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

January 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 05000269/2003005, 05000270/2003005, AND 05000287/2003005 AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION INSPECTION REPORT 72-04/2003001

Dear Mr. Jones:

On December 27, 2003, 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 January 12, 2004, 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.

This report documents three NRC-identified findings and three self-revealing findings of very low safety significance (Green). Five 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 five findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. Additionally, five licensee-identified violations which were determined to be of very low safety significance (Green) are listed in Section 4OA7 of this report. 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 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure and your response (if any) 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, 72-04
License Nos.: DPR-38, DPR-47, DPR-55

Enclosure: NRC Integrated Inspection Report 05000269/2003005, 05000270/2003005, 05000287/2003005, and 72-04/2003001 w/Attachment

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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:	50-269/2003005, 50-270/2003005, 50-287/2003005 72-04/2003001
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	September 28, 2003 - December 27, 2003
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Approved by:	R. Haag, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000269/2003-005, IR 05000270/2003-005, IR 05000287/2003-005, 72-04/2003-001; 09/28/2003 - 12/27/2003; Oconee Nuclear Station; Flood Protection, Radioactive Material Processing and Transportation, Other Activities

The report covered a three-month period of inspection by the resident inspectors and announced regional-based inspections by: a senior project manager, a project engineer, three senior reactor inspectors, two reactor inspectors, a senior health physicist, a health physicist, and a radiation specialist. Five 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. NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, was identified by the inspectors for failure to follow instructions in that a flood protection barrier was found improperly removed.

  The finding was considered to be more than minor because the missing flood barrier affected the mitigating systems cornerstone in that safety related equipment was no longer protected from external factors such as flooding. The Phase 1 screening concluded, that for accident scenarios involving breaks of smaller non-seismic piping in the auxiliary building, the low pressure injection safety function could be adversely affected. Auxiliary building flooding has been previously analyzed in a Phase 3 analysis. This analysis concluded that performance deficiencies related to mitigation of small piping breaks, such as those for which the flood protection barrier was intended to mitigate, would result in a "Green" finding because they would not affect the component cooling system (i.e., Reactor Coolant Pump seal cooling.) (Section 1R06.1)

- Green. The inspectors identified a finding for failure to implement the Standby Shutdown Facility (SSF) diesel generator manufacturer's recommended preventive maintenance schedule for replacement of grommets every six years. Consequently, at ten years some of the grommets were found to be "at or near failure."

  This finding is more than minor because a failure of the grommets could lead to diesel coolant leaks and loss of cooling to the diesel; thereby, affecting the reactor safety mitigating system cornerstone objective to ensure the availability, reliability, and capability of a system that responds to initiating events to prevent core damage. A Phase III evaluation, which credits the replenishment of SSF diesel generator cooling and recovery of offsite power, indicated that the performance deficiency was of very low safety significance. (Section 4OA5.4)

- Green. A NCV of 10 CFR 50 Appendix B, Criterion III, was identified by the inspectors for failure to properly translate design basis parameters for emergency core cooling systems (ECCS) into applicable specifications, drawings, procedures, and instructions. Specifically, design calculation OSC-6667 documented that post LOCA temperatures in the low pressure injection (LPI) and high pressure injection (HPI) pump rooms could reach ambient temperatures as high as 257 degrees; however, safety-related pumps and motors in those rooms (i.e., LPI, HPI, and reactor building spray pumps and motors) were not environmentally qualified for this type of environment.

  The finding was considered to be more than minor because it potentially affected the mitigating systems cornerstone, in that it affected the environmental qualification of safety-related equipment needed to mitigate a loss of coolant accident. The finding was determined to be of very low safety significance (Green) due to the fact that the re-calculated ambient temperatures were lower than 257 degrees and that actual testing indicated that the pumps and motors could operate successfully at the predicted ambient temperatures without adverse consequences. Therefore, there was no loss of function, and the issue was screened out in Phase 1 of the SDP as Green. (Section 4OA5.5)

Cornerstone: Barrier Integrity

- Green. A self-revealing NCV of 10 CFR 50 Appendix B, Criterion V, was identified for an inadequate maintenance procedure for inspection of the foreign material exclusion (FME) barrier in the 1B hot leg during steam generator replacement. This allowed the introduction of sealant material in the reactor coolant system (RCS) piping.

  The finding was considered to be more than minor because it potentially affected the barrier integrity cornerstone, as foreign material, if left uncorrected, could have an adverse impact on fuel cladding integrity during operation. Additionally, inadequate inspection activities, if not corrected could have adverse consequences on future activities affecting quality. The finding was determined to be of very low safety significance (Green) due to the fact that the foreign material was successfully removed and that the SDP phase one screening for findings that potentially affected the fuel barrier screen as Green. (Section 4OA5.3)

Cornerstone: Public Radiation Safety

- Green. A self-revealing NCV of 10 CFR 61.56(b)(2) was identified because the licensee transported a cask shipment for disposal at Chem-Nuclear Systems, Barnwell, South Carolina which contained liquid above regulatory limits.

  This finding is greater than minor because it was associated with the low level burial attribute of the Public Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain. The finding is of very low safety significance because the shipping cask was discovered to have minimal liquid exceeding the regulatory limit of one percent of the waste shipment total volume transported to the burial site for disposal and the liquid was discovered prior to waste disposal. (Section 2PS2b.(1))

- Green. A self-revealing NCV of 10 CFR 61.55(a)(2)(ii) was identified because the licensee transported a cask shipment for disposal at Chem-Nuclear Systems, Barnwell, South Carolina with the incorrect waste classification. The cask was originally shipped to Chem-Nuclear Systems, Barnwell, South Carolina, as Class A stable waste and later determined by the licensee to be Class B stable waste.

  This finding is more than minor because it was associated with the low level burial attribute of the Public Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain. The finding is of very low safety significance because the shipping container was discovered by the licensee to have been under-classified prior to its final disposal and the burial site representatives were properly notified of the classification error. (Section 2PS2b.(2))

B. Licensee Identified Violations

Five violations of very low safety significance, which was 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 entered the report period in a refueling outage. Following the outage, the unit entered Mode 1 on December 16, 2003, and reached 20 percent rated thermal power (RTP) on December 16, 2003. On December 17, 2003, the unit was shutdown to repair a RCP seal. The unit was shutdown for the remainder of the inspection period.

Unit 2 operated at or near 100 percent RTP for the inspection period except for one power reduction. The unit was reduced to approximately 88 percent RTP on October 5, 2003 to perform turbine valve movement testing. The unit was returned to 100 percent RTP on the same day.

Unit 3 operated at or near 100 percent RTP for the inspection period

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather

Cold Weather Preparations

a. Inspection Scope

The inspectors reviewed the licensee's preparations for the onset of seasonal cold weather. Specifically, the inspectors reviewed the completed maintenance work orders for checks of freeze protection circuits for the Unit 1, 2, and 3 borated water storage tanks and the essential siphon vacuum system. The inspectors ensured that the freeze protection circuit checks were performed before any significant cold weather impacted the plant. The inspectors reviewed the data from IP/0/B/1606/009, Preventive Maintenance and Operational Check of Freeze Protection, and IP/0/B/1606/009A, Preventive Maintenance and Operational Check of QA-1 Freeze Protection, to verify the applicable circuits met acceptance criteria. The inspectors discussed instances where acceptance criteria were not met with the maintenance supervisor and the corrective actions (work request 98295319) were reviewed.

In addition, previously during the summer months the inspectors reviewed the licensee's preparations for controlling emergency core cooling system room temperatures and monitored the rooms periodically to verify that temperatures remained below the design basis temperature limits. The low pressure injection (LPI) and high pressure injection (HPI) pump rooms, for all three units, were routinely monitored. This review was performed in conjunction with the normal plant tour and was not documented in the inspection report which covered that time period. However, the inspection of these two systems satisfied the objectives of the adverse weather inspection procedure (IP) and documenting the inspection in this report provides the historical record for completion of these inspections.

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 systems were included in this review:

- Unit 2 LPI and Reactor Building Spray (RBS) Trains during maintenance on 2LP-22
- Unit 3 LPI train B while the A train was out of service for scheduled preventive maintenance
- Unit 3 Emergency Feedwater (EFW) System while the Turbine Driven Emergency Feedwater (TDEFW) Pump was out of service for in-service testing

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and condition of the Unit 1 RCP seal injection system. The inspectors utilized the licensee's system alignment procedure and system drawings to verify proper system alignment. The review also included a verification that associated system instruments and support system instruments were properly calibrated per procedures IP/O/B/0240/001, Component Cooling Pressure, Flow and Control Rod Drive Filter D/P Instrument Calibration and IP/1/B/0202/001P, High Pressure Injection System RCP Header Flow Control Instrument Calibration. System valve testing was also reviewed to ensure proper operation per procedure PT/1/A/0152/011, High Pressure Injection System Valve Stroke Test.

The inspectors also verified electrical power alignment, labeling, and hanger and support installations. The operating HPI pump was monitored by computer data points to ensure that vibration was not excessive, bearing temperatures were normal, and the pump room was adequately ventilated. The walkdown also included an evaluation of the HPI and Standby Shutdown Facility (SSF) Makeup Pump systems and system piping and supports against the following considerations:

- Piping and supports did not show evidence of water hammer
- Hangers were properly sized and were within the setpoints
- The piping system did not show evidence of prior system leakage
- Component foundations were not degraded

A review of problem investigation Process reports (PIPs) and maintenance work orders was performed to verify that material condition deficiencies did not significantly affect the ability of the seal injection system to perform its design functions and that appropriate corrective action was being taken by the licensee. This review included PIP O-03-06716, which documented particulate damage to the RCP seals. Because of the apparent problem with FME control in the seal injection system, the inspectors also observed the subsequent seal injection line flushing accomplished per procedure IP/1/A/1104/002D, Seal Injection Line Flush.

The inspectors also held discussion with the system engineers regarding ongoing modifications to the RCP seal packages to ensure that the impact on the equipment functionality was properly evaluated and previous seal leakage problems were being adequately addressed.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours in eleven 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 1 Equipment Room (1)
- Keowee Hydro Units (2)
- Standby Shutdown Facility (1)
- Unit 1, 2 and 3 Turbine Building Ground Floor (3)

- Unit 1, 2 and 3 Control Battery Rooms (3)
- Radwaste Facility (1)

b. Findings

No findings of significance were identified.

1R06 Flood Protection

.1 Auxiliary Building Internal Flood Protection

a. Inspection Scope

The inspectors reviewed the mitigation barriers and abnormal procedures related to auxiliary building internal flooding.

b. Findings

Introduction: A Green NCV was identified by the inspectors for failure to follow instructions that resulted in an internal flood protection barrier not being reinstalled.

Description: On September 30, 2003, during a routine plant tour, the inspectors noted that the internal flood protection barrier at the Unit 2 access to the LPI and reactor building spray (RBS) pump rooms had been removed. The barrier had instructions stating, “CAUTION” Auxiliary Building Flood Related-Remove only for equipment passage. The inspector noted that no equipment or activity was in effect that would require the barrier to be removed. The control room was notified and the barrier was subsequently reinstalled.

Analysis: The finding was considered to be more than minor because the missing flood barrier affected the mitigating systems cornerstone in that safety related equipment was no longer protected from factors such as flooding. The Phase 1 screening concluded, that for accident scenarios involving breaks of smaller non-seismic piping in the auxiliary building, the LPI safety function could be adversely affected. Auxiliary building flooding has been previously analyzed in a Phase 3 analysis. This analysis concluded that performance deficiencies related to mitigation of small piping breaks, such as those for which the flood protection barrier was intended to mitigate, would result in a “Green” finding because they would not affect the component cooling system, i.e., RCP seal cooling. Therefore, based on previous risk analysis for flooding, the inspectors concluded that the issue of not maintaining proper flood protection for the LPI and RBS pump rooms to be of very low safety significance

Enforcement: 10 CFR 50, Appendix B, Criterion V, requires that activities affecting quality shall be prescribed by documented instructions, procedures or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures and drawings. Instructions on the internal flood protection barrier at the Unit 2 access to the LPI and RBS pump rooms state, “CAUTION” Auxiliary Building Flood Related-Remove only for equipment passage.

Contrary to the above, on September 30, 2003, activities affecting quality were not accomplished in accordance with documented instructions in that the flood protection barrier for Unit 2 access to the LPI and RBS pump rooms was found to be removed and no activity was in effect that would have required the gate to be removed. Because this finding is of very low safety significance and has been entered into the corrective action program as PIP O-03-06292, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. It will be identified as NCV 05000269,270/2003005-01: Failure to Maintain Flood Protection Barriers.

.2 Turbine Building Internal Flood Protection

a. Inspection Scope

The inspectors reviewed the licensee's flood protection engineering support document, vendor information, and held discussions with the flood protection engineer to determine if there were any flood related vulnerabilities to the solenoid valves incorporated with the air operated valves used to isolate condenser cooling water in the turbine building in the event of a turbine building flood event. The inspectors also inspected the installed valves to evaluate the material condition of the valves and electrical connections to verify they were not susceptible to water intrusion that could potentially affect the valves' reliability.

b Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

The inspectors observed Unit 1, Cycle 21 in-process ISI work activities during the 3^{rd} interval, 3^{rd} ISI period, and final outage, and reviewed selected ISI records. The observations and records were reviewed for compliance to the Technical Specifications (TS) and the applicable Code (ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no addenda). The following Unit 1 ISI examinations were observed/reviewed:

Penetrant (PT)	B09.021.060	1RC-200-161	Pipe to Safe End
		1RC-289-7V	Pipe Weld
		1AG1-R1	HL Pipe Weld
Radiographic (RT)		1LP-209-31	Pipe Weld
		1LP-209-32	Pipe Weld
		1FW-172-32V-R1	Pipe Weld
		1MS-069-35V	Pipe Weld
		1RC-289-1V	Pipe Weld
		1MS-076-1V	Pipe Weld
		1RC-289-7V	Pipe Weld
		1RC-289-4V	Pipe Weld

Ultrasonic (UT)		1-LP-209-32	Pipe Weld
	G02.001.005C	1-PDB1-46	HPI Nozzle to Safe End
	G02.001.005D	1-PDB2-46	HPI Nozzle to Safe End
	B09.011.115	1-PSL-3	Pressurizer Surge Line
	B09.011.116	1-PSL-4	Pressurizer Surge Line
Magnetic Particle (MT)		1RC-289-7V	Pipe Weld
		1RC-289-3V	Pipe Weld
		1RC-289-1V	Pipe Weld
		1RC-289-6V	Pipe Weld
		1B2LS1	HL Longitudinal Seam Weld

Qualification and certification records for examiners, equipment and consumables, and nondestructive examination (NDE) procedures for the above ISI examination activities were reviewed. In addition, a sample of ISI issues in the licensee's corrective action program were reviewed for adequacy.

The inspectors reviewed four NDE techniques including both volumetric and surface examinations, two reportable indications 1-03A-SK49 & 1-03A-SR85 from the last outage 1ECO 20 and one ASME Class 2 pressure boundary preservice examination for weld 1-LP-0209-31.

The inspectors also examined interior portions of the Unit 1 containment building and reviewed selected records. The observations and records were compared to the TS, ASME Boiler and Pressure Vessel Code, Article IWE of Section XI, 1992 Edition and 1992 Addenda, and 10 CFR 50.55a. The inspectors examined the interior surfaces of the containment liner and the moisture barrier at the intersection of the liner and interior concrete floor area.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator simulator training on December 3, 2003. The scenario involved performance of the infrequent operating procedure for a gaseous waste release, followed by an RCS leak outside containment and concluded with a steam generator tube rupture. 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 Specifications (TS) actions.

Documents reviewed within this inspection area are listed in the Attachment to this report.

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:

- PIP O-03-06716, Particulate Damage to the RCP seals found during outage inspection. Based on discussions with the licensee, the particulate had been left in the seal injection system during previous maintenance activities. The licensee's final root cause was still pending at the completion of this inspection period.

- PIP O-03-6344, Emergency Power source rendered inoperable due to Voltage Regulator Failure on Keowee Hydro Unit (KHU) KHU-1

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.

- 3B RBCU Fan Failure, PIP O-03-6836

- Maintenance on ACB-3 with CT-5 out of service and KHU-2 aligned to the underground power path, PIP O-03-7440

- 3A Once-Through Steam Generator (OTSG) EFW Train A Flow Indication, PIP O-03-7514

- KHU Swap with 2A Motor Driven Emergency Feedwater Pump out of service

- Unit 2 Orange risk condition (cycle 2LP-21 for Reactor Building Spray Maintenance)

- KHU -Lee Orange risk condition

- Unit 2 AFIS/RPS with Primary Instrument Air out of service

- Unit 1 outage Yellow risk condition (Decay Heat and Containment)

- Unit 1 Orange risk condition, Complex Plan for equipment lifts near main steam lines in Mode 3 and greater

- 1A1 RCP Upper Seal Failure, PIP O-03-8163

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 evolution reviewed during this inspection period included the following:

- Unanticipated transfer of power from the Main Transformer to the Startup Transformer during electro-hydraulic turbine control testing, PIP O-03-7137

- 3B Reactor Building Cooling Unit (RBCU) Fan Failure, PIP O-03-6836

- 1A1 RCP Upper Seal Failure, PIP O-03-8163

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. The inspectors reviewed the following items for operability evaluations:

- PIP O-03-6497, 1LP-36 failed as found set pressure
- PIP O-03-7425, 3B RBCU Damper Failure
- PIP O-03-7510, 3RC-50 packing and seat leaks
- PIP O-03-7511, 3CB Battery cell #49 leaking
- PIP O-03-7521, 3FDW-547 pipe cap leak
- PIP O-03-7637, SSF RCMU Pump Trip
- PIP O-03-7856, Water inside tendon cap for tendon 62H21

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

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.

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:

- OP/1/A/1104/002, Seal Injection Line Flush, performed to ensure the seal injection lines did not contain any further particulate that could damage the RCP seals.

- PT/1/A/0150/003, Reactor Building Integrated Leak Rate Test, following repair of SG replacement construction opening in containment side wall

- PT/1/A/0600/012, Unit 1 TDEFW Pump Test, following repair of the rotating element and maintenance on 1MS-93

- PT/1/A/0400/007, Unit 1 SSF RC Makeup Pump Test, following multiple pump trips (PIP O-03-7637)

- MP/0/A/1720/016, System/Component Pressure Test Controlling Procedure, following OTSG replacement

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors conducted reviews and observations for selected outage activities to ensure that: (1) the licensee considered risk in developing the outage plan; (2) the licensee adhered to the outage plan to control plant configuration based on risk; (3) that mitigation strategies were in place for losses of key safety functions; and (4) the licensee adhered to operating license and TS requirements. Between September 28, 2003, and December 27, 2003, the following activities related to the Unit 1 refueling outage were reviewed for conformance to applicable procedures and selected activities associated with each evaluation were witnessed:

- Activities involving the hanging of tags and subsequent removal and system restoration
- Installation, calibration and configuration of reactor coolant system instrumentation
- Electrical system lineups
- Defueled (no Mode) operations
- Refueling operations, including inventory and reactivity controls in both the reactor vessel and the spent fuel pool
- Reduced inventory and mid-loop conditions for installation and removal of steam generator nozzle dams
- Activities involving the reactor vessel head replacement
- Reactor heatup and startup
- Mode changes from Mode 6 to Mode 1, (Refueling to Power Operation)
- System lineups during major outage activities and Mode changes
- Final containment walkdown prior to startup

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/3/A/0230/015, High Pressure Injection Motor Cooler Flow Test
- IP/1/A/0305/001J, K and L, RPS Channel B, C, and D RC Flow Instrument Calibration
- PT/2/A/0261/010, Essential Siphon Vacuum System Test

- IP/2&3/B/0202/001C, RCP Seal Flow Inst Calibration
- CP/2/A/2002/001, Unit 2 RCS Sampling System
- PT/1/A/0151/019, Penetration 19 Leak Rate Test
- PT/3/A/0202/011, High Pressure Injection Pump Test.

b. Findings

No findings of significance were identified.

1R23 Temporary 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 items were reviewed under this inspection procedure:

- ONTM-2155, Install Temporary Jumper on 1HP-16 Limit Switch
- ONTM-2161, Temperature Compensate Unit 3 Pressurizer Level Channel 1 with Pressurizer Temperature Channel C
- ONTM-2162, Pressure Compensate Unit 3 Pressurizer Level Channels 1 and 2

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

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

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Controls - The inspectors evaluated licensee activities for monitoring and controlling worker access to radiologically significant areas during the Unit 1 (U1) steam generator replacement, reactor vessel head replacement, and refueling outage (RFO) . The inspection included direct observation of administrative and physical controls, appraisal of the knowledge and proficiency of radiation workers and health physics technicians (HPTs) in implementing radiological controls, and review of the adequacy of procedural guidance and its implementation.

The inspectors observed implementation of radiological controls for selected Radiation Area (RA), Radioactive Material Area, and High Radiation Area (HRA) locations within Radiologically Controlled Areas (RCAs). Posting and labeling of materials at these locations were evaluated for consistency with procedural guidance and compliance with regulations. The inspectors directly observed the posting and locking status of the only Very High Radiation Area (VHRA) in the U1 Reactor Building (RB), selected HRAs and Extra High Radiation Areas (EHRAs) in the U1 Auxiliary Building (AB), Radwaste Building, and Interim Radwaste Building. Independent dose-rate measurements were taken in the U1 Reactor Building, Radwaste Building, and Interim Radwaste 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 Protected Area but within the Owner-Controlled Area. The inspectors evaluated the use of radiological controls, observed the performance of HPTs and radiation workers, evaluated Radiation Work Permit (RWP) requirements and electronic dosimeter alarm setpoints, and discussed various task evolutions with selected personnel. During general observations of outage work, the inspectors queried radiation workers on RWP requirements associated with their tasks in progress.

The inspectors reviewed administrative guidance documents and procedures for control of material stored in spent fuel pools, posting of areas, access controls to EHRAs, surveys of areas, and RWP use. The inspectors reviewed selected RWPs and surveys of such areas to evaluate the adequacy of radiological controls for RAs, HRAs, and airborne areas. Records of internal dose assessments performed during the past year were reviewed and discussed with management. Health Physics supervisory personnel were interviewed regarding administrative control of EHRA and VHRA keys, as well as any changes to procedural guidance for access control.

Procedural guidance regarding the use of supplied-air bubble suits was reviewed and discussed. Specifically, training and guidance on the availability of standby rescue personnel during use of such suits were reviewed and discussed with licensee representatives.

Radiation Protection (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 commitments and approved procedures. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment to this report.

<u>Problem Identification and Resolution</u> - 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 responsible licensee representatives. The inspectors assessed the licensee’s ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure Nuclear System Directive 208, Problem Investigation Process. Specific assessments, audits, and PIP documents reviewed and evaluated in detail for this inspection area are identified in the Attachment to this report.

Independent Spent Fuel Storage Installation (ISFSI) - Access control and surveillance results for the licensee's ISFSI were evaluated. The evaluation included review of ISFSI radiation control surveillance procedures and assessment of radiological survey data. The inspectors toured the ISFSI and observed access controls, thermoluminescent dosimeter placement, and radiological postings on the perimeter security fence. The inspectors observed a licensee technician perform gamma and neutron radiation surveys of a spent fuel cask. Surveys made at locations procedurally designated for routine surveys within the perimeter fence were also observed. Survey results were compared to the most recent survey records.

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 Technical Specification (TS), 10 CFR Parts 20 and 72, and applicable licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed the Attachment to this report.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

ALARA Planning and Controls - The inspectors evaluated the licensee's As Low As Reasonably Achievable (ALARA) program guidance and its implementation for ongoing job tasks during the Unit 1 refueling/steam generator replacement/reactor head replacement outage. 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 dismantling of the old steam generators, insulation removal/replacement and scaffold installation/removal, and nondestructive examination inspections. 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, the reactor head upper internals lift, and steam generator 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 and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel.

Implementation and effectiveness of selected program initiatives with respect to source-term reduction were evaluated. Shutdown chemistry program implementation and the resultant effect on containment and auxiliary building dose rate trending data were 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.

The plant collective exposure history for the years 2000 through 2002, based on the data reported to the NRC pursuant to 10 CFR 20.2206(c), was reviewed and discussed with licensee staff, as were established goals for reducing collective exposure. The inspectors examined the dose records of three declared pregnant workers during 2003 to evaluate current gestation dose. The applicable RP procedure was reviewed to assess licensee controls for declared pregnant workers.

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, and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Section 2OS2 of the report Attachment.

Problem Identification and Resolution - 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 licensee procedures. Documents reviewed are listed in Section 2OS2 of the report Attachment.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization - During the week of November 3, 2003, the operability and configuration of selected liquid and solid radioactive waste (radwaste) processing systems and equipment were evaluated. Inspection activities included document review, interviews with plant personnel, and direct inspection of processing equipment and piping.

The inspectors directly observed radwaste processing equipment material condition and configuration for liquid and solid radwaste systems during plant tours with the Radwaste Supervisor. Liquid radwaste equipment was inspected for general condition and licensee staff were interviewed regarding equipment function and operability. The following components of the liquid radwaste system were inspected:

- Waste Hold-up Tanks, Auxiliary Building
- Waste Hold-up Tanks, Radwaste Building
- Waste Monitor Tanks
- Volume Reduction System

The piping and system components were inspected for material condition and for configuration compliance with the UFSAR. The radwaste supervisor was interviewed to assess knowledge of resin sluicing processes and solid radwaste operations. Procedural guidance involving transfer of resin and filling of waste packages was reviewed for consistency with the licensee's procedures and Chapter 11 of the UFSAR for system requirements. Documents reviewed are listed in the Attachment to this report.

Licensee radionuclide characterizations of each major waste stream were evaluated. For dry active waste (DAW), primary resin, secondary resin, and filters, the inspectors evaluated the licensee procedural guidance against 10 CFR 61.55 and the Branch Technical Position on Radioactive Waste Classification details. Comparison data between the licensee's waste sample gamma-emitter concentrations and those of a vendor laboratory were evaluated for the years 2001 - 2003. The licensee's analysis for, and the use of scaling factors for hard-to-detect nuclides were assessed. DAW stream radionuclide data were reviewed and discussed with the licensee for the period 2001-2003, to determine if known plant changes had an effect on radionuclide composition and were assessed by the licensee. The inspectors also reviewed waste shipment quantities for processing and burial for the years 2001-2003.

<u>Transportation</u> - The inspectors evaluated the licensee's activities related to the transportation of radioactive material. The evaluation included a review of shipping records and procedures and assessment of worker training and proficiency. The inspectors reviewed shipping-related procedures for compliance to applicable regulatory requirements. Also, selected shipping records were reviewed for consistency with licensee procedures and for completeness and accuracy. Training records for three individuals qualified to ship radioactive material were checked for completeness.

Transportation program guidance and implementation were reviewed against regulations detailed in 10 CFR 71, 49 CFR 170-189, UFSAR Chapter 11, and applicable licensee procedures listed in Section 2PS2 of the report Attachment. In addition, training activities were assessed against 49 CFR 172 Subpart H, and the guidance documented in NRC Bulletin 79.

<u>Problem Identification and Resolution</u> - Issues identified through licensee self-assessments and PIP documents, were reviewed and discussed with cognizant licensee representatives. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the issues identified in this RP program area in accordance with licensee procedure Nuclear System Directive 208, Problem Investigation Process.

In particular, the inspectors reviewed in detail the circumstances surrounding the licensee's actions associated with a January 8, 2003, cask shipment of resin to the Chem-Nuclear Systems, Barnwell, South Carolina low level waste disposal facility

(Shipment No. O103-11647) which, upon receipt, was determined to have liquid exceeding the one percent of the volume allowed by 10 CFR 61.56(b)(2).

The inspectors also reviewed the under-classification of this same shipment, Shipment No. O103-11647, which originally was reported to Chem-Nuclear Systems as a Class A Stable shipment that the licensee later determined should have been classified as a Class B shipment as required by 10 CFR61.55(2)(ii). For this shipment, the inspectors reviewed applicable shipping procedures, waste classification procedures, shipping records, shipment preparation and receipt surveys, and shipping personnel training records. The inspectors reviewed PIPs O-03-00624 and O-03-00224, which were generated by the licensee for the event.

In addition, the inspectors interviewed involved shipping and receiving health physics personnel and line supervisors, and assessed their knowledge of transportation requirements and procedural implementation relative to Shipment No. O103-11647. The inspectors also reviewed health physics correspondence related to the event, and assessed licensee actions with respect to NRC reportability.

The specific records, procedures, and documentation reviewed with respect to the licensee's problem identification and resolution program and Shipment No. O103-11647 are identified in the Attachment to this report.

b. Findings

(1) Introduction: A Green, self-revealing NCV was identified for the failure to comply with 10 CFR 61.56(b)(2), in that the licensee transported a shipment of waste (resin) for disposal to Chem-Nuclear Systems, Barnwell, South Carolina, which had liquid in the waste cask exceeding one percent of the volume.

Description: On January 08, 2003, the licensee offered a waste shipment in a 14-215H-2 cask, High Integrity Container (HIC) L501039-98, Shipment Identification No. 0103-11647 (Oconee Shipment Manifest No. RSR ONS03-2001) for disposal to Chem-Nuclear Systems, Barnwell, South Carolina. After receipt of the waste cask shipment for disposal, on February 04, 2003, Chem-Nuclear punctured the bottom of the cask and allowed the cask to drain for a period of time to examine the volume of free standing liquid in the cask. Chem-Nuclear determined the licensee had offered a cask shipment for disposal which contained 6.33 gallons of liquid, which exceeded one percent of 6.32 gallons by 1.28 ounces. Based on the sample results, Chem-Nuclear notified Oconee Nuclear plant of the liquid quantity exceeding the regulatory limit of one percent.

The inspectors discussed with licensee personnel the process control procedure used for draining a cask containing resin waste prior to shipment. The inspectors determined the licensee followed its process control procedure by conducting three separate eight hour cask pumping evolutions prior to performing the final sample for liquid. The licensee had determined the liquid in the cask to be less than one-half gallon. The licensee has conducted a root cause evaluation of this event and initiated interim corrective actions.

Analysis: The inspectors determined that the failure to comply with 10 CFR 61.56(b)(2) requirements for low level burial disposal is a performance deficiency, because the licensee is expected to meet the requirements of 10 CFR 61.56(b)(2). The inspectors determined that the finding was associated with the low level burial attribute of the Public Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain. Therefore, the finding is greater than minor. This finding was evaluated using the Public Radiation Safety Significance Determination Process and was determined to be of very low safety significance (Green) because the excess liquid was removed at the burial site prior to final disposal.

Enforcement: 10 CFR 61.56(b)(2) specifies minimum requirements for solid waste and requires that solid waste containing liquid to contain as little free standing and noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed one percent of the volume.

Contrary to these requirements, on January 8, 2003, the licensee transported a cask containing resin for disposal to Chem-Nuclear Systems, Barnwell South Carolina, which contained greater than one percent liquid by volume. Because this failure to comply with 10 CFR 61.56(b)(2) is of very low safety significance and has been entered into the licensee's corrective action program as PIP O-03-00624, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000269,270,287/2003005-02, Failure to Comply with 10 CFR 61.56(b)(2) Waste Characteristic Requirements involving the liquid content waste shipped to a licensed burial site for disposal.

(2) Introduction: A Green, self-revealing NCV was identified for the failure to comply with 10 CFR 61.55(2)(ii), in that the licensee transported a shipment of waste (resin) for disposal to Chem-Nuclear Systems, Barnwell, South Carolina, which was under-classified as a Class A Stable waste shipment when it was later determined by the licensee to be a Class B waste shipment.

Description: On January 8, 2003, the licensee offered a waste shipment in a 14-215H-2 cask, HIC L501039-98, Shipment Identification No. 0103-11647 (Oconee Shipment Manifest No. RSR ONS03-2001), for disposal to Chem-Nuclear Systems, Barnwell, South Carolina. During subsequent requests for radwaste radionuclide concentration analyses, a counting room technician noted a worker had used a counting room instrument with an unapproved counting geometry. Discussions among the workers indicated that a technician was unaware of the problem and that he had used the unapproved geometry to analyze the sample for the subject radwaste shipment. Use of the unapproved geometry resulted in changes in the actual radionuclide quantities in the shipment. The cask was originally shipped based on a total activity of 3.518 curies of radioactive material rather than the actual 10.47 curies based on an approved instrument geometry. The differences in curie content resulted in the shipment being reclassified from Class A to Class B waste. On January 14, 2003, the licensee called the burial site identifying the error and resultant change in waste class as a result of a fortuitously identified analytical measurement error.

Analysis: The inspectors determined that the failure to comply with 10 CFR 61.55(a)(2)(ii) requirements for classifying a Class B waste shipment is a performance deficiency, because the licensee is expected to meet the requirements of 10 CFR 61.55(a)(2)(ii). The inspectors determined that the finding was associated with the low level burial attribute of the Public Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain. Therefore, the finding is greater than minor. This finding was evaluated using the Public Radiation Safety Significance Determination Process and was determined to be of very low safety significance (Green) because the licensee notified the burial site prior to final waste disposal.

Enforcement: 10 CFR 61.55(a)(2)(ii) requires waste to be properly classified for disposal in that Class B waste is waste that must meet more rigorous requirements in waste form and must meet both the minimum physical and structural stability requirements set forth in 10 CFR 61.56.

Contrary to these requirements, on January 8, 2003, the licensee transported a cask containing resin for disposal to Chem-Nuclear Systems, Barnwell South Carolina, which was not properly classified. Because this failure to comply with 10 CFR 61.55(a)(2)(ii) is of very low safety significance and has been entered into the licensee’s corrective action program as PIP O-03-00224, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000269,270,287/2003005-03, Failure to Comply with 10 CFR 61.55 (a)(2)ii requirements for classifying waste shipped to a licensed burial site for disposal.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period from October 2002 through September 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2, were used to verify the basis in reporting for each data element.

Mitigating Systems Cornerstone

- Safety System Unavailability for the Residual Heat Removal System (all units)
- Safety System Unavailability for the Heat Removal System (Emergency Feedwater) (all units)
- Safety System Unavailability for the High Pressure Injection System (all units)

- Safety System Unavailability for the Emergency AC Power System (combined)

- Safety System Functional Failures (all units)

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

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

For the review period, the inspectors reviewed data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify occurrences involving EHRAs, VHRAs, and unplanned personnel exposures. The inspectors also interviewed licensee personnel who were responsible for collecting and evaluating the PI data. Documents reviewed are listed in the Attachment to this report.

Public Radiation Safety Cornerstone

- RETS/ODCM Radiological Effluent Occurrences

For the review period, the inspectors reviewed data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify potential radiological effluent occurrences. The inspectors also interviewed licensee personnel who were responsible for collecting and evaluating the PI data. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Annual Sample Review

a. Inspection Scope

The issue of degraded low pressure service water piping installed in containment was reviewed in detail. The inspectors had previously noted that the un-lagged low pressure service water (LPSW) piping in containment appears to be severely corroded. In addition, the inspectors noted previous leaks caused by excessive internal piping corrosion. The inspectors evaluated the licensee's actions against the requirements of the licensee's corrective action program, Nuclear System Directive (NSD) 208, Problem Investigation Process, and 10 CFR 50 Appendix B. The inspectors reviewed 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 the 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 documents were reviewed:

- PIP O-99-02433, identified an excessive corrosion problem of an LPSW drain line that broke during a valve manipulation. The PIP provided an operability assessment for the LPSW piping and recommended corrective actions to replace various sections of LPSW piping 2 inches or less in size. The replacement process was scheduled to start with the Unit 2 outage in the spring of 2004.
- PIP O-02-07286, was initiated to document an ongoing study of the LPSW system inside containment. A study was conducted by Structural Integrity Associates and recommended corrective actions were documented in PIP O-03-07385.
- PIP O-03-07385, initiated on November 13, 2003, documented the results of the study conducted by Structural Integrity Associates, which included various recommendations. The study noted that nondestructive examination (NDE) testing on various sections of the LPSW piping in containment did not identify any piping susceptible to near term leakage. Development of corrective actions to address the various recommendations is due January 19, 2004.
- PIP O-99-02475, identified valves and fittings on the LPSW system in containment were leaking. The valves and fittings were repaired prior to startup.
- PIP O-98-0550, identified that nine mechanical joints were found leaking during performance of the Hydro Pressure Test VT-2 examination of the LPSW system in containment. Corrective actions to repair the leaks appeared to be commensurate with the component that was leaking.
- PIP O-03-07124, identified fouling of the 1A2 RCP motor air cooler. The lower air cooler was found to be 80 percent blocked. Recommendations were made to establish a PM for cleaning the coolers.
- The inspectors also reviewed the results of NDE inspections on the LPSW system piping in containment performed during the Fall 2003 Unit 1 outage. Three inspections performed in November 2003, noted acceptable piping wall thicknesses on 4 inch, 6 inch and 10 inch piping locations.

b. Findings

No findings of significance were identified.

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

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

4OA5 Other Activities

.1 Unit 1 Reactor Pressure Vessel Lower Head Penetration Nozzle Inspection

a. Inspection Scope

The inspectors reviewed activities associated with the inspection of the Unit 1 reactor vessel (RV) lower head penetrations in response to NRC Bulletin 2003-02. The guidelines for the inspection are provided in NRC temporary instruction (TI) procedure 2515/152, “Reactor Pressure Vessel (RPV) Lower Head Penetration Nozzle Inspection” (NRC Bulletin 2003-02).

The inspection included a review of the licensee’s procedures, assessment of inspection personnel training and qualification, and observation and assessment of video documentation of the lower head inspections. Discussions were also held with licensee engineering personnel. The inspectors reviewed results of the licensee’s 100 percent Bare Metal Visual (BMV) examination. The activities and documents listed below were examined to verify licensee compliance with regulatory requirements and gather information to help the NRC staff identify possible future regulatory positions and generic communications.

Specifically, the inspectors reviewed and observed:

- MP/0/A/1150/030, Reactor Vessel - Lower Head Penetrations - Visual Inspection, Revision 2

- Critical Evolution Plan, Unit 1EOC21 Under Vessel Inspection

- Video documentation of BMV exam of U1 Reactor Vessel Lower Head

b. Findings

TI 2515/152 Reporting Requirements:

1.1 Was the examination performed by qualified and knowledgeable personnel?

The BMV examination of the RV lower head was conducted by licensee personnel with prior experience with the identification of boric acid deposits during previous inspections of the upper head penetrations for all three units. The lower head specific training documentation for the inspection personnel performing the BMV examinations were verified. The inspectors verified that operating experience from the South Texas Project Unit 1 examination results were incorporated into the inspectors' training, including photographs of the leaking penetrations. The inspectors found that the licencee's inspection personnel were very knowledgeable and experienced with conducting visual examinations of reactor vessel head penetrations.

1.2 Was the examination performed in accordance with demonstrated procedures?

The inspectors reviewed the applicable inspection procedures and verified they had been reviewed and approved through the licensee's procedure review process.

The BMV examination was performed in accordance with licensee procedure number MP/0/A/1150/030, Reactor Vessel - Lower Head Penetrations - Visual Inspection, Revision 2.

1.3 Was the examination able to identify, disposition, and resolve deficiencies?

The inspectors reviewed the procedures controlling the 100 percent Bare Metal VT-2 examination techniques, and determined that they provided adequate guidance to ensure that they would be able to identify, disposition and resolve relevant deficiencies in the RV lower head penetration materials.

1.4 Was the examination capable of identifying pressure boundary leakage and/or RPV lower head corrosion as described in BL 2003-02?

Based upon review of the results for the BMV examination, procedures, qualifications, appropriate lighting, and sensitivity requirements, the inspectors determined that the licensee was capable of identifying pressure boundary leakage and boric acid corrosion, if present.

2.0 Could small boron deposits, as described in the bulletin, be identified and characterized?

With the available lighting on the video inspection equipment and the clarity of the picture, the inspectors were able to verify that there were no indications of lower vessel head penetration leakage. Had boron deposits been present, as described in the bulletin, they could have been readily identified and characterized.

3.0 How was the visual inspection conducted?

The licensee utilized a combination direct visual observation with closeup video documentation.

4.0 How complete was the coverage?

Full 360 degree coverage around the circumference of all nozzles was achieved.

5.0 What was the condition of the reactor vessel lower head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

Prior to the lower head inspection all the insulation was removed, and the reactor vessel bottom head was entirely accessible for the BMV inspection. The lower vessel head had been originally coated with a silver metallic paint which exhibited uniform peeling and flaking on the vessel surface. Metal surface corrosion was identified on the lower vessel head exposed surfaces. There was no corrosion identified on the in-core guide tubes. Each of the 52 penetrations was videoed such that a complete 360-degree view of each penetration was obtained. Boron deposits were not noted by the inspectors on any of the lower pressure vessel surfaces. The inspectors did not see any “popcorn” type boric acid crystals at the penetration/vessel interface. There was no wastage, corrosion or cracks that needed repair. The inspection results were documented in MP/0/A/1150/030. The inspectors reviewed the video of the bottom head inspection to verify the licensee’s inspection results, and held discussions with the appropriate engineering and examination personnel.

6.0 What materiel deficiencies (associated with the concerns identified in the bulletin) were identified that required repair?

No material deficiencies were identified.

7.0 What, if any, impediments to effective examinations were identified.

There were no significant items that could impede effective examinations. The licensee was able to inspect 360 degrees around each of the 52 lower head penetration nozzles.

8.0 Did the licensee perform appropriate follow-up examinations for indications of boric acid leaks from pressure-retaining components above the RPV lower head?

There was no indication of boric acid leaks from pressure-retaining components above the RPV lower head.

9.0 Did the licensee take any chemical samples of any deposits?

There were no deposits present therefore no chemical samples taken.

10.0 Is the licensee planning to do any cleaning of the head?

The licensee pressure washed the lower head following the inspection to remove the loosely adherent metallic paint and video documented the as left condition.

11.0 What are the licensee's conclusions regarding the origin of any deposits present?

There were no deposits noted and therefore the licensee concluded that no leakage of the unit 1 lower head penetrations exists.

.2 Unit 1 Reactor Containment Sump Blockage Inspection

a. Inspection Scope

The inspectors reviewed activities associated with the inspection of the Unit 1 containment sump blockage concerns in response to NRC Bulletin2003-01. The guidelines for the inspection were provided in NRC temporary instruction (TI) 2515/153, Reactor Containment Sump Blockage (NRC Bulletin 2003-01).

The inspection included a review of the licensee's response describing interim compensatory measures (Option 2), inspection of the containment emergency sump, observations of repairs performed on the emergency sump, and a detailed inspection of sections of containment to identify debris still left in containment following the refueling outage.

Specifically, the inspectors reviewed and observed:

- The licensee's response to NRC Bulletin 2003-01 documented in a letter dated August 7, 2003

- PIP O-03-4376, which tracked Bulletin 2003-01 and the associated corrective actions

- Repairs to the containment emergency sump documented in work orders WO-98634160

b. Findings

TI 2515/153 Inspection of Responses Describing Interim Compensatory Measures (Section 03.02)

1. The inspector verified that an operation's focus item was prepared and covered by the Operation Shift Manger to raise awareness of this issue. The inspector verified that operators had been trained on the emergency operating procedure (EOP) changes associated with changes to the LPI and RBS systems.

2. The inspector verified that modifications were installed on Unit 1 for throttling the LPI and RBS systems. The inspector verified that the Units 2 and 3 EOPs were modified to throttle LPI flow.

3. The inspector verified that EOP guidance for refilling the borated water storage tank (BWST) currently exist. The inspector noted that EOP guidance for refilling the BWST was moved forward in the EOP to notify the Technical Support Center to start refilling sooner.

4. The inspectors performed a detailed inspection following the licensee's cleanliness inspection. Although some debris was identified, the cleanliness of the containment was considered to be good by the inspectors. A list of items identified during the inspection were given to the licensee and forwarded to NRC headquarters.

5. An inspection to ensure drainage paths are unblocked will performed during a later inspection period.

6. The licensee's response noted that reactor building emergency sump (RBES) inspections are performed every refueling outage and consist of three different independent inspections. However during the Unit 1 refueling outage, gaps at all four corners of the RBES structure and tears in the sumps screens were identified by the licensee as discussed in PIP O-03-7864. The licensee noted that these gaps and tears existed during the previous inspections but were not identified. The inspectors examined the deficiencies and reviewed the licensee's repair plan. Following the licensee's repairs to the RBES and QC inspections of the repairs, inspectors found that the licensee had failed to repair one of the previously identified screen tears. Additionally, the inspectors identified that the licensee had failed to weld the correct location of the RBES, thereby leaving the corner gaps intact. This information was passed on to the licensee and was documented in PIP O-03-7930. Following subsequent repairs to the RBES, the inspectors verified the adequacy of the repairs to the deficient corner welds and the screen tear.

<u>TI2515/153 Inspection of the Containment Sump and Condition Assessments (Section03.03)</u>

The inspection and reporting on the licensee's performance of containment condition assessments will be completed during a later inspection period.

<u>TI2515/153-05 Reporting Requirements</u>

a. For units that entered refueling outages (RFOs) after August 31, 2002, and subsequently returned to power: Was a containment walkdown to quantify potential debris sources conducted by the licensee during the RFO?

   A walkdown of containment was conducted by the licensee to quantify potential debris sources.

b. For units that are currently in a RFO: Is a containment walkdown to quantify potential debris sources being conducted during the current RFO?

   A walkdown was conducted on Unit 1 during the present refueling outage.

c. For units that have not entered a RFO between September 1, 2002, and the present: Will a containment walkdown to quantify potential debris sources be conducted during the upcoming RFO?

   A walkdown will be performed on Units 2 and 3 when they enter their refueling outages.

d. Did the walkdowns conducted check for gaps in the sumps' screened flowpath and for major obstructions in containment upstream of the sumps?

   The walkdown did check for gaps in the emergency sump screens and plant design prohibits major obstructions in the flow paths to the sumps.

e. Are any advanced preparations being made at the present time to expedite the performance of sump-related modifications, in case it is found to be necessary after performing the sump evaluation?

   There are no advance preparations being made at the present time to expedite any sump related modifications.

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

a. Inspection Scope

The inspectors reviewed the licensee's procedure for loading spent fuel shipments to the ISFSI (MP/0/A/1500/016) and reviewed associated PIPs (O-03-00127, 00437, 00730, 00732, and 04272) to verify that the ISFSI shipment activities for 2003 were performed in a safe manner and in compliance with the approved procedure.

The inspectors reviewed the licensee's completed Oconee Nuclear Engineering Instruction (ONEI-400) for Dry Storage Certification for selected ISFSI shipments 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 reviewed selected completed procedures for physical inspection and inventory of the ISFSI (IP/0/A/0750/003, Physical Inventory of Reportable Special Nuclear Material) and completed ONEI-400s 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.

.4 (Closed) Unresolved Item (URI) 05000269,270,287/2003003-03, Failure to Implement Manufacturer's Recommendations for Replacement of SSF Diesel Generator (DG) Coolant Grommets. The initial inspection was conducted Inspection Report (IR) 05000269,270,287/2003003 and specifically covered a review of the problems identified by Engine Services, Inc.(ESI) who was contracted by the licensee to provide technical oversight for the 10-year SSF diesel overhaul. A detailed description of the as-found condition of the grommets is provided this IR.

Introduction: A Green finding was identified for failure to implement the diesel manufacturer's 6-year recommended preventive maintenance for grommet replacement. Consequently, at 10 years some of the grommets were found to be "at or near failure".

Description: During various discussions regarding the grommets, the licensee noted that the diesel manufacturer had recommended a 6-year replacement interval for these grommets. However, the grommets were being replaced on a 10-year interval. In October 2002, the used grommets were sent off to ESI for analysis. On May 8, 2003, the results of the ESI analysis were received by the licensee. The report noted that many of the grommets had exceeded their useful life and some of the grommets were at or near failure. The report noted that "Diesel engines used in standby service see thermal cycling which contributes to the hardening of the grommets. Therefore, the recommended replacement interval is on a 6-year calendar basis."

Analysis: Failure of the grommets could have led to diesel coolant leaks and loss of cooling to the diesel. The finding is more than minor because it is associated with the reactor safety mitigating system cornerstone objective to ensure the availability, reliability, and capability of a system that responds to initiating events to prevent core damage. The results of the Phase 1 and Phase 2 analyses were documented in IR 05000269,270, 287/2003003. A regional Senior Reactor Analyst performed a Phase 3 evaluation under the Significance Determination Process. The evaluation results indicated that the performance deficiency was of very low safety significance (Green). The evaluation was performed assuming:

- the maximum number of grommets failing was four, with the earliest the DG could have failed within the mission time was eight hours.

- a modification of offsite power recovery probabilities over the full extent of the exposure time through regression analysis, stretching back one year or until the 24 hour mission time could be confirmed by testing.

- replenishment of the DG cooling water thru operator actions was credible with ample time to accomplish the recovery actions.

The primary difference between the Phase III analysis and the Phase II results (greater than Green) rested in recovery credit (i.e., SSF DG cooling replenishment and recovery of offsite power) and in crediting the Auxiliary Air System to maintain the Component Cooling Water air operated valve(s) to Reactor Cooling Pump thermal barrier cooling open upon loss of normal instrument air. The Phase II results do not include a replenishment term.

Enforcement: No violations of regulatory requirements were identified.

.5 (Closed) URI 05000269,270,287/2000008-01: Risk Significance of High Temperatures in the Low Pressure Injection and High Pressure Injection Pump Rooms. The inspectors reviewed licensee design calculation OSC 6667, which was revised to more accurately predict post loss of coolant accident (LOCA) room temperatures and reactor building spray pump testing results, in order to assess the risk significance of post LOCA pump room temperatures on safety-related pumps.

Introduction: A NCV of 10 CFR 50, Appendix B, Criterion III, was identified by the inspectors for failure to properly translate design basis parameters for emergency core cooling systems (ECCS) into applicable specifications, drawings, procedures, and instructions. Specifically, design calculation OSC-6667 documented that post LOCA temperatures in the low pressure injection (LPI) and high pressure injection (HPI) pump rooms could reach ambient temperatures as high as 257 degrees; however, safety-related pumps and motors in those rooms (i.e., LPI, HPI, and reactor building spray pumps and motors) were not environmentally qualified for this type of environment.

Description: In December 2000, the inspectors questioned the need for LPI and HPI room coolers following post LOCA conditions. Design calculation OSC 6667 had documented that room temperatures could reach 257 degrees F. Based on the motor and pump bearings being cooled by ambient air within the rooms and manufacturer's guidance that the equipment should be shut down if bearing temperatures exceeded 210 degrees, the operability of the LPI, RBS and HPI pumps was called into question. This issue was originally documented in IR 05000269,270,287/2000008. Subsequent to this IR, the licensee: 1) revised design calculation OSC-6667 and accordingly reduced the anticipated room temperatures that would be expected following a LOCA; 2) performed analyses of varius bearings and motors to demonstrate that the equipment would operate under post LOCA conditions for time frames specified in accident analyses; and 3) completed testing of a RBS pump and motor under the anticipated post LOCA temperature conditions. The testing results indicate that the LPI and RBS pumps will remain operable under the adverse room temperature conditions following a LOCA. The HPI pumps were not tested under actual post LOCA conditions since the analyses clearly demonstrated that the equipment would operate satisfactorily.

Analysis: The inspectors concluded that failure to ensure that safety related equipment would continue to function during accident conditions was a performance deficiency. The finding was considered to be more than minor because it is associated with the mitigating system cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent core damage. The finding was determined to be of very low safety significance (Green) due to the fact that actual testing and analyses indicated that the pumps and motors could operate successfully at ambient temperatures without adverse consequences, so there was no loss of function. Therefore the issue was screened out in Phase 1 of the SDP as Green.

Enforcement: 10 CFR 50, Appendix B, Criterion III, requires, in part, that measures shall be established to assure the design basis for structures, systems, and components are translated into specifications, drawings, procedures, and instructions. Design calculation OSC-6667 documented that post LOCA temperatures in the LPI and HPI pump rooms could reach ambient temperatures as high as 257 degrees. Contrary to the above, the licensee failed to establish accurate room temperatures that would be expected following a LOCA. In addition, the licensee failed to provide adequate assurance that the LPI, RBS and HPI pumps and motors were capable of operating within these temperature limits. Because this failure to comply 10 CFR 50, Appendix B, Criterion III, is of very low safety significance and has been entered into the licensee's corrective action program as PIP O-99-00193, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000269,270,287/2003005-05, Design Calculation Contains Inaccurate Post LOCA

Room Temperatures and a Lack of Assurance that Safety-Related Pumps were Capable of Operating in this Temperature Environment.

.6 Steam Generator Replacement Inspection Overview

This inspection report documents completion of inspections required by IP 50001, Steam Generator Replacement (SGR) Inspection, some of which were completed in accordance with baseline inspection procedures. The table below identifies and correlates specific IP 50001 inspection requirements examined during this inspection period with the corresponding sections of this report.

IP 50001 Section	Inspection Scope	Section of This Report
02.02.b 02.03.b	Rigging/Lifting Activities	4OA5.10
02.03.d	Restoration of Temporary Containment Opening	4OA5.8
02.03.e.3	Implementation of Foreign Material Exclusion (FME) Controls	4OA5.7
02.02.c 02.03.e.2 02.03.f	Implementation of Radiation Protection Controls	4OA5.7
02.04.1	Containment Testing	1R19
02.04.2	Post-installation Inspections and Verifications Program	4OA5.12
02.04.3 02.04.4	Leakage Testing (VT-2 inspections)	1R19
02.04.5	Calibration and Testing of Instrumentation	4OA5.13
02.03.a 02.04.7	Welding, Nondestructive Examination (NDE), and Preservice Inspection	4OA5.11

.7 SGR FME Controls and Radiation Protection Controls

a. Inspection Scope

Throughout this inspection period, the inspectors routinely inspected the following activities as they occurred:

- Implementation of foreign material exclusion (FME) controls - As required by IP 50001 Section 02.03.e.3, the inspectors periodically observed the implementation of FME controls for various RCS and steam generator (SG) openings to ensure the openings were sealed to prevent the introduction of debris into these systems. The inspectors also observed material control procedures implemented by contractor personnel during work on the 1B RCS hot leg, including the use of FME personnel and material accountability logs. The inspectors also reviewed the circumstances surrounding the loss of FME control that led to the introduction of sealant material into the 1B RCS hot leg piping.

- Implementation of radiation protection controls - The radiation protection program inspections required by IP 50001 Sections 02.02.c, 02.03.e.2, and 02.03.f were completed in accordance with applicable portions of baseline inspection procedures IP 71121.01 and 71121.02. The inspectors performed walkdowns of the reactor building to verify that the appropriate radiation postings were displayed and that radiation protection (RP) personnel were assigned to provide RP job coverage. The inspectors also discussed contamination control plans related to the removal of the steam generators from the reactor building with the project RP coordinator. Additional inspection details are delineated in Sections 2OS1 and 2OS2 of this report.

b. Findings

Introduction: A Green self-revealing NCV was identified for an inadequate maintenance procedure for inspection of the FME barrier in the 1B hot leg during steam generator replacement. This allowed the introduction of sealant material in the RCS piping.

Description: On October 5, 2003, contractor personnel installed an FME barrier in the 1B hot leg to support the initial cutting and removal of the 1B hot leg "candy cane". The procedure used to perform this task was QEP 11.01-3, Work Package 12540B, Perform RCS Pipe Cuts SG 1B. The instructions for this operation consisted of a line in the work package that stated, "Install an FME barrier in the RCS H/L riser piping below the cut location. Reference attachment 5." Attachment 5 is a drawing showing that the barrier location should be a minimum of 6 inches from the cut location with no other specific details of how the barrier should be installed. A quality control inspection sign-off followed the instructions to verify proper FME barrier installation. This was the extent of the procedural instructions. The FME barrier consisted of an inflatable bladder which was to be sealed with RTV sealant along the bladder/pipe interface. The bladder was to be placed in the pipe such that when the RTV was pumped at the top of the bladder it would flow down around the bladder/pipe interface thus completing the seal. However, the bladder was not installed at the appropriate location in the pipe and in the proper orientation. In actuality, a weight attached to the bladder used to orient the bladder correctly was pinned to the back of the pipe resulting in a gap between the bladder and the pipe where the RTV was pumped.

At this point in the procedure, the Quality Control (QC) person signed that the barrier was installed per QEP 10.04 which is the contractor's housekeeping, FME and cleanliness procedure. There were no specific attributes for how 1B Hot Leg barrier was to be installed. Bladder location and orientation in the pipe was critical for successful sealant application. A camera was utilized by QC to verify the installation but the bladder was too far down the pipe for the camera to see behind the bladder. Since there was no specific criterion for inspecting the bladder to pipe fit-up in the procedure or no specific details in the referenced drawing, the QC person was able to sign off the step for proper installation of the FME barrier. The sealant was then pumped in to complete the seal; however, due to the gap in the back, all of the sealant (approximately 1.5 gallons) was pumped down the pipe. There was no QC inspection after this step and the pipe section was cut and removed. The licensee discovered the sealant when the bladder was removed. In addition to the sealant, metal pipe cuttings were discovered that were allowed to fall down the pipe due to the lack of a seal. The licensee was successful in removing the foreign material from the hot leg during the outage.

Analysis: The finding was considered to be more than minor because it potentially affected the barrier integrity cornerstone, as foreign material, if left uncorrected, could have an adverse impact on fuel cladding integrity during operation. Additionally, inadequate inspection activities, if not corrected could have adverse consequences on future activities affecting quality. The finding was determined to be of very low safety significance (Green) due to the fact that the foreign material was successfully removed and that the SDP phase 1 screening for findings for potentially affecting the fuel barrier screen as Green.

Enforcement: 10 CFR 50, Appendix B, Criterion V, requires that activities affecting quality shall be prescribed by documented instructions, procedures or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures and drawings. Instructions for installation of an FME plug in the RCS hot leg are provided by QEP 11.01-3, Work Package 12540B, Perform RCS Pipe Cuts SG 1B, which states, “Install an FME barrier in the RCS H/L riser piping below the cut location. Reference attachment 5.” Attachment 5 is a drawing showing that the barrier location should be a minimum of 6 inches from the cut location with no other specific details of how the barrier should be installed. A quality control inspection sign-off to verify proper FME barrier installation was included with this step. Contrary to the above, instructions for inspection of the 1B hot leg FME barrier were not appropriate, such that the incorrect installation of the barrier was not identified and failed to prevent the introduction of foreign material in the RCS piping. Because the inadequate inspection activities were of very low safety significance and have been entered into the licensee’s corrective action program (PIP O-03-06560), this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000269/2003005-04, Inadequate maintenance procedure for inspection of RCS FME barrier.

.8 Containment Restoration Activities

a. Inspection Scope

The inspectors examined restoration activities associated with the temporary construction opening (approximately 23.5 feet by 25 feet) in the containment liner, as detailed in the licensee’s Modification Package ON-13086, Part AS9, Containment Opening, Revision 3, and Work Plan13550, Containment Opening Concrete Installation, Revision 3.

Activities associated with containment liner plate welding were observed/reviewed and compared with the applicable codes (ASME Boiler and Pressure Vessel Code (B&PV), Section VIII, 1998 Edition with no Addenda and Section XI, 1989 Addenda with no Addenda) and Oconee Specification OSS-0139.00-00-0004, Specification for Field Welding of Reactor Building Liner Plate By Manual Meta-Arc Process. The inspectors reviewed controls for the full penetration liner plate welds and the associated stiffener plate welds. For the liner plate welds (LP-1, LP-2, LP-3, and LP-4), the inspectors: visually inspected a sample of the final weld surfaces (inside and outside containment); observed outside surface back-grinding in preparation for final welding; observed in-process welding and inspection (magnetic particle (MT)) activities for the outside weld; and reviewed the final radiographic (RT) film, including rejects and repair film. In addition to observation of in-process work, the inspections included: review of the welding procedure specification, including the supporting procedure qualification records; review of welder qualification records; review of welding material testing and certification records; observation of welding material use control; review of in-process Weld Data Sheets; review of Quality Control (QC) involvement in the welding process; review of MT and RT examination records for the completed liner plate weld; and review of QC and nondestructive examination (NDE) personnel qualification and certification records.

The inspectors reviewed activities associated with installation of the containment opening reinforcing bar (rebar) and compared activities with the applicable Codes (ACI 318-63, Part IV-B, Building Code Requirements for Reinforced Concrete Institute, 1963; AWS D1.4-98, Structural Welding Code-Reinforcing Steel; and ASME Section III, Division 2, 1995 Edition, 1995 Addenda). The inspectors observed in-process mechanical splicing (Barsplice swaged couplers) of splices CO13-0505-H-1-A, H-5-A, H-4-A, H-3-A, H-2-A, H-6-A, H-7-A, H-8-A, H-9-A, H-10-A, and Sister Splice 6149 (11/5/03). In addition, the inspectors reviewed: (1) testing results for Sister Splices RS 243526 #19 Horizontal, LS 238362 #8 Horizontal, LS 238362 #71 Vertical, JG 243105 #54 Horizontal, LS 238362 #59 Horizontal, LS 238362 #76 Vertical, WLR 238891#30, and BT 100255 #43 Horizontal and (2) qualification records for seven splicers.

The inspectors also reviewed Modification Package ON-13086, Part AS9, Containment Opening, Revision 3, to verify that the modification was properly evaluated in accordance with 10 CFR 50.59.

Relative to installation of concrete, the inspectors witnessed placement of concrete in the containment wall to restore the temporary construction opening. The inspectors examined the reinforcing steel to ensure it was installed in accordance with design requirements, observed the concrete forms to ensure tightness and cleanliness, and that reinforcing steel was clean. The inspectors reviewed placement activities to ensure that activities pertaining to concrete delivery time, free fall, flow distance, layer thickness and concrete consolidation conformed to industry standards established by the American Concrete Institute. Concrete batch tickets were examined to ensure that the specified concrete mix was being delivered to the site. The inspectors also witnessed testing of the plastic concrete for slump, air, and temperature, unit weight, and molding of the concrete cylinders for testing. Reviews were performed to ensure concrete testing was performed and the cylinders were molded in accordance with applicable American Society for Testing and Materials (ASTM) requirements. In addition, the inspectors reviewed activities to ensure that concrete testing was performed by qualified inspectors, and that concrete placement activities were continuously monitored by licensee and contractor quality control and quality assurance personnel.

The inspectors reviewed concrete batching activities including proper storage and separation of materials, and temperature controls. The inspectors reviewed results of quality control acceptance testing performed on materials (cement, fine and coarse aggregate, water, and admixtures) used for batching the concrete. The inspectors also reviewed records documenting inspection of the concrete batch plant and the concrete truck mixers. Activities were reviewed to determine if the contractor's inspection of the trucks and batch plant were performed in accordance with the guidance of the National Ready Mixed Concrete Association (NRMCA); the batch plant scales were calibrated in accordance with NRMCA recommendations; and mixer efficiency tests were performed on the truck mixers in accordance with ASTM C-94. The inspectors reviewed the concrete mix data to ensure that mix proportions for delivered concrete were selected based on trial concrete mix results, that QC acceptance criteria for the plastic concrete were based on the trail mixes, and that the trail mix met concrete strength requirements. In addition, the inspectors reviewed the results of unconfined compression tests performed on concrete test cylinders to verify that the concrete met design strength prior to tensioning the replacement tendons. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.9 Reactor Pressure Vessel Head (RPVH) Replacement Activities

a. Inspection Scope

The inspectors observed work activities and reviewed fabrication records for the Unit 1 replacement RPVH as delineated in IP 71007. 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 RPVH forging, lifting lugs, control rod drive mechanism (CRDM) flanges, and CRDM tubes. Fabrication records reviewed included the results of radiographic examination of the sixty-nine (69) CRDM tube-to-flange welds, and heat treatment records for the replacement head assembly.

The inspectors also reviewed the corrective actions for the ASME rejectable liquid penetrant indications adjacent to nozzles 58, 60, and 68, reported on PIP O-03-05397. The inspectors reviewed the work orders for examination and grinding repair of the indications; completed NDE forms; certifications for PT materials used; and MP/O/A/1800/022, the controlling procedure for the repair and final examination. In addition to reviewing the documentation, the inspectors conducted a visual examination of the repaired areas to verify that the grinding repair did not appreciably affect the cladding thickness of the inside surface of the head.

The radiation protection program inspections required by IP 71007 Sections 02.02.c, 02.04.d.2, and 02.04.e were completed in accordance with applicable portions of baseline inspection procedures IP 71121.01 and 71121.02. Specific inspection details are delineated in Sections 2OS1 and 2OS2 of this report.

b. Findings

No findings of significance were identified.

.10 Review of Steam Generator Replacement Program (SGRP) Lifting and Transportation

a. Inspection Scope

The inspectors reviewed the adequacy of the SGRP rigging and handling program as described in ON-13086 AS6, “Steam Generator Rigging and Handling,” Rev. 0E1 to verify compliance with regulatory requirements, appropriate industrial codes, and standards, ANSI N45.2.15, Generic Letter 81-07 and NUREG 0612.

The inspectors examined portions of the SGRP lifting equipment necessary to perform steam generator rigging and transport, design evaluation/erection/use of the Outside Lift System (OLS) and Temporary Lifting Device (TLD), Hatch Transfer System (HTS), and a Self Propelled Modular Transport (SPMT). The inspectors reviewed the applicable engineering design, modification and analysis associated with SG lifting and rigging including: crane and rigging equipment, steam generator drop analysis, safe load paths, and load drop protection. The inspectors determined if appropriate load tests and functional tests were performed or documented in accordance with the ASME/ANSI code for both the TLD, OLS, and lifting links. The inspectors determined if the TLD and OLS cranes were operated by qualified and certified personnel, and that wire ropes and synthetic slings used during heavy lifts were appropriately tested and inspected prior to use. The inspectors determined if the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures.

For changes to the facility design as described in the UFSAR, the inspectors reviewed the10CFR 50.59 screens for modification packages. For those modification packages that did not involve a change to the facility as described in the UFSAR, a 10CFR 50.65 Risk Assessment was done and reviewed by the inspectors. The inspectors determined if Oconee Operations was aware of the potentially impacted Instrument Air (IA) and Low Pressure Service Water (LPSW) systems and if they had a contingency plan to isolate those two systems in case of an accident.

The inspectors also observed various portions of the original steam generators (OSG) being lifted from the steam generator cubicle through the temporary opening in the reactor building utilizing the TLD installed in the Containment Building, the HTS, the OLS, onto the SPMT. The inspectors also observed various portions of the replacement steam generators (RSG) being lifted back into the containment. During these observations the inspectors performed visual inspections of the TLD, HTS, OLS, and SPMT. For the task of rigging and movement of the SGs, the inspectors reviewed the work packages and procedures for content, technical adequacy and to verify that appropriate line items had been signed off and that required pre-lift equipment inspections had been performed and documented in the enclosures provided. This review was also to verify that operating experience was utilized and reflected in the procedures. Documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.11 SGRP Welding, NDE, and Preservice Inspection

a. Inspection Scope

Fit-up and Welding Activities

The inspectors conducted inspections of the fit up, welding and post weld heat treatment (PWHT) activities related to the SG replacement activities involving portions of Reactor Coolant System (RCS), Main Steam System (MS) and Feedwater System (FDW) piping. The replacement Reactor Coolant System (RCS) piping was procured in accordance with ASME Section II Part A, 1998, no addenda as directed by ANSI B31.1, 1998. The reinstallation and inspection of piping activities were performed under ASME Section III, Subsection NB, NC, 1989 no addenda, which is considered to be an acceptable substitute to the original B31.1 design code.

The inspectors reviewed records for calibration, examination results, fit-up, welding, certifications of personnel, materials, as-built configuration, and held discussions with cognizant engineering personnel. The fit-up inspection was to verify that the amount of movement, as-built “gap” associated with the cutting and fit up of the RCS, MS and FDW piping, for both SGs, was within specification allowable tolerance requirements and applicable codes.

The inspectors observed portions of the automatic welding of RCS hot-leg and cold-leg, MS and FDW system piping connections on both steam generators. The inspectors determined if the operators at the welding site and the operator at the control panels were in constant communication through head sets. The inspections, during welding operations determined if the welding machine settings were being maintained within the qualified welding parameters listed in the welding procedure specification. The inspectors reviewed WPS GT/1.1, Rev. 0E1, ASME Section IX Welding Procedure Specification and WPS GT/3.1, Rev. 0E1, ASME Section IX Welding Procedure Specification that provided the specifications for the automated welding performed on the RCS, FDW and MS piping.

Training and Qualifications

The inspectors observed work, examined selected records and reviewed procedures to evaluate the licensee's training and qualification efforts for personnel performing cutting, machining, welding and NDE. The inspectors also reviewed the programs and compared them with the regulatory requirements and codes that were utilized during the SGRP.

NDE Activities

To verify that the NDE activities including Ultrasonic Testing (UT), Magnetic Particle Testing (MT), Liquid Penetrant Testing (PT) and Radiographic Testing (RT) showed that the welds were free of rejectable indications, the inspectors reviewed NDE documentation including radiographs of completed RCS hot leg and cold legs, FDW and MS welds to verify compliance with ASME Code Section III, Class 1, 1989 Edition, No Addenda, ASME Section V, 1989 Edition, No Addenda, and ASME Section XI, 1998 Edition, 2000 Errata.

The inspectors reviewed NDE records which included Work Packages, NDE Test Reports UT, MT, PT and RT, equipment certifications, consumables certifications, NDE examiner certification and visual acuity documentation. For the RT exams the inspectors reviewed for proper penetrameter or wire type, size, placement, and sensitivity as well as film density, identification, quality, and weld coverage. Records were reviewed for completeness, accuracy and technical adequacy. The radiographs were examined for both film quality and acceptability.

Pre-Service NDE and Baseline Eddy Current of Replacement Steam Generators

The inspectors reviewed the baseline eddy current data as issued in the “Preservice Eddy Current Inspection, BWC-TR-2003-010, Rev. 0 and BWC-TR-2003-011 Rev. 0. The inspectors reviewed aspects of the examination program for the B&W Once-Through Steam Generators, which included full length bobbin examination of all tubes including bobbin profilometry of each tube end and 100% X-Probe acquired data. X-Probe data was only analyzed for special interest areas and a sample of the Manufacturing Burnish Marks (MBM) indications. The inspection was to determine if all examinations were in compliance the NRC Regulatory Guide 1.83, Oconee Technical Specifications (TS), and Section XI of the 1998 ASME Code with 2000 Addenda. The inspectors reviewed the final results of the SG examinations of the two steam generators.

b. Findings

No findings of significance were identified.

.12 Post-installation Inspections and Verifications Program

a. Inspection Scope

As required by IP 50001 Section 02.04.2., the inspectors reviewed the licensee’s post-installation inspections and verifications program. The inspectors reviewed the Unit 1 Startup Test Matrix and the SGRH Project Integrated Startup Plan to verify that the appropriate post-installation testing was identified and scheduled accordingly and that the required ASME code VT-2 inspections and operational steady state data collection were included. The following PIPs were reviewed to determine that any deficiencies were dispositioned appropriately: O-03-08083, O-03-08110, O-03-08151.

b. Findings

No findings of significance were identified.

.13 Calibration and Testing of Instrumentation

a. Inspection Scope

As required by IP 50001 Section 02.04.5., the inspectors reviewed the completed calibration test procedures for the EFW steam generator level, SSF auxiliary service water steam generator level, steam generator startup level, and steam generator full and operating range level transmitters for the A and B replacement steam generators. The inspectors reviewed the test documentation to verify that the calibrations were performed in accordance with the licensee's approved test procedures and to ensure that the “as left” transmitter output were within the acceptance criteria. The following calibration procedures were specifically reviewed.

- IP/0/A/0275/014, Steam Generator Startup Level Instrument Calibration
- IP/0/A/0275/019 B, Emergency Feedwater System Steam Generator Level Instrument Calibration
- IP/0/A/0275/015, Steam Generator Full and Operate Range Levels Instrument Calibration
- IP/0/A/0375/001 B, SSF Auxiliary Service Water System Steam Generator Level Instrument Calibration

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Ron Jones, Site Vice President, and other members of licensee management at the conclusion of the inspection on January 12, 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.

4OA7 Licensee Identified Violation

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

- TS 5.4.1 requires that written procedures be established, implemented, and maintained covering activities related to procedures recommended in Regulatory Guide 1.33 Rev. 2, Appendix A, 1978. Regulatory Guide 1.33, Section 6.a. covers emergency procedures to combat loss of coolant. Contrary to the above, the licensee failed to adequately maintain their emergency operating procedure (EOP) for loss of coolant following a procedure revision such that the EOP allowed the potential to line up the alternate boron dilution path at a RCS temperature and pressure that could result in sump screen damage and loss of coolant injection. The procedure change that introduced the error was approved December 20, 2001. The deficiency was discovered July 11, 2002. The licensee entered the deficiency into their Corrective Action Program under PIP O-02-03709 and immediately placed the procedure on hold and made the appropriate revisions to correct the EOP.

  The performance deficiency was evaluated using the Phase 2 SDP sheets which indicated the finding was greater than Green. Therefore, the issue was forwarded to the Regional SRA for a Phase 3 analysis. The licensee performed calculations that indicated boron dilution would only be required for some transients, and for a narrow range of LOCA break sizes. The small LOCAs which have the highest LOCA initiating event frequency, were outside the range of break sizes requiring boron dilution. The reduction in total initiating event frequency for the events of interest, and the mitigating systems shown by calculation to be available in the event of the blowdown, caused by opening the alternate boron dilution flowpath, reduced the CDP to a number in the Green range. The overestimation by the Phase 2 sheets was due to the early assumptions made with respect to the finding, which were appropriate at the time.

- 10 CFR 20, Appendix G, sections I. C. 8. and 11. requires the shipper of radioactive waste to provide, on the shipment manifest, the approximate volume and total radioactivity within each container. Contrary to the above, for a shipment of de-watered ion exchange resin sent to Chem-Nuclear Consolidation Facility on July 30, 2003, the waste volume was understated by 20.5 cubic feet and the total radioactivity was understated by 5.7 curies on the shipping manifest. This occurrence is documented in the licensee's Corrective Action Program under PIP O-03-07572. This finding is of very low significance because it did not result in the waste shipment being under-classified.

- 10CFR50.55a(g)(4), which requires meeting the ASME Boiler and Pressure Vessel Code, 1989 ASME Section III, Subsection NCA-3856 Identification, Marking, and Material Control, NCA 3856.1 General, which states in part that controls shall be established to assure that only correct and acceptable material or source material is used. Identification shall be maintained on these materials or on documents traceable to these materials, or in a manner which assures that the identification is established and maintained.

Also, 10 CFR Part 50 Appendix B, Criterion IX, Control of Special Processes, states in part that measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. 10 CFR Part 50 Appendix B, Criterion VIII, Identification and Control of Materials, Parts, and Components, states in part that measures shall be established for the identification and control of materials, parts, and components, including partially fabricated assemblies.

Contrary to the above, during the Unit 1 EOC-21 refueling/steam generator replacement outage in the Fall of 2003, established controls failed to assure that the ASME code was appropriately implemented. Specifically, the contractor improperly documented who actually worked on each individual weld and what weld materials were utilized while performing work on the Main Steam and Feedwater piping welds. This violation is considered to be of very low safety significance since the licensee/contractor initiated a stand down on all welding until the problem was reviewed, and appropriate corrective actions were taken. This issue is documented in PIP O-03-7118, Weld & Welder Identification Stamping, QA Weld Records and Weld Filler Metal Traceability and PIP O-03-7224, Base Metal and Filler Metal Traceability, weld & welders identification, QA weld records.

- 10 CFR Part 50.55a(g)(4) requires meeting the ASME Boiler and Pressure Vessel Code Section XI, IWA-7000, Replacement, and IWA-7220, Verification of Acceptability, which states in part that prior to authorizing the installation of an item to be used in replacement, the Owner shall conduct an evaluation of the suitability of that item.

  10 CFR Part 50 Appendix B, Criterion VII, Control of Purchased Material, Equipment, and Services, requires in part that measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. 10 CFR Part 50 Appendix B, Criterion IV, Procurement Document Control, requires in part that measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its contractors or subcontractors. 10 CFR Part 50 Appendix B, Criterion XVII, Quality Assurance Records, requires in part that sufficient records shall be maintained to furnish evidence of activities affecting quality.

  Contrary to the above, during the Unit 1 EOC-21 refueling/steam generator replacement outage in the Fall of 2003, measures taken to evaluate the suitability of replacement parts were not adequate in that they did not identify the non-conformance of the hot leg elbows to the design requirements, rejectable indications during the NDE-RT examination process, nor was the as-found condition of the elbows documented properly. This violation is considered to be of very low safety significance since the hot leg elbows, after numerous additional testing and reviews, were determined to comply with all design and procurement requirements prior to putting Unit 1 back into service. This issue is documented in PIP O-03-6224, Possible Problems with Replacement Hot Leg Elbows, PIP O-03-7044, Review of NDE Documentation for Hot Leg Elbows, and PIP O-03-6362, Hot Leg Elbow Fabrication Concerns.

- 10 CFR Part 50.55a(g)(4) and 10 CFR Part 50 Appendix B, Criterion IX, Control of Special Processes, states in part that measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. 10 CFR Part 50.55a(g)(4) also requires compliance with ASME Boiler and Pressure Vessel Code Section XI, IWA-7000, Replacement, which states in part that prior to authorizing the installation of an item to be used in replacement, the Owner shall conduct an evaluation of the suitability of that item.

  ASME Boiler and Pressure Vessel Code Section XI, invokes both ASME Section III, Subsection NB-5320, Radiographic Acceptance Standards, and ASME Section V, Article 2, T-281 Quality of Radiographs. ASME Section III, Subsection NB-5320 does not allow any elongated indication greater than the criteria defined in NB-5320(b). ASME Section V, Article 2, requires that all radiographs be free from artifacts that could mask or be confused with the image of any discontinuity in the area of interest.

  Contrary to the above, during the Unit 1 refueling/steam generator replacement outage in the Fall of 2003: (1) a non-qualified technician was utilized to perform the NDE-UT examination of the hot leg elbows; and (2) non-acceptable linear indications on radiographs of the two hot leg elbows were fortuitously discovered by the DPC Level III after being inappropriately accepted by TFES, TFES Shop ANI, and SGT RT Level II. These two violation examples are considered to be of very low safety significance since all non-acceptable indications were appropriately resolved through additional NDE and/or repairs to meet all code requirements prior to restart of Unit 1. These issues are documented in NCR 1143, Non-conformance Reports Primary Hot Leg Replacement Elbows for Generators A & B, and captured in the licensee’s corrective action program under PIPs O-03-6224, O-03-7044, and O-03-6362.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Batson, Mechanical/Civil Engineering Manager
J. Batton, Oconee Steam Generator Engineer
D. Baxter, Engineering Manager
R. Brown, Emergency Preparedness Manager
N. Constance, Operations Training Manager
C. Curry, Maintenance Manager
T. Curtis, Reactor & Electrical Systems Manager
D. Covar, Training Instructor
C. Eflin, Requalification Supervisor
W. Foster, Safety Assurance Manager
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
L. Nicholson, Regulatory Compliance Manager
R. Repko, Superintendent of Operations
J. Smith, Regulatory Affairs
J. Steeley, Training Supervisor
T. Tucker, NDE Level III Examiner
J. Twiggs, Manager, Radiation Protection
J. Weast, Regulatory Compliance

Licensee Contractors - (Steam Generator Group)

N. Alchaar, Civil Engineer
J. Brackett, Reactor Vessel Head Project
J. Cravens, Welding Implementation
D. Fisher, Design Engineer
R. Griffith, QA Manager
E. Kozlowski, QC Supervisor
B. Millsaps, Maintenance Coordinator
M. Phillips, Corporate Welding Engineer
B. Scarlata, Containment Opening Task Manger
J. Setzer, NDE Level III Examiner
R. Sharpe, Lead Licensing Engineer
F. Suchar, QC Supervisor

NRC

L. Olshan, Project Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000269,270/2003005-01	NCV	Failure to Maintain Flood Protection Barriers (Section 1R06.1)
05000269,270,287/2003005-02	NCV	Failure to Comply with 10 CFR 61.56(b)(2) Waste Characteristic Requirements Involving Liquid Content of Waste Shipped to a Licensed Burial Site for Disposal (Section 2PS2b.(1))
05000269,270,287/2003005-03	NCV	Failure to Comply with 10 CFR 61.55 (a)(2)ii requirements for Classifying Waste Shipped to a Licensed Burial Site for Disposal (Section 2PS2b.(2))
05000269/2003005-04	NCV	Inadequate maintenance procedure for inspection of RCS FME barrier (Section 4OA5.7)
05000269,270,287/2003005-05	NCV	Design Calculation Contains Inaccurate Post LOCA Room Temperatures and a Lack of Assurance that Safety-Related Pumps were Capable of Operating in this Temperature Environment (Section 4OA5.5)

Previous Items Closed

2515/152	TI	Reactor Pressure Vessel Lower Head Penetration Nozzle Inspection (NRC Bulletin 2003-02) - Unit 1 (Section 4OA5.1)
05000269,270,287/2003003-03	URI	Failure to Implement Manufacturer's Recommendations for Replacement of SSF Diesel Coolant Grommets (Section 4OA5.4)
05000269,270,287/2000008-01	URI	Risk Significance of High Temperatures in the Low Pressure Injection and High Pressure Injection Pump Rooms (Section 4OA5.5)

Items Discussed

2515/153	TI	Reactor Containment Sump Blockage (NRC Bulletin 2003-01 - Unit 1 (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

(Section 1R08)

NDE-640, Ultrasonic Examination Using Longitudinal Wave and Shear Wave, Straight Beam Techniques, Rev. 2
NDE-690, Ultrasonic Examination of the High Pressure Injection Nozzle Inner Radius at Oconee Nuclear Station, Rev. 0
NDE-35, Liquid Penetrant Examination, Rev. 19
NDE-600, Ultrasonic Examination of Similar Metal Welds in Ferric and Austenitic Piping, Rev. 15
NDE-B, Training, Qualification and Certification of NDE Personnel
NDE-10, General Radiography Procedure, Rev. 22
Duke Power Oconee Units 1, 2, & 3 Third Interval Inservice Inspection Plan, Rev. 6
UT Calibration Report, UT Base Metal Lamination , UT Pipe Weld Examination, and Ultrasonic Indication Report for Pipe Weld 1-LP-0209-32, Dated 11/6/2003
Radiographic Examination Report/Technique for Pipe Welds 1-LP-0209-31 & 32, Dated 11/7/2003 & 11/6/2003
Weld Process Control for Pipe Welds 1-LP-0209-31 & 32, Dated 11/6/2003 & 11/5/2003
Duke Energy Quality Assurance Technical Services In-Service Inspection Database Management System Oconee 1 Inservice Inspection Listing Internal 3 Outage 5 EOC 20

(Section 2OS1)

Procedures

Standard Health Physics Procedure (SH) SH/0/B/2000/012, Access Controls for High, Extra High, and Very High Radiation Areas, Revision (Rev.) 1
SH/0/B/2000/005, Posting of Radiation Control Zones, Rev. 1
Health Physics Procedure (HP) HP/0/B/1000/054, Radiation Protection Routines, Rev. 37
HP/0/B/1010/004, Selection of Proper Respiratory-Protective Equipment and Respiratory Surveillance Requirements, Rev. 2
HP/0/B/1000/099, Diving Operations, Rev. 2
Radiation Dosimetry Procedure (RD) RD/0/B/4000/15, Nuclear Site Area, Monitoring, Rev. 7
HP/0/B/1000/090 Radiological Protection Requirement For Independent Spent Fuel Storage Installation (ISFSI) - Phase I and II, Rev. 9
HP/0/B/1000/097 Radiological Protection Requirement For Independent Spent Fuel Storage Installation (ISFSI) - Phase III and IV (DSCs 41-84), Rev. 4
Certificate of Compliance for Spent Fuel Storage Cask No. 1004, Amendment No. 4, Effective Date 2/12/02
Independent Spent Fuel Storage Installation (ISFSI) Technical Specifications (TS), 3/4/02

Radiation Work Permits (RWPs)

RWP 2167, Unit 2 Reactor Building - Control Rod Drive Mechanism Nozzle Inspection/ Decontamination and Repair, Rev. 5
RWP 2219, Unit 2 Reactor Building - “A” OTSG Eddy Current/Inspections and Associated Work, Rev. 12
RWP 15, Routine Radiological Surveys, Rev. 14

RWP 1098, U1 Rx Bldg NSM 13093 Passive LPI Cross Connect Mod and Associated Work, Rev. 1
RWP 6101, Radiation Protection Surveillance in Support of U1 SG/RHA Replacement, Rev. 1
RWP 6105, Unit 1 - SG/RHA - Install and Remove Scaffolding in Support of SG/RHA Replacement, Rev. 1
RWP 6177, Unit 1 - RHA - Work on Head Stand, Rev. 0
RWP 6178, Unit 1 - RHA-CRDM Removal/Prep/Inspect/Transport/Install, Rev. 0

Records and Data

Radiation Survey No. 022403-15, Interim Radwaste Building, Room 390, 02/24/2003
Radiation Survey No. 111503-7, Radwaste Building, Room 114, 11/15/2003
Radiation Survey No. 111903-10, Interim Radwaste Annex, 11/19/2003
Radiation Survey No. 081603-2, ISFSIr5.WMF - Independent Spent Fuel Storage Installation, 8/16/2003
Radiation Survey No. 071103-5, ISFSI6.WMF - Horizontal Storage Module (New Type), 7/11/2003
Oconee Nuclear Station Internal Dose Assessments, 11/01/2002 - 10/31/2003

Corrective Action Program Documents

Duke Power Assessment Report: Radiation Protection, Functional Area Evaluation, Rev. 1, 3/28/03
Problem Identification Process (PIP) O-03-05483, Personnel Crossing Radiography boundaries While Radiography was in Progress, 8/29/03
PIP O-03-06381, Workers Crossed EHRA Boundary Without Job Coverage, 10/1/03
PIP O-03-07113, Failure to Comply with Procedure for Posting Areas Requiring Radiological Control, 10/22/03
PIP O-03-07336, Employee Violated RT Boundary, 11/12/03
PIP O-03-02673, Employee Crossed RMA Rope Without Being Logged onto RWP, 5/6/03

(Section 2OS2)

Procedures, Guidance Documents, and Manuals

HP/0/B/1010/004, Selection of Proper Respiratory Protective Equipment and Respiratory Surveillance Requirements, Rev. 23
HP/0/B/1000/016, Radiological Protection Requirements for Steam Generation Work, Rev. 20
SH/0/B/2000/004, Taking, Counting, and Recording Surveys, Rev. 5
SH/0/B/2000/001, Operational Beta Program, Rev. 0
SH/0/B/2000/008, Operational Alpha Program, Rev. 2
SH/0/B/2000/012, Access Controls for High, Extra High, and Very High Radiation Areas, Rev. 1
SH/0/B/2002/003, Declared Pregnant Worker, Rev. 1
System Chemistry Manual (SCM)-9, Optimized Crud Burst Program, Rev. 4
System ALARA Manual, ALARA Planning

ALARA Planning Worksheets (APWs)

NSM-13093 Unit 1 (U1) LPI Passive Cross-Connect Modification for RWP 1098
RCS Piping - Cutting/Machining/Welding/FOSAR in Support of U1 S/G Replacement for RWP 6110
Install/Remove Scaffolding in Support of SG/RHA Replacement for RWP 6105
CRDM Removal/Prep/Inspect/Transport/Install in Support of the U 1 RHA for RWP 6178
RHA Work on Head Stand in Support of the U1 RHA Replacement for RWP 6177

Radiation Work Permits (RWPs)

RWP 1098, U1 Rx Bldg NSM 13093 Passive LPI Cross Connect Mod and Associated Work, Rev. 1
RWP 6101, Radiation Protection Surveillance in Support of U1 SG/RHA Replacement, Rev. 1
RWP 6105, Unit 1 - SG/RHA - Install and Remove Scaffolding in Support of SG/RHA Replacement, Rev. 1
RWP 6109, Remove - Install Main Steam, Feed Water and Aux Feed Water / Install MS, FW and Aux FW Seal Plates / Remove - Install Temporary - Permanent Restraints, Rev. 1
RWP 6110, Unit 1 - SGRP -RCS Pipe Cutting / Machining / Welding, Rev. 1
RWP 6175, Paint Abatement / Coating Activities in Support of the U1 SG/RHA, Rev. 1
RWP 6177, Unit 1 - RHA - Work on Head Stand, Rev. 0
RWP 6178, Unit 1 - RHA-CRDM Removal/Prep/Inspect/Transport/Install, Rev. 0

Records and Data

Oconee CRUD Team Charter, February 25, 2003
Oconee CRUD Team Meeting Minutes for: February 25, 2003, March 10, 2003, April 14, 2003, and July 14, 2003
Active Hot Spot Data Base (as of November 17, 2003)
Year 2003 Cleared Hot Spots (as of November 17, 2003)
Job Worker Lists for 2003 for RWPs 6100, 6101, 6102, 6103, 6104, 6105, 6106, 6107, 6108, 6109, 6110, 6111, 6112, 6113, 6114, 6115, 6116, 6117, 6118, 6175, 6176, 6177, 6178, and 6179

Corrective Action Program Documents

Duke Power Assessment Report: Radiation Protection, Functional Area Evaluation, Rev. 1, 3/28/03
PIP O-03-07524, RWP 6103, Miscellaneous Support Activities, Exceeded 125% of Its Outage Total Dose Goal
PIP O-03-05311, Problems Encountered While Installing Lids on High Integrity Containers of Dry Active Waste Resulted in More Exposure than Anticipated, 8/21/03
PIP O-03-05827, Lead Shielding Did Not Cover Area Identified by RP's Request for Temporary Shielding, 9/18/03
PIP O-03-03480, Premature Removal of Shielding in L/D Cooler Room Resulted in Increased Dose to Workers, 5/29/03
PIP O-03-02833, Steam Generator Maintenance HEPA Units for Lower Channel Head Downdraft Tables were Unplugged, 5/9/03
PIP O-03-02878, 3LP-20 Work Stopped by RP Due to Unexpected Dose Rates, 5/10/03

(Section 2PS2)

Procedures, Guidance Documents

SH/O/B/2004/002, Preparation and Shipment of Radioactive Waste, Rev. 2
MP/O/A/170/015, CNS 14-215-H-Handling Procedure, Rev. 18
Oconee Nuclear Station, 10CFR61 and Waste Form Implementation Program, 06/20/2002

Records and Data

Oconee Processing and Disposal Status, 2002 and 2003
Individual Shipping Qualification Certificates
Outgoing Radioactive Material Shipping Logs for 2001 - 2002
Duke Engineering, 10 CFR Part 50/61 Analysis Reports for 2001-2003
Transportation Shipping Records ONS-03-2039 (9/30/03), ONS-03-2060 (11/3/03), ONS-03-2001 (1/8/03), and ONS-03-2003 (5/20/03)

Corrective Action Program Documents

Oconee Radiation Protection Functional Area Evaluation, GO-02-15, February 04-14, 2002
PIP O-03-07144, Accident involving Laundry Shipment, Container Not Breached, 11/03/03
PIP O-03-00224, Under Classification of Shipment, 01/14/03
PIP O-02-07070, Shipment of Radioactive Material, 12/11/02
PIP O-02-05873, Shipment of Radioactive Material, 09/19/03
PIP O-02-07102, Shipment of Radioactive Material, 12/12/02

Annual Reports

Oconee Radioactive Effluent Release Report, January 2002-December 2002 date April 24, 2003

(Section 4OA1)

Procedures

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

Records

NRC Performance Indicator Data Review for December 2002, dated 1/13/2003
NRC Performance Indicator Data Review for July 2003, dated 8/12/2003
NRC Performance Indicator Data Review for August 2003, dated 9/10/2003
NRC Performance Indicator Data Review for September 2003, dated 10/08/2003

(Section 4OA5.8)

Modification Package ON-13086, Containment Opening, Part AS9, Revision 3
Work Package 13555, Unit 1 Construction Opening Steel Liner Installation
Work Package 13550, Unit 1 Construction Opening Concrete Installation

SGT Certificate of Engineering Calculation 0CS-8420, SGRP and RVHRP Code Reconciliation (Other than Reactor Coolant System)
Duke Specification No. OS 139-4, Specification For Field Welding of Reactor Building Liner Plate By Manual Metal-Arc Process
Concrete Reinforcing Bar Splicer Qualification Records for Seven SGT Rebar Splicers
Wiss, Janney, Elstner, Associates, Inc. Letter dated September 15, 2003, documenting static tensile tests qualification of rebar splicing system
SGT Specification SGRP-SPEC-C-04, Reactor Building-SGRP Construction Opening Reinforcing Steel, Revision 3
Procedure BPI-GRIP Systems Splicing Manual and Operating Instructions, Revision 10/18/01
NDE Examiner Qualification Records for the following SGT NDE Examiners: 3 Level II (VT, PT, and MT) Examiners, including IWE/IWL Endorsement; 2 Level III (VT, PT, and MT) Examiners, and 1 Level III (RT) Examiner.
Radiographic Examination Reports and Film for Liner Plate Welds LP-1, LP-2, LP-3, and LP-4
Sample of Magnetic Particle Examination Reports for Liner Plate Welds LP-1, LP-2, LP-3, and LP-4
SGT Quality Execution Procedure 12.06, Radiographic Examination (ASME) , Revision 1
SGT Welding Procedure Specification GT-SM/1.1-2, Revision 3
SGT Procedure Qualification Record GT-SM/1.1-Q6
SGT Procedure Qualification Record UE-47, Revision 3
Welder Qualification Records for Chicago Bridge and Iron (CB&I) Welders 004, 010, 011, 015, and 016
Certified Material Test Reports for 3/32" E7018 - Lot 2L216C02, and 1/8" E7018 - Lot 4D215A04 Welding Electrodes
Specification No. SGRP-SPEC-C-003, Reactor Building - SGRP Construction Opening and Concrete Placement, Rev. 3, dated 10/28/03
Specification No. SGRP-SPEC-C-004, Reactor Building - SGRP Construction Opening Reinforcing steel, Rev. 3, dated 10/28/03
Specification No. SGRP-SPEC-C-002, Reactor Building - SGRP Construction Opening Tendon Work, Rev. 6, dated 9/10/03
Work Plan 13550, Unit 1 Construction Opening Concrete Installation
Work Plan 13551, Unit 1 Construction Opening Tendon Installation
Quality Execution Procedure QEP 11.03, Concrete and Grout Placement, Rev. 0E1, dated 9/10/02
Quality Execution Procedure QEP 12.02, Conduct and Control of Inspection and Surveillance Activities, Rev 2/AFU, dated 10/4/03
Duke Procedure MP/0/A/3005/010, Containment Structural Inspection, Rev 1
Drawing number SK-13086AS9-008, Rev 3, Containment Opening Unit 1, Notes, References, and Schedules
AWS WPS for Reinforcing Steel, SM-RS-1
National Ready Mixed Concrete Association (NRMCA) certificate for batch plant, truck mix
National Ready Mixed Concrete Association (NRMCA) certificates for concrete truck mixers, Zupan & Smith concrete truck numbers 63, 64, 70, 72, 75, & 85
Records for calibration of concrete batch plant cement and aggregate scales, and batch plant water meter
Concrete mixer uniformity (ASTM C-94) tests performed on truck numbers 70 & 85
Concrete mix design data
Result of testing performed on concrete materials: Type III cement (ASTM C-150), CTS Komponent admixture, air entraining admixture MBEA lot number 1371740N3, high range water reducer Glenium 3030 lot numbers 1372067N3 and 1312354T3 , fine aggregate (ASTM

C-33), number 67 coarse aggregate (ASTM C-33), and batch plant water
Concrete placement records which included the pre-pour check list, the concrete pour card, the results of testing performed on the plastic concrete (slump, air content, temperature and unit weight), and the results of unconfined compression tests performed on concrete test cylinders at 4 days
SGT nonconformance report numbers NCR 1151, Damage to Liner Plate
SGT NCR 1169, Reinforcing Steel Welder Qualification Records
SGT NCR 1195, Reinforcing Steel Splice Backing Bars not Removed
SGT NCR 1201, Indications on Liner Plate
SGT NCR 1207, Concrete Coverage Over Reinforcing Steel
SGT NCR 1208, Some Concrete Placed in Construction Opening With Density Slightly Below Specification Requirements
SGT NCR 1209, Concrete Uniformity Testing of Concrete Trucks
SGT NCR 1214, Quantity of High Range Water Reducer Used in Concrete Exceeded Specification Limits
Duke QA Supplier Surveillance Report VS-03-093 (7/1 - 31/03), Tendon Fabrication Records
PSC tendon fabrication records for replacement tendon 45V3, including tendon wire physical and chemical materials testing data, and anchor head material testing data, heat treatment data, and fabrication records
PSC records for pre-post tensioning calibration of stressing rams and pressure gauges

(Sections 4OA5.10 and 11)

QEP 10.04 General Housekeeping, Foreign Material Exclusion (FME) and Cleanliness Requirements
QEP-10.05, Rigging and Handling, Rev. 1
QEP 11.01, Work Packages, Rev. 1E1
QEP 12.03, Visual Weld Examination, Rev. 2
QEP 12.04, Liquid Penetrant Examination, Rev.
QEP 12.05, Magnetic Particle Examination, Rev. 2
QEP 12.06, Radiographic Examination, Rev.
QEP 12.13, Reference and Layout of ASME Section XI Welds, Rev. 1
QEP 12.14, Ultrasonic Examination, Rev.
QEP 12.16, Ultrasonic Examination of Ferritic Piping Welds (ASME Section XI), Rev. 0
QEP 15.01, Identification and Control of Deviations, Rev. 3
QEP 20.01, Control and Documentation of Welding, Rev. 1E2
QEP 20.02, Welding Procedure Specifications, Rev. 0E1
QEP 20.03, ASME General Welding Requirements, Rev. 3
QEP 20.04, Welder Performance Qualification, Rev.
QEP 20.05, Welding Material Control, Rev.
QEP 20.06, Preheat and Post Weld Heat Treatment, Rev.
QEP 20.07, Weld and Base Metal Repairs, Rev.
Modification Package ON-13086 AM1, Main Steam Piping, Rev. OE3
Modification Package ON-13086 AM2, Feedwater Piping, Rev. OE3
Modification Package ON-13086 AM3, Emergency Feedwater Piping, Rev. 1
Modification Package ON-13086 AM4, Steam Generator Replacement, Rev. OE3
Modification Package ON-13086 AM9, Auxilary Crane, Rev. 1
Modification Package ON-53086 AS1, Steam Generator Transport, Rev. 1
Modification Package ON-13086 AS6, Steam Generator Rigging and Handling, Rev. 1
WPS GT/1.1, Rev. 0E1, ASME Section IX Welding Procedure Specification

WPS GT/3.1, Rev. 0E1, ASME Section IX Welding Procedure Specification
Work Package 13065A - RCS Machining / Welding SG1A
Work Package 13065B - RCS Machining / Welding SG1B
Work Package 11032, Installation of Unit 1 Outside Lift System (OLS)
Bishop Lifting Products, Certificate of Test, TPXC40000 x 34'5" Twin Path Extra Cordura Covermax
Bishop Lifting Products, Certificate of Conformance, Gator Laid Sling, Crosby Body Shackle
W/O 512540, Certificate of Test and Examinations of Chains, Rings, Hooks, Shackles, Swivels and Pulley Blocks
W/O 524821, Certificate of Test and Examinations of Chains, Rings, Hooks, Shackles, Swivels and Pulley Blocks
MEI-80255-D10, Mammoet Test Procedure, Outside Lifting Structure, Rev. 1
MEI-80255-D08, Mammoet Test Procedure, Temporary Lifting Device, Rev. 2
Preservice Eddy Current Inspection, BWC-TR-2003-010, Rev. 0
Preservice Eddy Current Inspection, BWC-TR-2003-011 Rev. 0.
Duke Power Company Assessment Report GO-03-52, Procurement Process Effectiveness for ASME Code Items
DPC Spec. No. OSS-0210.00-00.0001 SGRP-SPEC-M-002, Rev. 1, Reactor Coolant System Hot Leg Elbows.
Memo McRainey to Wilkerson, 11/5/03, Accurate and timely documentation of Weld Cards
Memo McRainey to Wilkerson, 11/3/03, Documentation of Welder Identification and Welding Filler Material Identification on Weld Cards.
PIPs O-03-00338, 00339, 00340, 00341, 04004, 04096, 06994, 06191, 06224, 06362, 07044, 07060, 07307, 07308, 07309, 07310, 07333, 07158, 07159, 07118, 01081, 07224, 07667
NCR 1165, 1143, 1105
DR TW/9-25-03/01
SGT CAR-03-04

LIST OF ACRONYMS

ADAMS	-	Agency wide Documents Access and Management System
ALARA	-	As Low As Reasonably Achievable
ASME	-	American Society of Mechanical Engineers
ASTM	-	American Society for Testing and Materials
BMV	-	Bare Metal Visual
BTP	-	Branch Technical Position
CFR	-	Code of Federal Regulations
CRs	-	Condition Reports
DAW	-	Dry Active Waste
DEC	-	Duke Energy Corporation
EFW	-	Emergency Feedwater
EOP	-	Emergency Operating Procedure
FME	-	Foreign Material Exclusion
HEPA	-	High Efficiency Particulate Air
HP	-	Health Physics
HPI	-	High Pressure Injection
HPT	-	Health Physics Technician
HRA	-	High Radiation Area
IP	-	Inspection Procedure
IR	-	Inspection Report

ISFSI	-	Independent Spent Fuel Storage Installation
KHU	-	Keowee Hydro Unit
LPI	-	Low Pressure Injection
LPSW	-	Low Pressure Service Water
NCV	-	Non-Cited Violation
NDE	-	Nondestructive Examination
NRC	-	Nuclear Regulatory Commission
NRMCA	-	National Ready Mixed Concrete Association
NRR	-	Nuclear Reactor Regulation
ONEI	-	Oconee Nuclear Engineering Instruction
ONS	-	Oconee Nuclear Station
OTSG	-	Once-Through Steam Generator
PI	-	Performance Indicator
PIP	-	Problem Investigation Process report
PT	-	Performance Test
PMT	-	Post-Maintenance Testing
QC	-	Quality Control
RADWASTE	-	Radioactive Waste
RBCU	-	Reactor Building Cooling Unit
RBSE	-	Reactor Building Emergency Sump
RBS	-	Reactor Building Spray
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RG	-	Regulatory Guide
RP	-	Radiation Protection
RTP	-	Rated Thermal Power
RV	-	Reactor Vessel
RWP	-	Radiation Work Permit
SDP	-	Significance Determination Process
SG	-	Steam Generator
SGR	-	Steam Generator Replacement
SRA	-	Senior Reactor Analyst
SSC	-	Structure, System and Component
SSF	-	Standby Shutdown Facility
TDEFW	-	Turbine Driven Emergency Feedwater
TI	-	Temporary Instruction
TS	-	Technical Specification
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item