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

April 26, 2004

Duke Energy Corporation
ATTN.: Mr. R. A. Jones
Site Vice President
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION - INTEGRATED INSPECTION REPORT 05000269/2004002, 05000270/2004002, AND 05000287/2004002

Dear Mr. Jones:

On March 27, 2004, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Oconee Nuclear Station. The enclosed report documents the inspection findings which were discussed on April 1, 2004, with Mr. Dave Baxter 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.

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

Robert Haag, Chief
Reactor Projects Branch 1
Division of Reactor Projects

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

Enclosure: NRC Integrated Inspection Report 05000269/2004002, 05000270/2004002, and 05000287/2004002 w/Attachment (Supplemental Information)

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REGION II

Docket Nos:	50-269, 50-270, 50-287
License Nos:	DPR-38, DPR-47, DPR-55
Report No:	50-269/2004002, 50-270/2004002, 50-287/2004002
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	December 28, 2003 - March 27, 2004
Inspectors:	M. Shannon, Senior Resident Inspector A. Hutto, Resident Inspector E. Riggs, Resident Inspector L. Mellen, Senior Emergency Preparedness Inspector (Sections 1EP1, 1EP4, 1EP5, and 4OA1) J. Kreh, Emergency Preparedness Inspector (Sections 1EP1, 1EP5, and 4OA1) W. Loo, Senior Health Physicist (Sections 2OS1, 4OA5.1 and 4OA5.2) H. Gepford, Health Physicist (Sections 2OS1, 4OA5.1 and 4OA5.2) R. Chou, Reactor Inspector (Section 4OA5.3) G. Laska, Operations Engineer (Section 1R11.2)
Approved by:	R. Haag, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000269/2004002, IR 05000270/2004002, IR 05000287/2004002; 12/28/2003 - 03/27/2004; Oconee Nuclear Station, Units 1, 2, and 3; Personnel Performance During Nonroutine Plant Evolutions, and Surveillance Testing.

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

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

Cornerstone: Initiating Events

- <u>Green</u>. A self-revealing finding was identified for inadequate foreign material exclusion (FME) controls during electro-hydraulic control (EHC) system filter replacement which resulted in a Unit 3 turbine/reactor trip.

  The finding was considered to be more than minor because it resulted in a reactor trip which is considered a transient initiator in the SDP. This issue was determined to be of very low safety significance (Green) based on the Phase 1 SDP screening results. While the inadequate EHC foreign material controls caused a reactor trip, it did not affect any mitigation equipment or functions. (Section 1R14.b.(1))

- <u>Green</u>. A self-revealing non-cited violation of Technical Specification 5.4.1, was identified for an inadequate primary sampling procedure which resulted in a waterhammer-generated pressure wave that caused a thermal relief valve to lift. The relief valve stuck open, resulting in a 14 gpm unidentified reactor coolant system (RCS) leak and a Notice of Unusual Event. The licensee's investigation of the event concluded the relief valve stuck open due to foreign material trapped between its seat and disc.

  The finding was considered to be more than minor because it affected the initiating event cornerstone, in that, the inadequate primary sampling system procedure increased the likelihood of a small loss of coolant accident (LOCA) occurring. However, the stuck open thermal relief valve was readily isolable by the excessive RCS leakage procedure, consequently, the finding screened out of the SDP Phase 1 analysis as being of very low safety significance. (Section 1R14.b.(2))

- <u>Green</u>. A non-cited violation of 10 CFR 50, Appendix B, Criterion X, Inspection, was identified by the inspectors for failure to establish an adequate inspection program for certain feedwater system and main steam system piping supports associated with high energy line break scenarios.

The lack of piping support inspections and resulting inability to identify adverse conditions related to the supports could affect the ability of the feedwater and/or the steam line systems to withstand various events such as seismic induced loading, which in turn could result in damage to other mitigation systems. This issue was considered to be more than minor because if left uncorrected it could prevent the detection of piping support defects which would increase the probability of an initiating event (feed line and steam line rupture). A Phase 1 evaluation was conducted using the initiating event screening criteria. Because the inadequate inspection of the supports had not caused an actual increase the likelihood of an initiating event, the issue was screened out as Green. The determination of no actual increase in the likelihood of an initiating event was based on no significant loading events, such as seismic events, having occurred. (Section 1R22)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 entered the report period in an outage to repair a reactor coolant pump (RCP) seal. On January 2, 2004, while in Mode 3, a notification of unusual event (NOUE) was declared due to unidentified reactor coolant system (RCS) leakage in excess of 10 gpm. The 14 gpm leak was subsequently identified as a stuck open thermal relief valve on the pressurizer water space sample line. Following the repairs, the unit was taken critical on January 3, 2004, and although it achieved 28 percent rated thermal power (RTP), the unit was not placed on-line due problems encountered with the electro-hydraulic control (EHC) system. On January 8, 2004, the unit performed a TS required shutdown due to unidentified leakage in excess of 1 gpm. The leak was identified as a failed tubing connection of the reactor vessel level indication system (RVLIS) connection on the 1B RCS hot leg. Following repairs to the RVLIS tubing connection, the unit was taken critical and entered Mode 1 on January 10, 2004. Several small power reductions occurred in January and February to perform integrated control system (ICS) tuning. The unit was reduced to approximately 86 percent RTP on March 6, 2004, to perform turbine valve testing, during which an EHC/ICS induced power transient occurred. The transient increased the unit's power output by 6 percent of RTP. For the remainder of the inspection period, the unit operated at or near 100 percent RTP.

Unit 2 entered the report period at 100 percent RTP. The unit was reduced to 99.5 percent RTP on February 3, 2004, due to a perceived mismatch between primary and secondary power indications. On February 23, 2004, a unit coast down was commenced in advance of the end-of-cycle (EOC) 20 refueling outage. The unit was shutdown from approximately 80 percent RTP on March 20, 2004, and remained shutdown for the remainder of the inspection period.

Unit 3 entered the report period at 100 percent RTP. The unit automatically tripped on February 26, 2004, due to high RCS pressure caused by the Number 2 main turbine stop valve closing from foreign material in the EHC system. The unit entered a forced outage to identify the cause of the trip and conduct repairs. Following repairs, the unit was taken critical on February 27, 2004, and returned to 100 percent RTP on February 29, 2004. For the remainder of the inspection period, Unit 3 operated at or near 100 percent RTP.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

Cold Weather Actual Conditions

a. Inspection Scope

The inspectors walked down cold weather protection features related to the protection of the borated water storage tanks (BWSTs) during periods of cold weather (<28F) that occurred on January 7, February 5, and February 17, 2004. The inspectors also walked down cold weather protection features related to the protection of the essential syphon system on February 17, 2004. The inspectors observed the freeze protection circuit panels associated with Units 1, 2 and 3 BWSTs to verify that the circuits were functioning properly with no circuits in the trip position. The inspectors utilized an infrared temperature measuring instrument to verify that heat trace circuits were operating above ambient temperatures as a quantitative measure that the freeze protection circuits were performing their functions. The inspectors also reviewed IP/0/A/1606/009, Preventive Maintenance and Operational Check of Freeze Protection, to verify that appropriate maintenance checks of freeze protection circuits, instrument enclosures, and insulation had been performed prior to the onset of cold weather. The inspectors verified that instrument enclosures were in place and piping insulation had been installed where appropriate.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

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 systems were included in this review:

- Unit 1 Emergency Feedwater (EFW) system during the performance of a surveillance on the 1A Motor Driven Emergency Feedwater (MDEFW) Pump
- Unit 1/Unit 2 Low Pressure Service Water (LPSW) system during preventive maintenance on the B LPSW Pump

- Unit 2 Auxiliary Steam (AS) line up to the Unit 1 Turbine Driven Emergency Feedwater (TDEFW) pump (when the main steam supply was not operable and only with AS was available for the Unit 1 TDEFW pump).

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Area Walkdowns

a. Inspection Scope

The inspectors conducted tours in twelve 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 3 Low Pressure Injection (LPI) Pump Rooms (1)

- CT-5 Transformer (Lee Station Power Path) (1)

- Unit 1, 2 and 3 Blockhouses (Keowee Hydro Units (KHUs) Underground power path via CT-4 Transformer) (2)

- CT-1, CT-2 and CT-3 Transformers (KHUs Overhead Power Paths) (3)

- Unit 1, 2 and 3 Equipment Rooms (3)

- Unit 1, 2 and 3 Control Rooms (2)

b. Findings

No findings of significance were identified.

.2 Fire Drill Observation

a. Inspection Scope

The inspectors observed a fire drill conducted on February 6, 2004, to assess the licensee's capability to fight fires. The fire was simulated at the Unit 3 lube oil purification skid. This was the first training exercise where the fire teams were practicing with fully installed self contained breathing apparatuses. The inspectors evaluated the following attributes:

- Protective clothing/self contained breathing apparatus properly worn
- Adequacy/appropriateness of fire extinguishing methods
- Controlled access to the fire area by the fire brigade members
- Adequacy of fire fighting equipment
- Command and control effectiveness of the fire brigade leader
- Adequate communications
- Effectiveness of smoke removal gear

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Simulator Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on February 24, 2004. The first scenario observed involved a dropped safety control rod and subsequent ICS asymmetric rod runback. During the runback, the main turbine tripped off-line without an associated reactor trip. A small break loss of coolant accident (LOCA) was initiated and cooling water flow was lost to the 1A MDEFW pump. The TDEFW pump was utilized to provide feedwater to the 1A once-through steam generator (OTSG), while the 1B MDEFW was utilized to provide feedwater to the 1B OTSG. Condenser vacuum was then lost, rendering the turbine bypass valves inoperable. This placed the simulated unit in a condition requiring a cool down with the atmospheric dump valves. The second scenario observed involved a steam generator tube leak on the 1B OTSG and subsequent reactor plant shutdown. During mitigation of the tube leak, all four RCPs remained operating and the turbine bypass valves were available with adequate condenser vacuum. This placed the simulated unit in a condition requiring a forced circulation cooldown with a steam generator tube leak on the 1B OTSG. The inspectors observed crew performance in terms of communications; ability to take timely and proper actions; prioritizing, interpreting, and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; and oversight and direction provided by the shift supervisor, including the ability to identify and implement appropriate Technical Specification (TS) actions.

b. Findings

No findings of significance were identified.

.2 Annual Operating Test Results

a. Inspection Scope

On March 19, 2004, the licensee completed the annual operating tests required to be given to all licensed operators by 10 CFR 55.59(a)(2). The inspectors reviewed the overall pass/fail results of the individual operating tests, and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee's effectiveness in performing routine maintenance activities. This review included an assessment of the licensee's practices pertaining to the identification, scoping, and handling of degraded equipment conditions, as well as common cause failure evaluations. For each item selected, the inspectors performed a detailed review of the problem history and surrounding circumstances, evaluated the extent of condition reviews as required, and reviewed the generic implications of the equipment and/or work practice problem. For those systems, structures, and components (SSCs) scoped in the maintenance rule per 10 CFR 50.65, the inspectors verified that reliability and unavailability were properly monitored and that 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. The inspectors reviewed the following items:

- Problem Investigation Process report (PIP) O-04-0613, Six Unit 1, 2 and 3 Unplanned TS Limited Condition for Operation (LCO) Action Statement Entries Due to RIA-47 Being Declared Inoperable With RIA-49 Inoperable

- The inspectors reviewed several problem reports due to continuing problems with the interface of the recently installed digital EHC system with the ICS system. The problems included the following:

  PIP O-04-00079 noted that the ICS turbine master station tripped to hand while performing post maintenance testing of the ICS following an unsuccessful modification of the EHC system.

  PIP O-04-00177 noted an unexpected Unit 1 power increase when the ICS turbine master tripped to manual.

  PIP O-04-00201 noted an unexpected Unit 1 power increase of 1.4 percent during return of ICS to automatic following ICS tuning at 90 percent power.

  PIP O-04-00205 noted that ICS was not responding to small incremental changes in the ICS demand setpoint.

  PIP O-04-00248 noted that the ICS turbine master would not transfer to automatic.

PIP O-04-00430 noted an EHC system trouble alarm and computer alarm due to EHC communication failure, which was caused by a deadband associated with the speed error discriminator. This caused a slight power transient of 0.3 - 0.4 percent.

PIP O-04-00644 noted Unit 1 ICS swings and noted that the ICS system was experiencing less stability than last cycle.

PIP O-04-00760 noted an EHC to operator aid computer (OAC) communication failure.

PIP O-04-00845 noted that the ICS turbine master station unexpectedly tripped to hand and reactor power increased to 100.2 percent.

PIP O-04-01075 noted multiple EHC to OAC communication error alarms.

PIP O-04-01189 noted that the ICS turbine master tripped to hand during the performance of the control valve movement test and reactor power unexpectedly increased 6 percent (reactor power had been reduced to 86 percent power for the control valve test).

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluations

a. Inspection Scope

The inspectors evaluated, as appropriate, for the selected SSCs 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-04-0309, Unexpected Unit 2 Automatic Feedwater Isolation System (AFIS) Alarms
- PIP O-04-0430, Unit 1 EHC System Trouble
- PIP O-04-0683, 2A OTSG EFW Train A Flow Indication Out of Tolerance
- Unit 2 Orange risk condition, Complex Plan for equipment lifts near main steam lines in Mode 3 or greater
- PIP O-04-1271, Temperature Rise on Inboard Packing of SSF Auxiliary Service Water (ASW) Pump
- Unit 2 CB battery out of service, for test and recharge longer than expected

- PIP O-04-1469, 3FDW-316 Backup Nitrogen Bottle Pressure Low

- PIP O-04-1610, The Station ASW Pump was Removed From Service Earlier Than Expected for Unit 2 Condenser Circulating Water (CCW) Work

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine 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-04-0017, Unit 1 NOUE Declaration Due to Unidentified RCS Leakage of 14 gpm While in Mode 3

- PIP O-04-0133, Unit 1 TS Required Shutdown Due to Unidentified RCS leakage in Excess of 1 gpm While in Mode 1

- PIP O-04-0845, ICS Turbine Master Unexpectedly Tripping to Hand

- PIP O-04-0991, Unit 3 Reactor Trip Due to High RCS Pressure Following the FME Induced Closure of the Number 2 Main Turbine Stop Valve

- PIP O-04-1036, Both Digital Channels of Unit 2 AFIS Declared Inoperable

- PIP O-04-1189 and PIP O-04-1190, Turbine Master Tripped to Hand During Performance of Control Number 1 Movement Test.

b. Findings

(1) Introduction: A Green self-revealing finding was identified for inadequate foreign material exclusion (FME) controls during EHC system filter replacement, which resulted in a Unit 3 turbine/reactor trip.

Description: On February 26, 2004, a Unit 3 reactor trip occurred after all the main steam stop valves (MSSVs) closed. The cause of the MSSVs to close was a small piece of plastic that became clogged in the pilot supply port of the Number 2 MSSV servo valve. The small piece of plastic was introduced into the system as a result of human error during EHC filter replacement that was completed a short time before the reactor trip. This piece of plastic was torn from the plastic wrapping on the new filter and was left adhered to the surface of the filter when it was installed. The inspection and self-checking of the maintenance technicians performing the work was not of sufficient rigor to identify the plastic foreign material prior to installing the filter cartridge into the system.

Analysis: The finding was considered to be more than minor because it resulted in a reactor trip which is considered a transient initiator in the SDP. This issue was determined to be of very low safety significance (Green) based on the Phase 1 SDP screening results, as the EHC foreign material, while causing a reactor trip, did not affect any mitigation equipment or functions.

Enforcement: This finding was not considered to be a violation of NRC requirements as the EHC system is not safety related and is not covered under 10 CFR 50, Appendix B.

(2) Introduction: A Green self-revealing NCV was identified for an inadequate primary sampling procedure, which resulted in a waterhammer-generated pressure wave that caused a thermal relief valve to lift. The relief valve stuck open, resulting in a 14 gpm unidentified RCS leak. The licensee's investigation of the event concluded the relief valve stuck open due to foreign material trapped between its seat and disc.

Description: On January 2, 2004, while in Mode 3 preparing for a reactor startup, numerous pressurizer water-space samples were drawn. At 8:12 p.m., indications of an excessive RCS leak were observed by the control room operating crew, and procedure AP/1/A/1700/002, Excessive RCS Leakage, was entered. The leakage was determined to be in excess of 10 gpm; consequently, a Notification of Unusual Event (NOUE) was declared at 8:38 p.m. The leak was identified as a stuck open thermal relief valve on the pressurizer water-space sample line and was quantified to be 14 gpm. The leak was isolated at 9:20 p.m., and an NRC event notification was made at 9:30 p.m. A licensee investigation concluded the cause of the event to be a lack of adequate procedural guidance to prevent waterhammer from occurring during the performance of procedure CP/1/A/2002/001, Primary Sampling System. Specifically, the Primary Sampling System procedure directs the operation of several sample valves in such a manner as to create a waterhammer of sufficient magnitude to exceed the 2790 psig setpoint of relief valve 1RC-206.

Analysis: The finding was considered to be more than minor because it affected the initiating event cornerstone, in that, the inadequate primary sampling procedure increased the likelihood of a small LOCA initiating event occurring. However, the stuck open thermal relief valve was readily isolable by the excessive RCS leakage procedure, consequently, the finding screened out of the SDP Phase 1 analysis as Green (very low safety significance).

Enforcement: TS 5.4.1 requires that written procedures be established, implemented, and maintained covering activities related to procedures recommended in Regulatory Guide 1.33 Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Section 2(g), General Plant Operating Procedures, requires procedures for Power Operation and Process Monitoring. Contrary to the above, the licensee utilized an inadequate procedure to sample the RCS, as the procedure failed to prevent waterhammer from occurring; thereby, lifting the thermal relief valve. Because the inadequate primary sampling procedure is of very low safety significance and has been entered into the licensee's corrective action program as PIP O-04-0017, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000269/ 2004002-01, Inadequate Primary Sampling Procedure Results in Excessive RCS Leakage and a NOUE.

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. The inspectors reviewed the following items for operability evaluations:

- PIP O-04-0197, Unit 1, 2 and 3 TS LCO 3.0.3 Action Statement Entry Due to a Loss of Both Control Room Chillers
- PIP O-04-0362, Safety Analysis of Unit 1 Replacement OTSG Operating Levels
- PIP O-04-0371, Unit 1 Reactor Building (RB) Tendons not Reinstalled per Specification Following Repairs to Unit 1 RB After OTSG Replacement
- PIP O-04-0564, SSF Submersible Pump Motor's Power Leads Swapped
- PIPs O-02-3173, RB Coating Degradation; O-03-5918, Paint on Polar Crane Needs to be Redone; O-03-7413, Flaking Paint on Containment Liner Plate With no Corrosion; O-03-7707, Several Paint Chips Were Observed Falling Into and Around the Core; and O-04-0743, Valve Installed in Containment With Unqualified Coatings Applied
- PIP O-03-5237, Water in the SSF ASW Pump Bearing Housing
- PIP O-04-0853, Unit 3 Misalignment of Control Rod Group 1, Rod 1
- PIP O-04-1059, Loss of Power to an Analog Voltage Isolation Module in One AFIS Digital Channel Affects Main Steam Pressure Signal to the Other Digital Channel

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed an operator workaround to determine if the functional capability of the 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 if the workaround could not be properly performed what impact it would have on the unit.

The workaround reviewed was documented by Work Order (WO) 98473463, Preventive Maintenance (PM) on 1CS-173 isolated makeup capability from the 1B and 2B Bleed Holdup Tanks to the Unit 1 and 2 Letdown Storage Tanks, respectively. Therefore, makeup flow path from 3B BHUT via a rarely used cross-connect flow path was utilized.

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 tests:

- IP/0/A/3000/003, 125 VDC Instrument and Control Battery Service Test and Annual Surveillance, cleaning and inspection of battery cell terminals and inter-cell connectors

- IP/0/A/3000/026, Battery Cell Connections Resistance Test, and BCT-2000, Battery Load Test Report, following the replacement of Cells 33 and 49 of the 3CB Battery

- PT/1/A/0251/001, Low Pressure Service Water Pump Test, following a test and inspection of the 1/2C LPSW pump motor and a lubrication PM on the pump

- PT/0/A/0400/015, SSF Submersible Pump Test, following an overhaul by an offsite vendor

- PT/3/A/0251/001, LPSW Pump Test, following a test and inspection of the 3A LPSW pump motor, a backwash of the pump's suction strainer, lubrication of the pump, and the replacement of the expansion joint on the suction side of the pump

- PT/3/A/0600/013A, MDEFW Pump Test, following a lubrication PM on the pump

- PT/3/A/0251/001, LPSW Pump Test, following a test and inspection of the 3B LPSW pump motor, a backwash of the pump's suction strainer, lubrication of the pump, and the replacement of the expansion joint on the suction side of the pump

- Control room chiller testing associated with WO98587487-01, Perform 50,000 Hour Maintenance of A Chiller

- PT/0/A/0400/005, SSF ASW Pump Test, following ASW pump packing adjustment

b. Findings

Introduction: An unresolved item (URI) was identified regarding the incorrect wiring of the primary SSF submersible pump's motor leads, which would have caused the pump to rotate in the reverse direction for actual events when the submersible pump would have been powered from the SSF power supply. This issue is being identified as an URI pending a Phase 3 risk evaluation.

Description: On February 3, 2004, the licensee performed a PMT of the SSF submersible pump following a complete overhaul of the pump by a vendor. During the PMT, it was noted by a technician that the pump's impeller was rotating in the reverse direction; thereby indicating that the pump's motor leads were wired incorrectly. The licensee checked the rotation of the spare/backup submersible pump's impeller and found it to be incorrect. A licensee investigation concluded that the primary SSF submersible pump's motor leads had been wired incorrectly since November 19, 1992, and the backup SSF submersible pump's motor leads had been wired incorrectly since March 13, 1997, and prior to April 3, 1995. During these respective times, the licensee performed twelve surveillance's on the primary submersible pump (six surveillance's on the backup submersible pump), and failed to identify that either pump's motor leads were wired incorrectly also. During surveillance testing of the submersible pump at the CCW intake structure, the pump is powered from a local receptacle which is not the power supply that would be used in an actual event. This may have factored into the incorrect wiring of the motor leads not being identified during surveillance testing. The licensee also concluded that a May 14, 1998, revision to the emergency operating procedure for installing and removing the SSF submersible pump, MP/0/A/1300/059, eliminated a step to verify that the pump rotated in the correct direction prior to lowering it into a CCW intake bay.

Analysis: Failure to maintain the SSF submersible pump in a ready to operate condition was considered to be more than minor; in that, the SSF submersible pump with incorrectly wired motor leads directly affected the equipment reliability of a mitigating system (i.e., the SSF). The finding was evaluated in an SDP Phase 1 analysis due to the degraded reliability of a mitigating system under the Reactor Safety Cornerstone. For the Phase 1 analysis it was assumed that the finding represented an actual loss of the safety functions of the SSF. This was based on the conclusion that the incorrectly wired motor leads of the SSF submersible pump would adversely affect the licensee's ability to provide timely makeup flow to the Unit 2 CCW header. An SDP Phase 2 analysis was performed which indicated that the significance of this finding could be greater than Green (very low safety significance); therefore, a Phase 3 analysis is required.

Enforcement: 10 CFR 50, Appendix B, Criterion XI, requires, in part, that a test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, the licensee failed to establish a test program capable of demonstrating that the SSF submersible pump would operate correctly when called upon, in that, the pump's power leads were reversed. This finding does not represent an immediate safety concern since both the primary and backup SSF submersible pumps' motor leads have been rewired correctly and the pumps have been verified to rotate in the correct direction. In addition, the licensee has issued guidance requiring a post-maintenance test to verify proper pump rotation using the submersible pump's cable and its remote starter following any work that could affect the pump's motor leads or how the cable wiring is connected. Pending determination of this finding's safety significance, it will be identified as: URI 05000269, 270,287/2004002-02, Incorrect Wiring of the SSF Submersible Pump's Motor Leads. This issue is in the licensee's corrective action program as PIP O-04-0564.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors conducted reviews and observations for selected outage activities to ensure that: (1) the licensee considered risk in developing the outage plan; (2) the licensee adhered to the outage plan to control plant configuration based on risk; (3) that mitigation strategies were in place for losses of key safety functions; and (4) the licensee adhered to operating license and TS requirements. Between March 20, 2004, and March 27, 2004, the following activities related to the Unit 2 refueling outage were reviewed for conformance to applicable procedures and selected activities associated with each evaluation were witnessed:

- Outage risk management plan/assessment
- Plant cooldown
- Mode changes from Mode 1 (power operation) to Mode 6 (refueling)
- Shutdown decay heat removal and inventory control

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of the selected risk-significant SSCs listed below, to assess, as appropriate, whether the SSCs met TS, Updated Final Safety Analysis Report (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/0/A/0620/009, Keowee Hydro Operation
- PT/0/A/0400/001, SSF Diesel Generator Test
- PT/1/A/0600/013, 1A Motor Driven EFW Pump Test
- PT/1/A/0150/022M, 1FDW-315 and 1FDW-316 Stroke Test
- UFSAR Section 3.6.1, MDS Report OS-73.2, Main Feedwater System and Main Steam System Support Inservice Inspections
- PT/0/A/0205/005, Unit 2 Thermal Power and Reactor Coolant Flow Calculations
- PT/1/A/0600/012, Unit 1 Turbine Driven EFW Pump Test

b. Findings

Introduction: A Green NCV was identified by the inspectors for failure to establish an adequate inspection program for certain feedwater (FW) system and main steam (MS) system piping supports associated with high energy line break scenarios.

Description: The inspectors noted that in the licensee's April 25, 1973, response to the Giambusso letter related to General Design Criteria 4 (which was included as part of the original Final Safety Analysis Report (FSAR) for high energy line break) the licensee responded that they would perform inspections of certain FW system and MS system piping supports every four years. During a review of the licensee's piping support inspection activities, the inspectors noted that some of these FW and MS supports were not in the licensee's inservice inspection (ISI) program for supports, some of the snubbers had not been inspected in approximately 30 years, and that the majority of the supports were not being inspected on a 10 year interval.

The licensee initiated PIP O-04-01226 and described how the inspection requirements were documented in MDS Report OS-73.2, "Analysis of Effects Resulting from Postulated Breaks Outside Containment." The licensee noted that upon completion of Oconee's ongoing HELB design basis revalidation effort, a License Amendment will address this discrepancy. In addition, the licensee has placed all of these supports in a special ISI program to ensure inspections would be conducted on each support at least once per ten year intervals.

Analysis: The lack of these piping support inspections and resulting inability to identify adverse conditions related to the supports could affect the ability of the feedwater and/or the main steam line systems to withstand various events such as seismic induced loading, because potential problems would go undetected/ corrected. This in turn could result in damage to other mitigation systems. This issue was considered to be more than minor because if left uncorrected it could prevent the detection of piping support defects which would increase the probability of an initiating event (feed line and steam line rupture). A Phase 1 evaluation was conducted using the initiating event screening criteria. Because the inadequate inspection of the supports had not caused an actual increase the likelihood of an initiating event, the issue was screened out as Green. The lack of a previous increase in the likelihood of an initiating event was based on no significant loading events, such as a seismic event, having occurred. If subsequent licensee inspections note degradation of the supports, then the specific problem will be evaluated at that time as a potential contributor for increasing the likelihood of initiating a secondary system LOCA.

Enforcement: 10 CFR 50, Appendix B, Criterion X, Inspection, requires, in part, that a program for inspection of activities affecting quality shall be established and executed. Contrary to the above, the licensee failed to establish and execute an adequate program for inspection activities affecting quality, in that, they failed to include certain feedwater and steam line system supports associated with high energy line break scenarios in the program, and they failed to inspect the supports in accordance with their response to the Giambusso letter which is included as part of the initial FSAR. Because this failure to comply with 10 CFR 50, Appendix B, Criterion X, is of very low safety significance and has been entered into the licensee's corrective action program as PIP O-04-01226, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000269,270,287/2004002-03, Inadequate Inservice Inspection Program for Inspections of Feedwater System and Steam Line System Supports.

1R23 Temporary Modifications

a. Inspection Scope

The inspectors reviewed documents and observed portions of the installation of temporary modification MP/0/A/3007/062, Temporary Cooling of Chilled Water (WC) System With Portable Chiller - Installation, Operation, and Removal. 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 and was in accordance with the configuration control process; that adequate procedural changes were made; and that post installation testing was adequate.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP1 Exercise Evaluation

a. Inspection Scope

The inspectors reviewed the emergency exercise and scenario for the biennial, full participation 2004 emergency response exercise for the Oconee Nuclear Station. The review covered whether the licensee created a scenario suitable to test the major elements of their emergency plan in accordance with 10 CFR 50, Appendix E.

Licensee activities inspected during the exercise include observations in the Control Room Simulator, Emergency Off-site Facility, Technical Support Center, and Operational Support Center. The exercise was conducted on January 13, 2004. The inspectors reviewed a sample of corrective actions identified in the past and developed a list of areas to be observed in this exercise. The inspectors' evaluation focused on the risk-significant activities of event classification, notification of governmental authorities, onsite protective actions, offsite protective action recommendations, and accident mitigation. The inspectors also evaluated command and control, the transfer of emergency responsibilities between facilities, communications, adherence to procedures, and the overall implementation of the emergency plan. The inspectors attended the post-exercise critique to evaluate the licensee's self-assessment process, as well as the presentation of critique results to plant management.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes

a. Inspection Scope

The inspector reviewed the changes made to Oconee Nuclear Site Emergency Plan, Revision 2004-01, dated March 3, 2004, against the requirements of 10 CFR 50.54(q) to determine whether any of the changes decreased the effectiveness of the Oconee Nuclear Station Emergency Plan.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspectors evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in emergency preparedness. The procedure governing the plant corrective action program was reviewed for applicability to the emergency preparedness program. The inspectors reviewed this data bases for any item that may have been added since the last inspection of this program area which was conducted in September 2003.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Control: The inspectors evaluated the licensee's activities for monitoring and controlling worker access to radiologically significant areas during the Unit 2 (U2) refueling outage (RFO), steam generator replacement (SGR), and reactor vessel head replacement (RVHR), as well as selected activities associated with the previous Unit 1 (U1) RFO, SGR and RVHR not reviewed during the previous NRC inspection. The inspection included direct observation of administrative and physical controls, appraisal of the knowledge and proficiency of radiation workers and health physics technicians (HPTs) in implementing radiological controls, and review of the adequacy of procedural guidance and its implementation.

The inspectors observed implementation of radiological controls for selected U1, U2, and Unit 3 (U3) Radiation Areas (RA), Radioactive Material Areas, and High Radiation Areas (HRA) within Radiologically Controlled Area (RCA) locations. Postings and labeling of materials at these locations were evaluated for consistency with the licensee's procedural guidance and compliance with NRC regulatory requirements. The inspectors directly observed the posting and locking status of the one Very High Radiation Area (VHRA) in the U2 containment at the time of the onsite inspection, and selected HRAs and Extra High Radiation Areas (EHRAs) in U1, U2 and U3 Auxiliary Buildings. Independent dose-rate measurements were taken in the U2 Reactor Building, and the results of those measurements were compared to current licensee surveys. In addition, the inspectors toured and reviewed radiological controls for the Steam Generator Retirement Facility, which was located outside the plant's protected area but within the owner-controlled area and evaluated procedures and controls for radioactive material stored in the spent fuel pools. Health Physics (HP) supervisory personnel, including the Radiation Protection (RP) Manager, were interviewed regarding the administrative control of EHRA and VHRA keys, as well as any changes to procedural guidance for access control.

The inspectors reviewed selected Radiation Work Permits (RWPs) and surveys to evaluate the adequacy of radiological controls for RAs, HRAs, and airborne areas. The inspectors evaluated the use of radiological controls, observed the performance of HPTs and radiation workers, evaluated RWP requirements and electronic dosimeter alarm setpoints, and discussed various task evolutions with selected personnel associated with SGR and RVHR activities. During general observations of U2 RFO work, the inspectors discussed with select radiation workers RWP requirements associated with their tasks in progress. Discussions with licensee personnel were held regarding internal doses over the past year, although none had exceeded 50 mrem committed effective dose equivalent (CEDE).

During the onsite inspection, the inspectors directly observed licensee activities associated with the decontamination and inspection of the U2 reactor vessel annulus. Procedural guidance regarding the use of supplied-air bubble suits, training, and guidance on the availability of standby rescue personnel were reviewed and discussed with cognizant licensee representatives associated with the U2 annulus work. In addition, the inspectors reviewed the licensee's process for key control associated with this activity since the annulus area was designated by the licensee as a VHRA.

RP program activities and their implementation were evaluated against 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, I, and J; Updated Final Safety Analysis Report (UFSAR) Section 12.4, RP Program; and licensee approved procedures. Licensee procedures, records, and other documents reviewed within this inspection area are listed in Attachment to this report.

Issues identified through department self-assessments, Functional Area Evaluation audits, and Problem Investigation Process (PIP) documents associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with cognizant licensee representatives. Specific areas reviewed included TLD and ED usage/loss, personnel contamination events, and RWP usage. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure Duke Power Quality Assurance Program Related, Nuclear Policy Manual, Nuclear System Directive (NSD) 208, Problem Investigation Process Report, Revision (Rev.) 25, dated May 13, 2003. Specific assessments, audits, and PIP

Documents reviewed and evaluated in detail for this inspection area are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

On January 12-14, 2004, licensee records were reviewed to determine whether the submitted PI values through the third quarter of 2003 for the following PIs were calculated in accordance with the guidance contained in Section 2.4 (Emergency Preparedness Cornerstone) of NEI 99-02, Regulatory Assessment Performance Indicator Guideline Revision 2. Additionally the inspectors reviewed the completed fourth quarter data for 2003.

• Emergency Response Organization (ERO) Drill/Exercise Performance
• ERO Drill Participation
• Alert and Notification System Reliability

The inspector assessed the accuracy of the PI for ERO drill and exercise performance over the past eight quarters by reviewing a sample of drill and event records. The inspector reviewed training records to assess the accuracy of the PI for ERO drill participation during the previous eight quarters for personnel assigned to key positions in the ERO. The inspectors assessed the licensee's ability to notify members of the public within the 10-mile EPZ.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

As required by 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.

The inspectors performed an in-depth review of PIP O-02-3173, Degradation of Reactor Building (RB) Qualified Coatings. This issue dealt with the mitigating systems cornerstone and involved a risk significant system. The inspectors reviewed the actions taken to determine if the licensee had adequately addressed the following attributes:

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

b. Findings

Introduction: An unresolved identified (URI) was identified regarding degraded coatings within the Unit 1, 2 and 3 RBs. This issue is designated as an URI pending further inspection and assessment of the affect on the reactor building emergency sump (RBES).

Description: On November 11, 1998, the licensee responded to Generic Letter 98-04, Potential for Degradation of Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment. The licensee's response to the NRC stated:

> "Oconee conducts periodic condition assessments of Service Level 1 coatings used inside containment. These assessments are used for evaluating the condition of in-service coating systems and are performed during each refueling outage. As localized areas of degraded coatings are identified, those areas are evaluated and scheduled for repair or replacement, as necessary. The periodic condition assessments and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings that may be susceptible to detachment from the substrate during a LOCA event is minimized."

The response also discussed the applicability of 10 CFR 50, Appendix B to RB coatings, as well as documented the degree/amount of degraded coatings within each RB and the date of the last coatings inspections (at the time of the response).

During the Unit 1 EOC21 refueling outage in the Fall of 2003, the inspectors discovered what appeared to be a significant amount of Service Level 1 coatings that were severely blistered, delaminated, peeling and falling off of the RB dome and liner, polar crane, and sprinkler grid support assembly. Similar degraded coating conditions were discovered by the inspectors during the current Unit 2 EOC20 refueling outage and the Unit 3 forced outage following its February 26, 2004, reactor trip. The licensee is currently in the process of quantifying the amount of degraded coatings in the Unit 2 RB and developing a restoration plan that will be implemented during this refueling outage. Licensee assessments of the coating in the Unit 1 and 3 RBs are also in progress.

Analysis: During accident conditions, degraded coatings could be released and transported to the RBES; thereby, potentially clogging the sump's screens. This could impact the safety function of the RBES during accident scenarios that require sump recirculation, because clogged sump screens could result in a loss of net positive suction head (NPSH) to the unit's LPI and BS pumps.

Enforcement: This issue will remain unresolved pending further inspection to determine: (1) what impact the degraded coatings may have had on the three Units' RBES; and (2) if the licensee took adequate corrective action to ensure the amount of coatings that may be susceptible to detachment from the RB substrate during a LOCA event was minimized. Accordingly, it will be identified as: URI 05000269,270,287/2004002-04, Potential Failure to Maintain RB Coatings per GL 98-04 Commitments, Resulting in

Potential Loss of RBES Recirculation. This issue is in the licensee's corrective action program as PIP O-02-3173.

4OA3 Event Followup

.1 Recent Events

a. Inspection Scope

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

- PIP O-04-0995, Unit 3 Reactor Trip - [This event was discussed in Section 1R14.b.(1) of this report.]

- PIP O-04-1036, Both Digital Channels of Unit 2 Automatic Feedwater Isolation System (AFIS) Declared Inoperable - [The licensee later determined that only one digital channel of AFIS had been inoperable and the 10 CFR 50.72 report was withdrawn.]

b. No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 269/2004-01-(00 and 01), Unit Shutdown Due to Reactor Coolant Leak Above Technical Specification Limits. The licensee determined that the leaking RVLIS sensing line was due to a high localized stress condition due to crimping of the tubing ferrule (expected), combined with possible additional stress due to a slight bend (within limits), and the fact that the tubing wall thickness was less than that called for in the design specification (still met ASME code allowable). In combination with these conditions, expected vibration led to the cracking of the sensing line and the subsequent leak. No violations of regulatory requirements were identified during this review.

4OA5 Other Activities

.1 Independent Spent Fuel Storage Installation (ISFSI)

a. Inspection Scope

Access control and surveillance results for the licensee's ISFSI were evaluated by the inspectors. The evaluation included a review of ISFSI radiation control surveillance procedures and assessment of radiological survey data. The inspectors toured the ISFSI and observed access controls, TLD placement, and radiological postings on the perimeter security fence. The inspectors observed a licensee technician perform gamma radiation surveys of the ISFSI at the procedurally designated location. Survey results were compared to the most recent survey records by the inspectors.

Program guidance, access controls, postings, equipment material condition and surveillance data results were reviewed against applicable sections of the cask Certificate of Compliance, Safety Analysis Report, ISFSI TS, 10 CFR Parts 20 and 72, and applicable licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in the Attachment to the report.

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>Steam Generator Replacement (SGR) Inspection and Reactor Vessel Head Replacement (RVHR) Inspection</u>

a. <u>Inspection Scope</u>

In addition to the SGR and RVHR activities described in Section 2OS1, the inspectors evaluated the licensee's "As Low As Reasonably Achievable" (ALARA) program guidance and its implementation for ongoing job tasks during the U2 RFO to include those activities associated with the SGR and RVHR. The inspectors reviewed, and discussed with licensee staff, ALARA planning, dose estimates, and prescribed ALARA controls for selected outage work activities expected to incur significant collective doses. Those activities included grout removal on steam generator (SG) A insulation removal on SG B, scaffold installation associated with the staging area for the RVHR activities, and reactor vessel annulus decontamination and inspection. Also reviewed were the implementation of dose-reduction initiatives for high person-rem-expenditure tasks and assessment of the effectiveness of source-term reduction efforts. These elements of the ALARA program were evaluated for consistency with the methods and practices delineated in applicable licensee procedures.

The implementation and effectiveness of ALARA planning and program initiatives during work in progress were evaluated. The inspectors made direct field and closed-circuit television observations of work activities involving the disassembly of the old reactor head and SG replacement work. Projected dose expenditure estimates detailed in current ALARA planning documents were compared to actual dose expenditures, and noted differences were discussed with cognizant ALARA staff. Changes to dose budgets relative to changes in job scope also were discussed. The inspectors attended pre-job briefings associated with the decontamination and inspection of the reactor vessel annulus and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to the job crew personnel by the ALARA Supervisor.

Implementation and effectiveness of selected program initiatives with respect to source-term reduction were evaluated. Dose rate trending data was reviewed and discussed with the ALARA Coordinator. The inspectors reviewed the licensee's process for generating and evaluating shielding requests. The effectiveness of selected shielding packages installed for the current outage was assessed from a review of survey records as well as direct observations made by the inspectors during tours of the licensee's facilities. During the onsite inspection, the inspectors reviewed the licensee's actions taken in response to a hot spot that was identified on pressurizer surge line.

RP program activities and their implementation were evaluated against 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, and J; and approved licensee procedures. In addition, licensee performance was evaluated against Regulatory Guide (RG) 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable.

Licensee PIP documents associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with responsible licensee representatives. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with NSD 2O5, PIP. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.3 Unit 2 Reactor Vessel Head Replacement Project

a. Inspection Scope

The inspectors observed work activities and reviewed fabrication records for the Unit 2 RVHR. Activities and records were inspected to the requirements of the applicable fabrication and inspection codes. The code of record for the replacement activities was the ASME B&PV Code Section XI, 1989 Edition with No Addenda, supplemented by the ASME B&PV Code Section XI, 1998 Edition with 2000 Addenda for nondestructive examinations (NDE), ASME B&PV Code Section III-NF, 1989 Edition with No Addenda, and AWS D1.1, 2000 Edition for supports and structures.

Material certifications and portions of the fabrication records were reviewed for the RVHR forging, lifting lugs, control rod drive mechanism (CRDM) flanges, and CRDM guide tubes. Fabrication records reviewed included the results of nondestructive examinations (NDE) for welds of the sixty-nine (69) CRDM guide tubes to the replacement reactor head and heat treatment records for the replacement head assembly.

The inspectors reviewed the work orders for the installation of supporting plates for the service structure for CRDM. The inspectors reviewed documents such as head replacement specifications, heavy load rigging and handling procedures, the pendant load test record, drawings, calculations, and modification packages, 10 CFR 50.59 Screening, and a 10 CFR 50.65(a)(4) assessment for the activity.

In addition to reviewing the documentation, the inspectors independently measured weld, bolt, and member sizes for the installed service structure components. The inspectors observed portions of the lifting tower installed outside the planned containment opening for the outside lifting devices to move the Reactor Head in and out from the containment.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. Dave Baxter, Engineering Manager, and other members of licensee management at the conclusion of the inspection on April 1, 2004. 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.

.2 Annual Assessment Meeting Summary

On April 13, 2004, the NRC's Deputy Regional Administrator and the Chief of Reactor Projects Branch 1, met with Duke Energy to discuss the NRC's Reactor Oversight Process and the Oconee Nuclear Station (ONS) annual assessment of safety performance for the period of January 1, 2003 - December 31, 2003. The major topics addressed were the NRC's assessment program and the results of the ONS assessment. Attendees included ONS site management, members of site staff, and members of the public and local news media.

This meeting was open to the public. The presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML041120482. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Batson, Mechanical/Civil Engineering Manager
J. Batton, Oconee Steam Generator Engineer
D. Baxter, Engineering Manager
J. Brackett, SGT Reactor Vessel Head Project Coordinator
R. Brown, Emergency Preparedness Manager
N. Constance, Operations Training Manager
D. Covar, Training Instructor
C. Curry, Maintenance Manager
T. Curtis, Reactor & Electrical Systems Manager
G. Davenport, Compliance Manager
C. Eflin, Requalification Supervisor
P. Fowler, Access Services Manager, Duke Power
T. Gillespie, Operations Manager
B. Hamilton, Station Manager
R. Hester, Civil Engineer
B. Jones, Training Manager
R. Jones, Site Vice President
T. King, Security Manager
B. Lowrey, Steam Generator Engineer
B. Millsaps, SGT Maintenance Manager
L. Nicholson, Safety Assurance Manager
R. Repko, Superintendent of Operations
R. Sharpe, Lead Licensing Engineer, Steam Generator Replacement
J. Smith, Regulatory Affairs
J. Steeley, Training Supervisor
T. Tucker, NDE Level III Examiner
J. Twiggs, Manager, Radiation Protection
R. Waterman, Emergency Planning Specialist
J. Weast, Regulatory Compliance

NRC

R. Haag, Chief of Reactor Projects Branch 1
L. Olshan, Project Manager
L. Plisco, Deputy Regional Administrator

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

050000269,270,287/2004002-02	URI	Incorrect Wiring of the SSF Submersible Pump's Motor Leads (Section 1R19)
050000269,270,287/2004002-04	URI	Potential Failure to Maintain RB Coatings per GL 98-04 Commitments, Resulting in Potential Loss of RBES Recirculation (Section 4OA2)

Opened and Closed

05000269/2004002-01	NCV	Inadequate Primary Sampling Procedure Results in Excessive RCS Leakage and a NOUE (Section 1R14.b.(2))
05000269,270,287/2004002-03	NCV	Inadequate Inservice Inspection Program for Inspections of Feedwater System and Steam Line System Supports (Section 1R22)

Previous Items Closed

LER 269/2004-01-(00 and 01)	LER	Unit Shutdown Due to Reactor Coolant Leak Above Technical Specification Limits (Section 4AO3)

Items Discussed

None

DOCUMENTS REVIEWED

(Sections 1EP1, 1EP4, and 1EP5: Emergency Preparedness)

RP/0/B/1000/001, Emergency Classification, Rev. 14
RP/0/B/1000/015B, Offsite Communications from the Technical Support Center, Rev. 3
RP/0/B/1000/024, Protective Action Recommendations, Rev. 3
SH/0/B/2005/003, Distribution of Potassium Iodide Tablets in the Event of a Radiological Release, Rev. 0 (scheduled to be implemented on 01/20/2004)
Oconee Nuclear Site Emergency Plan, Revision 2004-01

(2OS1: Access Control To Radiologically Significant Areas)

Procedures

Duke Power Company (DPC), Radiation Dosimetry and Records, Nuclear Site Area Monitoring, Procedure Number (No.) RD/0/B/4000/15, Undated
Duke Power Quality Assurance Program Related (DPQAPR), Nuclear Policy Manual (NPM), Nuclear System Directive (NSD): 208, Problem Investigation Process (PIP), Revision (Rev.) 25, Dated 05/13/03
DPQAPR, NPM, NSD: 223, Trending Program, Rev. 4, Dated 07/10/03
Oconee Nuclear Station (ONS), Non-Routine Surveillance Requirements, Procedure No. HP/0/B/1000/093, Rev. 015, Undated
ONS, Radiation Protection Routines, Procedure No. HP/0/B/1000/054, Rev. 38, Undated
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. 001, Undated
SPOMCNS, Preparation of a Radiation Work Permit (RWP), Procedure No. SH/0/B/2000/003, Rev. 006, Undated

RWPs

RWP 2, Entry for Routine Surveillance, Rev. 21, Dated 02/25/04
RWP 15, Routine Radiological Surveys, Rev. 15, Dated 02/25/04
RWP 5095, U-1 & 2 Spent Fuel Pool, Cask, Storage Building, FRA, Horizontal Storage Module (HSM), Prep, Load, Transport, and Storage of HSM Load No. 78, Rev. 0, Dated 01/05/04

Records and Data

Appendix A. 223. Group Trending Report, Group/Quarter RPS/3rd, 2003, Dated 11/25/03
Appendix A. 223. Group Trending Report, Group/Quarter RPS/4th, 2003, Dated 02/25/04
ONS Nuclear Site Area Monitoring Results for Calendar Year 2003
“A” J Leg Weld Locations, Survey No. 032304-1, Dated 03/23/04

PIPs

PIP No. O-03-01483, Contaminated Material Found in a Clean Area
PIP No. O-03-03733, Contaminated Material was Found in a Clean Area
PIP No. O-03-03744, Due to a Personnel Contamination Event the Floor for Room 304 was Found to be Contaminated
PIP No. O-03-06051, Worker Entered U1 Reactor Building on RWP 1 Rather Than the Correct RWP of 6100
PIP No. O-03-06381, Workers Were on the Wrong RWP When They Crossed an EHRA Boundary Without Job Coverage RP Approval
PIP No. O-03-06407, Workers Entered the Reactor Building on RWP 1297 but the Correct RWP was 1032
PIP No. O-03-06504, Worker Failed to Log Onto Correct RWP Before Entering the U1 Reactor Building
PIP No. O-03-06690, An Individual was Observed Exiting the RCA Without Monitoring His Coat
PIP No. O-03-07036, An RCA Barrier was Found to be Down Around a Section of 1 CTP Due to a Clump of Brush Becoming Entangled in the Barrier Rope

PIP No. O-03-07336, An Individual Crossed an Radiography Boundary
PIP No. O-03-07479, SGT Worker Experienced a Dose Rate Alarm Due to Use of Wrong RWP
PIP No. O-03-07523, Worker Entered U1 Reactor Building on Invalid RWP
PIP No. O-03-07576, Worker Left Site While Still Logged Onto an RWP
PIP No. O-03-07825, Radioactive Material was Found Outside the RCA
PIP No. O-04-00426, Radioactive Material was Found in an Unauthorized Area
PIP No. O-04-01518, Personnel Contamination Event No. 04-008, Skin Contamination on an Individuals Neck
PIP No. O-04-01606, Evaluation of Letdown Filter Change During the Crud Burst Activity on 2EOC20

(Section 4OA5.1 and 4OA5.2: Other Activities)

Procedures, Guidance Documents, and Manuals

Duke Power Company System ALARA Manual, For Use in the Design, Construction, Operation, and Decommissioning of Nuclear Power Stations and Supporting Facilities, Undated
DPQAPR, NPM, NSD: 208, PIP, Rev. 25, Dated 05/13/03
DPQAPR, NPM, NSD: 223, Trending Program, Rev. 4, Dated 07/10/03
ONS, Procedure for Quantifying Airborne Radioactivity, Procedure No. HP/0/B/1010/057, Rev. 031
ONS, Radiological Protection Requirement for Independent Spent Fuel Storage Installation (ISFSI) Phase III and IV (DSCs 41-84), Procedure No. HP/0/B/1000/097, Rev. 4, Undated
ONS, Radiological Protection Requirements for SG Maintenance, Procedure No. HP/0/B/1000/016, Rev. 020
ONS, Respirator Equipment Issue and Use Procedure, Procedure No. HP/0/B/1010/001, Rev. 24
ONS, Selection of Proper Respiratory-Protective Equipment and Respiratory Surveillance Requirements, Procedure No. HP/0/B/1010/004, Rev. 24
SPOMCNS, Access Controls for High, Extra High, and Very High Radiation Areas, Procedure No. SH/0/B/2000/012, Rev. 001
SPOMCNS, Investigation of Skin and Clothing Contaminations, Procedure No. SH/0/B/2001/003, Rev. 006
Task # RP-078-O, ETQS, Training and Qualification Guide, Titled “Low Exposure Job Coverage”, Rev. 22, Dated 03/03/04

ALARA Pre-Planning and Planning Worksheets

CRDM Removal/Prep/Inspect/Transport/Install for RWP No. 6278, Dated 02/13/04
Entry into Annulus Area for Bare Metal Inspection and Mode 3 Inspection to Include All Support Tasks (Insulation, Scaffold & Wash Down) for RWP No. 2026, Dated 03/02/04
Install/Remove Scaffolding for RWP No. 6205, Dated 02/11/04
Load, Weld Transport and Store ISFSI Load No. 78 Long Cavity Cask with RWP 5094, Dated 01/01/04
Load, Weld Transport and Store ISFSI Load No. 79 Long Cavity Cask with RWP 5095, Dated 02/01/04
NSM-23093 U2 LPI Passive Cross-Connect Modification for RWP No. 2098, Dated 03/14/04
RCS Pipe Cutting/Machining/Welding for RWP No. 6210, Dated 02/11/04
Remove/Install RCS Temporary/Permanent Supports for RWP No. 6208, Dated 02/12/04

RWPs

RWP No. 2026, Rev. 8, Job Title: U2 Reactor Building Annulus Inspections and Associated Work; Dated 03/13/04
RWP No. 6100, Rev. 0, Job Title: U1 - SG/RHA - Inspections, Surveillance & Firewatch, Dated 09/10/03
RWP No. 6102, Rev. 0, Job Title: U1 - SGRP - Radiography Testing, Dated 09/10/03
RWP No. 6103, Rev. 0, Job Title: U1 - SG/RHA - Miscellaneous Support Activities, Dated 09/10/03
RWP No. 6104, Rev. 0, Job Title: U1 - SG/RHA - Install/Remove Lead Shielding & Decontamination Activities, Dated 09/10/03
RWP No. 6105, Rev. 0, Job Title: U1 - SG/RHA - Install and Remove Scaffolding, Dated 09/10/03
RWP No. 6106, Rev. 0, Job Title: U1 - SGRP - Install and Remove Insulation, Dated 09/10/03
RWP No. 6107, Rev. 0, Job Title: U1 - SGRP - RCS Pipe End Decontamination, Dated 09/10/03
RWP No. 6108, Rev. 0, Job Title: U1 - SGRP - Remove/Install RCS/Lateral Supports/ Restraints, Dated 09/10/03
RWP No. 6109, Rev. 0, Job Title: U1 - SGRP - Remove/Install MS, FDW, EFDW Piping and Associated Seal Plates, Dated 09/10/03
RWP No. 6110, Rev. 0, Job Title: U1 - SGRP - RCS Pipe Cutting/Machining/Welding, Dated 09/10/03
RWP No. 6111, Rev. 0, Job Title: U1 - SGRP - Remove/Install Drain Piping, Secondary Piping and RCS Instrumentation Lines, Dated 09/10/03
RWP No. 6112, Rev. 0, Job Title: U1 - SGRP - RCS Drain Line Removal, Dated 09/10/03
RWP No. 6114, Rev. 0, Job Title: U1 - SGRP - Remove/Install Mechanical/Structural/ Electrical Interferences, Dated 09/10/03
RWP No. 6115, Rev. 0, Job Title: U1 - SG/RHA - Install/Remove HTS, TLD, AUX Crane and RCD, Dated 09/10/03
RWP No. 6116, Rev. 0, Job Title: U1 - SG/RHA - Construction Opening: Tendons/Concrete/ Liner Plate, Dated 09/10/03
RWP No. 6118, Rev. 0, Job Title: U1 - SGRP - Rig/Remove OSG/Transport & Store, Dated 09/10/03
RWP No. 6200, Rev. 0, Job Title: U2 - SG/RHA - Inspections, Surveillance & Firewatch, Dated 03/12/04
RWP No. 6200, Rev. 0, Job Title: U2 - SG/RHA - Radiation Protection Surveillance, Dated 03/12/04
RWP No. 6202, Rev. 0, Job Title: U2 - SGRP - Radiography Testing, Dated 03/12/04
RWP No. 6203, Rev. 0, Job Title: U2 - SG/RHA - Miscellaneous Support Activities Inside Containment, Dated 03/12/04
RWP No. 6204, Rev. 0, Job Title: U2 - SG/RHA - Decontamination Activities, Dated 03/12/04
RWP No. 6205, Rev. 0, Job Title: U2 - SG/RHA - Install and Remove Scaffolding, Dated 03/12/04
RWP No. 6206, Rev. 0, Job Title: U2 - SGRP - Install and Remove Insulation, Dated 03/12/04
RWP No. 6207, Rev. 0, Job Title: U2 - SGRP - RCS Pipe End Decontamination, Dated 03/12/04
RWP No. 6208, Rev. 0, Job Title: U2 - SGRP - Remove/Install RCS Temporary/Permanent Supports, Dated 03/12/04

RWP No. 6209, Rev. 0, Job Title: U2 - SGRP - Remove/Install Large Bore Piping/Secondary Temp/Perm Restraints/Install Seal Plates on MS Pipe End, Dated 03/12/04
RWP No. 6210, Rev. 0, Job Title: U2 - SGRP - RCS Pipe Cutting/Machining/Welding, Dated 03/12/04
RWP No. 6211, Rev. 0, Job Title: U2 - SGRP - Remove/Install Small Bore Piping, Dated 03/12/04
RWP No. 6212, Rev. 0, Job Title: U2 - SGRP - RCS Drain Line Removal, Dated 03/12/04
RWP No. 6213, Rev. 0, Job Title: U2 - SGRP - Nitrogen/Chemical Addition Modification, Dated 03/12/04
RWP No. 6214, Rev. 0, Job Title: U2 - SGRP - Remove/Install Mechanical/Structural/ Electrical Interferences, Dated 03/12/04
RWP No. 6215, Rev. 0, Job Title: U2 - SG/RHA - Install/Remove HTS, TLD, AUX Crane and Reactor Canal Decking, Dated 03/12/04
RWP No. 6216, Rev. 0, Job Title: U2 - SG/RHA - Construction Opening: (Tendon/Concrete/ Liner Plate), Dated 03/12/04
RWP No. 6217, Rev. 0, Job Title: U2 - SG/RHA - Temporary Power, Dated 03/12/04
RWP No. 6218, Rev. 0, Job Title: U2 - SGRP - Rig/Remove Original Steam Generator/ Transport/Store, Dated 03/13/04
RWP No. 6219, Rev. 0, Job Title: U2 - SGRP - Shielding Activities, Dated 03/12/04
RWP No. 6220, Rev. 0, Job Title: U2 - SG/RHA - Support Activities Outside of the Reactor Building, Dated 03/12/04
RWP No. 6221, Rev. 0, Job Title: U2 - SGRP - OTSG Bowl Entry (EHRA), Dated 03/12/04
RWP No. 6275, Rev. 0, Job Title: U2 - SG/RHA - Paint Abatement/Coating Activities, Dated 03/12/04
RWP No. 6276, Rev. 0, Job Title: U2 - RHA - Head Work In Canal, Dated 03/12/04
RWP No. 6277, Rev. 0, Job Title: U2 - RHA - Work on Head Stand, Dated 03/12/04
RWP No. 6278, Rev. 0, Job Title: U2 - RHA - CRDM Removal/Prep/inspect/Transport/Install, Dated 03/12/04
RWP No. 6279, Rev. 0, Job Title: U2 - RHA - Rigging/Lifting/Loading/Transport of RHA/ RRHA, Dated 03/12/04

Records and Data

DPC, ONS, Air Sample Title: U2 - A Cavity Back Up A/S, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - A Cavity Crow Nest Insulation Removal, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - Abate Concrete at A SG, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - B Cavity Back Up Insulation Removal, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - B Cavity Cold Leg Insulation Removal, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - B Cavity Hot Leg Insulation Removal, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - B Cavity Insulation Removal, Dated 03/23/04
DPC, ONS, Air Sample Title: U2 - B Cavity Surge Line Insulation Removal, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - B Cold Leg Insulation Removal, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - B/Hot Leg Insulation Removal, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - B Drain Line Insulation Removal, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - Cold Leg B/Cavity Removal Insulation, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - Containment at Entrance U2 Under Reactor Vessel Undress Area, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - RAB Remove Brick from Under Vessel, Dated 03/24/04
DPC, ONS, Air Sample Title: U2 - Under Vessel, Dated 03/24/04
ONS Alpha Air Sample Counting Results, Sample ID 040119007 for RWP 5094, Sample Title:

U1 - Spent Fuel Pool (Routine)
ONS Alpha Air Sample Counting Results, Sample ID 040119008 for RWP 5094, Sample Title: U1 - U1/2 S.P. DSC# 78 Full Train Sample
ONS Alpha Air Sample Counting Results, Sample ID 040119018 for RWP 5094, Sample Title: U1/2 S.P. Routine DSC# 78
ONS Alpha Air Sample Counting Results, Sample ID 040120001 for RWP 5094, Sample Title: U1 - Spent Fuel Pool (Routine)
ONS Alpha Air Sample Counting Results, Sample ID 040120021 for RWP 5094, Sample Title: U1 - U-1&2 S.P. Routine During DSC# 78
ONS Radiation Protection Pre-Job Briefing; Task Description: Under Vessel Entries for Insulation, Wash, and Inspection; RWP: 2026, Undated
ONS Steam Generator/Reactor Head Assembly Replacement Radiation Protection/Sub Plan, RP-003, Rev. 5, Insulation Removal Plan, Undated
ONS Steam Generator/Reactor Head Assembly Replacement Radiation Protection Sub Plan, RP-009, Rev. 1, Movement of OSG, Undated
ONS Steam Generator/Reactor Head Assembly Replacement Radiation Protection Sub Plan, Rev. 1, RP-013, Movement of RHA/RRHA, Undated
Oconee Steam Generator/Reactor Head Replacement Project, 2EOC20 Waste Management Plan, Rev. 0, Undated
HSM (New Type), Survey No. 012604-1, Dated 01/23/04
HSM (New Type), Survey No. 021004-11, Dated 02/10/04
ISFSI, Survey No. 011104-8, Dated 01/11/04
ISFSI, Survey No. 021304-2, Dated 02/13/04
ISFSI, Survey No. 031104-30, Dated 03/11/04
ISFSI, Survey No. 031704-9, Dated 03/17/04
ISFSI, Survey No. 071301-1, Dated 07/13/01
ISFSI, Survey No. 111101-30, Dated 11/11/01
ONS, Job Dose Card Report, for RWP 2026, U2 Reactor Building Annulus Inspections and Associated Work, Dated 03/25/04
Unit 1 (U1) “A” Steam Generator (SG) Lower Playpen (Grout Abatement), Survey No. 032404-3, Dated 03/24/04
U2 3rd Floor Containment, Survey No. 032004-41, Dated 03/20/04
U2 “A” Cavity 5th Grain Level, Survey No. 032104-9, Dated 03/21/04
U2 “A” Cavity Cut Away View, Survey No. 032404-30, Dated 03/24/04
“A” SG (Shown in Cavity), Survey No. 032204-28, Dated 03/22/04
Lower SG Bowl Survey, Survey No. 032404-2, Dated 03/23/04
RP Survey Points at ISFSI, Undated
RP & Site Safety Status Report 2EOC-20, Dated 03/23/04
RP & Site Safety Status Report 2EOC-20, Dated 03/25/04
U1 EOC21 Steam Generator Reactor Head Replacement Outage, 1EOC21 RP/ALARA Outage Report, Undated
U2, EOC-20 Refueling Outage Updates, Dated 03/23/04 and 03/25/04

<u>CAP Documents</u>

PIP No. O-03-02701, Dose Rates at U3 BWST Increased Due to Draining to Fill Canal Deep End
PIP No. O-03-04458, Post Job Radiation Levels on Resin Piping was Higher than Pre-Job Levels
PIP No. O-03-04707, 3EOC20 ALARA Planned Task Exceeded Estimated Exposure by >25%

PIP No. O-03-06136, 2A Letdown Filter Radiation Levels Increased from 3,600 Millirem per Hour (mrem/hr) to 26,400 mrem/hr After Performing HPI Pump Testing
PIP No. O-03-06767, Change in Radiological Conditions Due to a Change in the Work Evolution
PIP No. O-03-06914, Piping was Found to be Reading 3,500 mrem/hr on Contact
PIP No. O-03-07330, RWP 1011 U-1 Reactor Building In Service Inspections (ISI) Total Dose has Exceeded the Dose Estimate by >25%
PIP No. O-03-07347, ALARA Package/RWP 6111 Actual Total Exposure Exceeds 125% of the Dose Estimate
PIP No. O-03-07352, ALARA Package/RWP 6175 Total Actual Dose Exceeds 125% of the Original Estimate
PIP No. O-03-07524, RWP 6103 Exceeds 125% of the Estimated Dose Goal
PIP No. O-03-07562, RWP/ALARA Package 6111 Exceeded 125% of the Dose Goal
PIP No. O-03-07617, RWP/ALARA Package 6177 Exceeds 125% of the Dose Estimate
PIP No. O-03-07794, Increase in General Dose Rates Due to Removal of Temporary Shielding in the Reactor Building
PIP No. O-04-00444, RWP 5094 Exceeded Estimated Exposure by >25%
PIP No. O-04-00700, RWP 5095 Exceeded Exposure Estimate by >25%
PIP No. O-04-01552, 2 Workers Exceeded Their Dose Alarm Set Points
PIP No. O-04-01556, Crud Burst Activities and Increased Dose Rates

(Section 4OA5.3: Other Activities)

Procedures And Specifications

Specification No. OSS-0279.00-00-005, Rev. 1, Reactor Vessel Head Replacement Installation Procurement Specification
Specification No. OSS-0279.00-00-0003, Rev. 2, Certified Design Specification for Replacement Reactor Vessel Closure Head Assemblies
Specification No. TS-2790, Rev. 8, Design Interfaces for Reactor Pressure Vessel Heads & Service Structure at Oconee Nuclear Station Units 1, 2, & 3
Final Scope Document NSM-ON-23112-000, Rev. 0, Reactor Vessel Head Replacement Final Scope Document
Procedure No. QEP 10.05, Rev. 1, Rigging and Handling

Other Documents

Modification No. 23112 Part No. AM2, Rev. 2, Unit 2 Reactor Head Replacement
Modification No. 23112 Part No. AM3, Rev.0E1, Unit 2 Top Head Piping Modification
Modification No. 23112 Part No. AS1, Rev. 0, Reactor Vessel Head Rigging and Handling
Document No. 259236, Rev. 1, Shop Instruction for Testing of Lift Pendant Assembly
Babcock & Wilcox Canada Inspection and Test Report No. B/S 5708, Test Pendant Assemblies for Unit 2
NRC letter to Oconee on March 20, 2002, Subject: Request to Use Alternative Materials Per Code Cases for the Fabrication of Replacement Reactor Vessel Closure Heads at Oconee Nuclear Station Units 1, 2, and 3
Receiving Inspection Report for Reactor Vessel Closure Head & Service Structure Assemblies (RVCHA) in Accordance With OSS-0279.00-00-003
Drawing No. 083DC144, Rev. D05A, SS Ventilation Fan Mounting Assemblies
Drawing No. 083DE001, Rev. 03, RPVH Service Structure General Arrangement

Drawing No. 083DE106, Rev. D03A, SS Monorail Installation
Drawing No. 083DE107, Rev. D03A, SS Monorail Component Assemblies
Drawing No. 083DE165, Rev. D03A, Service Structure to Platform Assemblies
OM 201.R-0141.001, Rev.D03, Input Document for Replacement RVCHA Licensing and Safety Evaluation
BWC-CONT 068S-02, History Docket for Closure Head
BWC-OONT 083-02, History Docket for Service Structure Components
Calculation No. OSC-8411, Replacement Reactor Vessel Closure Head, Design and Service Loading Condition Report

LIST OF ACRONYMS

ADAMS	-	Agency wide Documents Access and Management System
AFIS	-	Automatic Feedwater Isolation System
ALARA	-	As Low As Reasonably Achievable
AS	-	Auxiliary Steam
ASME	-	American Society of Mechanical Engineers
ASW	-	Auxiliary Service Water
BWST	-	Borated Water Storage Tank
CCW	-	Condenser Circulating Water
CEDE	-	Committed Effective Dose Equivalent
CFR	-	Code of Federal Regulations
CRDM	-	Control Rod Drive Mechanism
DEC	-	Duke Energy Corporation
EFW	-	Emergency Feedwater
EHRA	-	Extra High Radiation Area
ERO	-	Emergency Response Organization
FME	-	Foreign Material Exclusion
FSAR	-	Final Safety Analysis Report
GPM	-	Gallons per Minute
HP	-	Health Physics
ICS	-	Integrated Control
IP	-	Inspection Procedure
IR	-	Inspection Report
ISFSI	-	Independent Spent Fuel Storage Installation
ISI	-	Inservice Inspection
KHU	-	Keowee Hydro Unit
LCO	-	Limiting Condition for Operation
LER	-	Licensee Event Report
LOCA	-	Loss of Coolant Accident
LPI	-	Low Pressure Injection
LPSW	-	Low Pressure Service Water
MDEFW	-	Motor Driven Emergency Feedwater
NCV	-	Non-Cited Violation
NOUE	-	Notification of Unusual Event
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
OAC	-	Operator Aid Computer
ONS	-	Oconee Nuclear Station
OTSG	-	Once-Through Steam Generator

PIP	-	Problem Investigation Process Report
PM	-	Preventive Maintenance
PMT	-	Post-Maintenance Testing
PSIG	-	Pounds per Square Inch Gauge
RA	-	Radiation Area
RB	-	Reactor Building
RBCU	-	Reactor Building Cooling Unit
RBES	-	Reactor Building Emergency Sump
RCA	-	Radiation Controlled Area
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RFO	-	Refueling Outage
RTP	-	Rated Thermal Power
RVHR	-	Reactor Vessel Head Replacement
RVLIS	-	Reactor Vessel Level Indication System
SDP	-	Significance Determination Process
SGR	-	Steam Generator Replacement
SSC	-	Structure, System and Component
SSF	-	Standby Shutdown Facility
TDEFW	-	Turbine Driven Emergency Feedwater
TS	-	Technical Specification
U1	-	Unit 1
U2	-	Unit 2
U3	-	Unit 3
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
VHRA	-	Very High Radiation Area
WO	-	Work Order