklist, and Post-Job ALARA Critique for Reactor Vessel Annulus
APW, ALARA Briefing Checklist, and Post-Job ALARA Critique for Unit 2 "A" and "B" SG - Install and Remove Nozzle Dams (with RWPs 2218 and 2241)
APW for "A" and "B" SG - Eddy Current Test Inspections and Associated Inspections (with RWPs 2219 and 2242)
APW for Reactor Building Emergency Sump Strainer Modification OD20049, Rev. 1 (with RWP 2053)
APW for Remove,/Replace 2B2 RCP Motor and associated hangers, snubbers, misc. steel (with RWP 2193)
APW for Unit 2 "A" Letdown Heat Exchanger - Investigate Tube Leak, Repair in Place (with RWP 2025)
APW for Unit 2 SG - Tube Pull for Inspection (with RWP 2229)
APWs for Load, Weld Transport and Store ISFSI Load Nos. 83 and 84 Long Cavity Cask (with RWPs 5099 and 5099)
Dose Rate Monitoring - Average Hourly Dose Rates at Unit Auxiliary Building Room 121 (2A LPI Cooler), 10/22-28/2005 [graph]
Entries (Inspections, Decon, Insulation, Paint Abatement) (with RWP 2026)
Oconee Unit 2, EOC-21 Crud Burst Plan, 10/22/2005
ONS 2005 TLD Dose Estimates (by work groups)
Records of ALARA Committee Meetings for 4th Quarter 2004, 1st Quarter 2005, 2nd Quarter 2005 Active Hot Spot Data Base
Revisions to U2EOC21 Exposure Estimate
Room 63 General Area Dose Rates During Canal Fill, 10/27-28/2005 [graph]
U2EOC21 - Crud Burst Date RCP Discharge [graph]
Unit 1, End of Cycle 22 RP/ALARA Outage Report [undated]

CAP Documents

PIP O-03-07619, 1EOC21 Personnel Radiation Exposure is exceeding the estimate
PIP O-04-01522, Two workers exceeded dose alarm set point
PIP O-04-01840, Dose estimate exceeded for plenum move
PIP O-04-02724, Dose for Unit 2 normal sump decon exceeded the estimate by more than 25%
PIP O-05-00561, Estimated dose for DSC 83 was exceeded by 1150 mrem
PIP O-05-03135, Some ALARA program administrative requirements are not being met
PIP O-05-04859, 2nd Quarter 2005 Chemistry ALARA Assessment
PIP O-05-06736, Several team members requested support from RP for initial Unit 2 containment entry did not attend the initial entry pre-job brief
PIP O-05-06854, Mirror insulation that was removed to inspect bottom of reactor vessel needs its support rods modified because of ALARA concerns
PIP O-05-07306, Disassembly of Control Rod Drive #41 will exceed estimated exposure by more than 25%

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

Procedures, Guidance Documents and Manuals

DPC, ONS, RB Purge System, Procedure No. OP/2/A/1102/014, Rev. 20
DPC, ONS, Reactor Containment Building Sampling and Release Rate Determination for Gaseous Purge, Procedure No. HP/0/B/1000/060B, Rev. 54
DPC Nuclear Guide 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, Rev. 1B
Offsite Dose Calculation Manual 2004, Rev. 44

Records and Data

DPC Engineering Change No. OD100437, Reconfigure the sample tubing for 1RIA-43, 10/20/05
Gaseous Waste Release (GWR) 2005-050, Gaseous Waste Decay Tank 3A, 9/25/05
GWR 2005-009, Gaseous Waste Decay Tank 3C, 5/9/05
GWR 2005-016, Radwaste Facility Vent, 3/1/05
GWR 2005-028, Miscellaneous Elevated (Release to U1 RB during primary system depressurization), 4/13/05
GWR 2005-058, U2 Reactor Building Purge, 10/25/05
White Paper: Anisokinetic Sampling Error - Affect on Unit Vent Radiation Monitors and Effluent Release Determination

CAP Documents

PIP O-98-0710, Inlet tubing for 2-RIA-43 through 46 does not appear to meet design requirements of ANSI N13.1-1969, 2/12/98
PIP O-03-1868, “Tee” connection on RIA-43 may not meet isokinetic requirements of ANSI N13.1- 1969, 4/2/03
PIP O-04-7084, Tees/elbows on RIA-43 skid do not conform to ANSI N13.1-1969 Appendix B, 10/21/04
PIP O-05-4874, An operability assessment should have been performed on the unit vent particulate sampling program, since this sampling is credited for meeting SLC requirements, 7/28/05

2PS2: Radioactive Material Processing and Transportation

Procedures, Guidance Documents and Manuals

10 CFR Part 61 Waste Classification and Waste Form Implementation Program Manual, 09/26/05
DPC, ONS, Radwaste Facility Liquid Waste Processing, Procedure No. CP/0/B/5200/054, Rev. 46
DPC, ONS, Cask - CNS 14-215H - Handling Procedure, Procedure No. MP/0/A/1701/015, Rev. 20

DPC, ONS, Cask - CNS 8-120 B - Handling Procedure, Procedure No. MP/0/A/1701/016, Rev. 20
DPC, ONS, Liquid Waste Release from RWF, Procedure No. CP/0/B/5200/045, Rev. 70
DPC, ONS, Radwaste Dewatering and Operating Guidelines, Procedure No. CP/0/B/5400/001, Rev. 25
DPC, ONS, Primary Demineralizer Sluice, Procedure No. CP/0/B/5200/071, Rev. 13
DPC, SPOMCNS, Preparation and Shipment of Radioactive Material, Procedure No. SH/0/B/2004/001, Rev. 004
DPC, SPOMCNS, Preparation and Shipment of Radioactive Waste, Procedure No. SH/0/B/2004/002, Rev. 004
DPQAPR, NPM, NSD 208, Problem Investigation Process (PIP), Rev. 27
Duratek Engineering Report ER-99-011, Conformance of CNS 14-215H Cask with Specifications for DOT 7A, Type A Packagings, Rev. 2
Duratek Procedure TR-OP-032, Handling Procedure for Transport Cask Number CNS 14-215H, Rev. 13
Standard Radiation Protection Management Procedures (SRPMP) 10-3, Annual Radiation Protection Source Term Data Review, Rev. 0

Records and Data

10 CFR 50.59 Evaluation for Procedure CP/0/B/5400/001, Radwaste Dewatering & Operating Guidelines - Proprietary, Revs. 24 and 25
2003 and 2004 Annual Radioactive Effluent Release Reports
2004 Source Term Review, 01/08/04 (PIP O-04-0120)
2005 Source Term Review, 03/22/05 (PIP O-05-1936)
2005 Radwaste Report
ONS Waste Streams Data Sample Report, 07/05/05
Radioactive Shipment Log, Calendar Years 2004 and 2005
Radioactive Shipping Record Nos. ONS 04-2036, 05/20/04; ONS 04-2068, 09/21/04; ONS 05-2089, 08/02/05; ONS 05-2094, 10/06/05; ONS 05-2099, 10/25/05; ONS 05-2104, 11/09/05; and ONS 05-2015, 02/18/05
Training Content Summary and Attendance Record: Radwaste Certification, 08/07/03

CAP Documents

2004 Radiation Protection FAE Working Checklist, Radioactive Material Control, 12/05/03
NPA Assessment GO-04-007(NPA)(RP)(ALL), Radiation Protection Functional Area Evaluation, 04/14/04
PIP O-03-7656, Trend in radioactive material shipment events, 11/24/03
PIP O-04-0954, Current method for obtaining confirmation that vendor supplied procedures and documentation for radioactive shipping packages has been incorporated into station procedures in not efficient, 2/25/04
PIP O-04-1270, Oconee Pre-INPO Assessment GO-04-03(NPA)(Pre-INPO CFA)(ONS), 03/10/04
PIP O-04-1984, Shipping cask procedures contain areas that are not perceived to be in agreement with guidance provided by governing Duratek documents, 04/02/04

PIP O-05-2190, Routine review of vendor document revisions per the DPC Process Control Program, 04/04/05
PIP O-05-2842, Duratek provided the following revisions to their 8-120 cask manual, 04/22/05
PIP O-05-6444, NPAS Audit GO-05-018(NPA)(CHM)(ALL), Chemistry Functional Area Evaluation, 10/07/05
PIP O-05-7511, Transport Index found to be incorrect on one of the forms in RSR #ONS 05-2094, 11/09/05

2PS3: Radiological Environmental Monitoring Program and Radioactive Material Control Program

Records and Data

PIP O-4-8873, Oconee Root Cause Analysis, SGRP RAM Release Event (includes event timeline, inter-office communications, condition reports, radiological surveys of equipment, interviews with RPT, supervisors, and managers, etc.)

CAP Documents

PIP O-04-8873, On 12/17/04, Oconee RP was notified that clean material shipped from Oconee to Arkansas Nuclear One to support their outage was surveyed and found to have indications of contamination, 12/18/04
PIP O-04-8972, RP follow-up survey on previously released material was found to be contaminated on site, 12/21/04
PIP O-05-0615, PORC review of interim procedural guidance for the releases of SGT equipment and materials from ONS, 01/24/05
PIP O-05-4965, Follow up to issues concerning Radioactive Material Control deficiencies documented in PIP 04-8873, 08/02/05

4OA1: Performance Indicator Verification

Procedures, Guidance Documents and Manuals

DPQAPR, NPM, NSD: 208, Problem Investigation Process Rev. 27

Records and Data

PIP No. O-05–03139, Documentation of 12 Areas for Improvement (AFI) identified during an NPAS Audit, GO-05-014 (NPA)(RP)(ALL) Radiation Protection FAE

4OA5.7 and 4OA5.8: Other Activities

ISFSI Documents Reviewed

ONS, Survey Nos. 012805-9, 020405-3 and 020405-4 for ISFSI2.WMF - Transfer Cask on Trailer Survey; 020105-2 and 020305-1 for DECONPIT.WMF - I/S Decon Pit @ Grating Level - With Cask; 020105-10 for RM619A.WMF - Room 619 Unit ½ Spent Fuel Pool; 020405-2 for

ISFSI5.WMF - Transfer Cask Survey - Top Shield Plug On; 020405-8 for ISFSI6.WMF - Horizontal Storage Module (New Type); and 020405-10 for ISFSLR5.WMF - Independent Spent Fuel Storage Installation

TI 2515/161 Documents Reviewed

Radioactive Shipment Log: 2002, 2003, 2004, 2005

LIST OF ACRONYMS

ADAMS	-	Agency wide Documents Access and Management System
ALARA	-	As Low As Reasonably Achievable
ANO	-	Arkansas Nuclear One
ANSI	-	American National Standards Institute
AP	-	Abnormal Procedure
ASW	-	Auxiliary Service Water
BS	-	Building Spray
CAP	-	Corrective Action Program
CCPM	-	Corrected Counts per Minute
CCW	-	Condenser Circulating Water
CFR	-	Code of Federal Regulations
CRD	-	Control Rod Drive
CTPD	-	Core Thermal Power Demand
DAW	-	Dry Active Waste
DC	-	Direct Current
DEC	-	Duke Energy Corporation
DG	-	Diesel Generator
DSC	-	Dry Storage Certification
EDG	-	Emergency Diesel Generator
EFW	-	Emergency Feedwater
EHRA	-	Extra High Radiation Area
EOC	-	End-of-Cycle
ET	-	Eddy Current Testing
FDW	-	Feedwater
HDP	-	Heater Drain Pump
HEPA	-	High Efficiency Particulate Air
HPI	-	High Pressure Injection
HPSW	-	High Pressure Service Water
HPT	-	Health Physics Technician
HRA	-	High Radiation Area
IP	-	Inspection Procedure
IR	-	Inspection Report
ISFSI	-	Independent Spent Fuel Storage Installation
ISI	-	Inservice Inspection
IST	-	Inservice Testing
KHU	-	Keowee Hydroelectric Unit

LCO	-	Limiting Condition for Operation
LER	-	Licensee Event Report
LPI	-	Low Pressure Injection
LPT	-	Liquid Penetrant
MDEFW	-	Motor Driven Emergency Feedwater
MS	-	Main Steam
NCV	-	Non-Cited Violation
NDE	-	Non-Destructive Examination
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
NSD	-	Nuclear System Directive
ONS	-	Oconee Nuclear Station
OOS	-	Out of Service
PARS	-	Publicly Available Records
PCP	-	Process Control Program
PI	-	Performance Indicator
PIP	-	Problem Investigation Process report
PM	-	Preventive Maintenance
PMT	-	Post-Maintenance Testing
PT	-	Performance Test
RB	-	Reactor Building
RBCU	-	Reactor Building Cooling Unit
RBES	-	Reactor Building Emergency Sump
RBS	-	Reactor Building Spray
REMP	-	Radiological Environmental Monitoring Program
Rev.	-	Revision
RFO	-	Refueling Outage
RII	-	Region II
RP	-	Radiation Protection
RWP	-	Radiation Work Permit
RTP	-	Rated Thermal Power
SDP	-	Significance Determination Process
SG	-	Steam Generator
SGT	-	Steam Generator Team
SGR	-	Steam Generator Replacement
SLC	-	Selected Licensee Commitments
SSC	-	Structure, System and Component
SSF	-	Standby Shutdown Facility
TDEFW	-	Turbine Driven Emergency Feedwater
TI	-	Temporary Instruction
TLD	-	Thermoluminescent Dosimetry
TS	-	Technical Specification
U2	-	Unit 2
U3	-	Unit 3
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
UT	-	Ultrasonic Examination

VHRA	-	Very High Radiation Areas
VT	-	Visual Examination
WO	-	Work Order