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

July 29, 2008

EA 08-207

Mr. Jeffrey B. Archie
Vice President
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P.O. Box 88
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000395/2008003 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Archie:

On June 30, 2008, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Virgil C. Summer Nuclear Station. The enclosed integrated inspection report documents the inspection results, which were discussed with you and other members of your staff on July 10, 2008.

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

This report documents one self-revealing finding of very low safety significance (Green). This finding was determined not to involve a violation of NRC requirements. In addition, the inspectors reviewed the events associated with heavy load lifts and industry uncertainty regarding the licensing bases for handling of reactor vessel heads, which led to the issuance of EGM 07-006, "Enforcement Discretion for Heavy Load Handling Activities," on September 28, 2007. The Nuclear Energy Institute (NEI) has informed NRC of industry approval of a formal initiative that specifies actions each plant will take to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices. NRC inspection of Virgil C. Summer heavy load lift licensing bases identified a violation of requirements to update the final safety analysis report pursuant to 10 CFR 50.71(e) to reflect aspects of heavy load lifts involving the reactor vessel head and include information from a reactor vessel head drop analysis. The NRC staff believes implementation of the NEI initiative will resolve uncertainty in the licensing bases for heavy load handling, and enforcement discretion related to the uncertain aspects of the licensing basis is appropriate during the implementation of the initiative. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violation.

In accordance with 10 CFR 2.390 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/

Mark A. Bates, Acting Chief
Reactor Projects Branch 5
Division of Reactor Projects

Docket No.: 50-395
License No.: NPF-12

Enclosure: NRC Integrated Inspection Report 05000395/2008003
w/Attachment: Supplemental Information

cc w/encl: (See next page)

implementation of the initiative. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violation.

In accordance with 10 CFR 2.390 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/

Mark A. Bates, Acting Chief
Reactor Projects Branch 5
Division of Reactor Projects

Docket No.: 50-395
License No.: NPF-12

Enclosure: NRC Integrated Inspection Report 05000395/2008003
w/Attachment: Supplemental Information

cc w/encl: (See next page)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE
ADAMS: ☐ Yes ACCESSION NUMBER:

OFFICE	RII:DRP	RII:DRP	RII:DRS	RII:DRS	RII:DRS	RII:DRS	RII:DRP
SIGNATURE	/RA By E-Mail/	/RA By E-Mail/	/RA By BBonser/	/RA By BBonser/	/RA/	/RA/	/RA/
NAME	JZeiler	JPolickoski	RHamilton	ANielsen	RCarrion	AVargasMendez	JDodson
DATE	7/ /2008	7/ /2008	7/ /2008	7/ /2008	7/ /2008	7/ /2008	7/ /2008
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\RPB5\SUMMER\REPORTS\2008\SUM 08-03\SUM 08-03 REV3.DOC

cc w/encl:
R. J. White
Nuclear Coordinator
S.C. Public Service Authority Mail Code 802
Electronic Mail Distribution

Kathryn M. Sutton, Esq.
Morgan, Lewis & Bockius LLP
Electronic Mail Distribution

Susan Jenkins
Director
Division of Radioactive Waste Mgmt.
S.C. Department of Health and
Environmental Control
Electronic Mail Distribution

R. Mike Gandy
Division of Radioactive Waste Mgmt.
S.C. Department of Health and
Environmental Control
Electronic Mail Distribution

Bruce L. Thompson
Manager
Nuclear Licensing (Mail Code 830)
South Carolina Electric & Gas Company
Electronic Mail Distribution

Robert M. Fowlkes
General Manager
Engineering Services
South Carolina Electric & Gas Company
Electronic Mail Distribution

Thomas D. Gatlin
General Manager
Nuclear Plant Operations (Mail Code 303)
South Carolina Electric & Gas Company
Electronic Mail Distribution

David A. Lavigne
General Manager
Organization Development
South Carolina Electric & Gas Company
Electronic Mail Distribution

Letter to Jeffrey B. Archie from Mark A. Bates, July 29, 2008

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000395/2008003

Distribution w/encl:
C. Evans, RII EICS
L. Slack, RII EICS
OE Mail
RIDSNRRDIRS
PUBLIC
R. Martin, NRR

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-395
License No.:	NPF-12
Report No.:	05000395/2008003
Licensee:	South Carolina Electric & Gas (SCE&G) Company
Facility:	Virgil C. Summer Nuclear Station
Location:	P. O. Box 88 Jenkinsville, SC 29065
Dates:	April 1, 2008 – June 30, 2008
Inspectors:	J. Zeiler, Senior Resident Inspector J. Polickoski, Resident Inspector R. Carrion, Senior Reactor Inspector (Sections 1R08, 4OA5.2) R. Hamilton, Senior Health Physicist (Sections 2OS2, 2PS1, 4OA1.2) A. Nielsen, Health Physicist (Sections 2OS1, 2PS2) A. Vargas-Mendez, Reactor Inspector (Sections 1R08, 4OA5.2)
Approved by:	Mark A. Bates, Acting Chief Reactor Projects Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000395/2008-003; 04/01/2008 - 06/30/2008; Virgil C. Summer Nuclear Station; Identification and Resolution of Problems.

The report covered a three-month period of inspection by resident inspectors and Regional based reactor and health physics inspectors. One Green self-revealing finding was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. Self-Revealing Findings

- Green. A Green self-revealing finding was identified for the failure to implement effective and timely corrective actions to prevent failure of a main feedwater regulating valve IFV000498 that resulted in a reactor trip. This valve failed due to previously identified pneumatic positioner pilot valve malfunction caused by either pilot valve stem fretting and/or foreign material intrusion from various internal air supply sources. All three loop feedwater regulating valve positioners and air supply components subject to potential sources of contamination were replaced prior to startup from the reactor trip. During Refueling Outage 17, modifications were completed to reduce vibration induced wear of control air system components and improve air quality to the positioners until the current positioner models can be replaced with a new design. This finding was entered into the licensee's corrective action program as Condition Report 08-00292.

  This finding is greater than minor because it is associated with the Initiating Event Cornerstone attribute of equipment performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the aspect of appropriate and timely corrective action in the cross-cutting area of Problem Identification and Resolution (Corrective Action component) because actions to address previously identified feedwater regulating valve positioner pilot valve fretting and foreign material intrusion were not implemented in a timely manner (P.1.d). (Section 4OA2.3)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period at 100 percent rated thermal power (RTP). On April 23, 2008, power was reduced to 85 percent RTP to conduct scheduled lift setpoint testing of the main steam line code safety valves. The unit remained at 85 percent RTP until April 25, when a planned shutdown was commenced to implement the seventeenth refueling outage (RF-17). Following outage related work activities reactor criticality was achieved on June 13. The main turbine was placed on-line June 14 and the unit was returned to full RTP operation on June 17. The unit remained at or near full RTP for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

Seasonal Weather Susceptibilities

a. Inspection Scope

The inspectors performed one adverse weather inspection for readiness of hot weather. The inspectors verified the licensee had implemented applicable sections of operations administrative procedure (OAP) -109.1, "Guidelines for Severe Weather." The inspectors walked down accessible areas of risk-significant equipment, including the emergency diesel generator (EDG) rooms and the control building reactor protection system and inverter room to verify the reliability of the heating, ventilation and air-conditioning (HVAC) systems to provide adequate cooling for the associated equipment. Also, the inspectors reviewed licensee plant computer data associated with certain large pump motor stator and bearing temperatures to verify the values were within expected operational ranges to prevent any challenge to equipment operation. The inspectors reviewed the licensee's corrective action program (CAP) database to verify that high temperature weather related problems were being identified at the appropriate level, entered into the CAP, and appropriately resolved.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors conducted three partial equipment alignment walkdowns (listed below) to evaluate the operability of selected redundant trains or backup systems with the other train or system inoperable or out of service (OOS). Correct alignment and operating conditions were determined from the applicable portions of drawings, system operating procedures (SOPs), final safety analysis report (FSAR), and technical specifications (TS). The inspections included review of outstanding maintenance work orders (WOs) and related condition reports (CRs) to verify that the licensee had properly identified and resolved equipment alignment problems that could lead to the initiation of an event or impact mitigating system availability. Documents reviewed are listed in the Attachment to this report.

- "B" component cooling water (CCW) system and "C" CCW pump (while "A" CCW pump was OOS for planned maintenance to replace pump seals);
- "B" residual heat removal (RHR) system (while "A" RHR was OOS for planned preventive maintenance); and,
- "B" EDG system (while "A" EDG was OOS for planned refueling outage maintenance).

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a detailed review and walkdown of the RHR system and related piping to identify any discrepancies between the current operating system equipment lineup and the designed lineup. This walkdown included accessible areas outside the containment and inside the containment building once the licensee completed system lineups to support plant restart. In addition, the inspectors reviewed completed surveillance procedures, outstanding WOs, system health reports, and related CRs to verify that the licensee had properly identified and resolved equipment problems that could affect the availability and operability of the system. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified

1R05 Fire Protection

Fire Protection - Tours

a. Inspection Scope

The inspectors reviewed recent CRs, WOs, and impairments associated with the fire protection system. The inspectors reviewed surveillance activities to determine whether they supported the operability and availability of the fire protection system. The inspectors assessed the material condition of the active and passive fire protection systems and features and observed the control of transient combustibles and ignition sources. The inspectors conducted routine inspections of the following nine areas (respective fire zones also noted):

- Control Room (fire zone CB-17.1);
- Relay room including solid state protection system instrumentation and inverters (fire zones CB-6, 10, and 12);
- "A" and "B" EDG rooms (fire zones DG 1.1/1.2 and DG 2.1/2.2);
- "A," "B," and "C" HVAC chill water system rooms (fire zones IB-7.2, 9, and 23.1);
- "A," "B," and "C" battery and charger rooms (fire zones IB-2, 3, 4, 5, and 6);
- Service water (SW) pump house (fire zone SWPH-1, 3, 4, 5.1, and 5.2);
- RHR pump rooms (fire zones AB-1.1, 1.2, and 1.3);
- Control building 482' elevation (fire zones CB-22, 23); and,
- Auxiliary building switchgear room 412' elevation (1DA2Y) (fire zone AB-1.10).

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

Internal Flooding

a. Inspection Scope

The inspectors reviewed and walked down one area (the control building relay room) regarding internal flood protection features and equipment to determine consistency with design requirements, FSAR, and flood analysis documents. Risk significant structures, systems, and components (SSCs) in this area included the solid state protection system and vital inverters. The inspectors reviewed the licensee's CAP database to verify that internal flood protection problems were being identified at the appropriate level, entered into the CAP, and appropriately resolved.

b. Findings

No findings of significance were identified.

1R08 In-Service Inspection (ISI) Activities (71111.08P)

.1 In-Service Inspection Activities Other Than Steam Generator Tube Inspections, PWR Vessel Upper Head Penetration Inspections, and Boric Acid Corrosion Control Program

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping boundaries during the Unit 1 Spring 2008 refueling outage (RF-17). The inspectors' activities consisted of an on-site review of nondestructive examination (NDE) and welding activities to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Sections XI (Code of record for the third 10-year ISI interval was 1998 Edition through 2000 Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI acceptance standards.

The inspectors reviewed NDE activities including examination procedures, NDE reports, equipment and consumable certification records, personnel qualification records, calibration reports, and calibration block fabrication drawings (as applicable) for the following examinations:

Ultrasonic Testing (UT):

- Pressurizer safety/relief nozzle 4A weld overlay as part of Alloy 600 mitigation, Reactor Coolant System (ASME Class 1);
- Pressurizer safety/relief nozzle 4B weld overlay as part of Alloy 600 mitigation, Reactor Coolant System (ASME Class 1);
- Pressurizer Power-Operated Relief Valve (PORV) nozzle weld overlay as part of Alloy 600 mitigation, Reactor Coolant System (ASME Class 1);
- RHR System, weld # 2-2501-53, 14" elbow to pipe weld (ASME Class 2);
- RHR System, weld # 2-2501-54, 14" pipe to elbow weld (ASME Class 2);
- RHR System, weld # 2-2525-7, 10" pipe to elbow weld (ASME Class 2);
- RHR System, weld # 2-2525-8, 10" elbow to pipe weld (ASME Class 2); and,
- RHR System, weld # 2-2525-9, 10" pipe to elbow weld (ASME Class 2).

In addition, the inspectors reviewed the In-service Examination Evaluation for the previous Refueling Outage (RF-16) for relevant indications addressed by the licensee. One rejectable indication was identified by the licensee via a liquid penetrant (PT) examination on a reinforcing plate-to-pipe fillet weld on the "B" RHR heat exchanger outlet nozzle (XHE0005B-RH). The licensee generated CR-06-03321 to address and disposition the issue.

The inspectors review of welding activities included a sample of in-process welding activities for ASME Class 1 piping to evaluate compliance with procedures and the ASME Code. The inspectors directly observed part of the welding process and verified welding machine settings for the welding activity described below. The inspectors also reviewed weld process control reports, welding procedures, procedure qualification records, certified material test reports for filler material, and welder qualification records.

- Weld Overlay on pressurizer safety/relief nozzle 4C as part of Alloy 600 mitigation, Reactor Coolant System (ASME Class 1).

b. Findings

No findings of significance were identified.

.2 PWR Vessel Upper Head Penetration (VUHP) Inspection Activities

a. Inspection Scope

The bare metal visual examination reactor vessel upper head penetrations required by Order EA 03-009 was completed during the previous refueling outage. However, the inspectors reviewed RF-17 results of the visual examination (VT-2) credited for identifying potential boric acid leaks from pressure-retaining components above the reactor pressure vessel head, as required by Order EA 03-009. The inspectors also reviewed the effective degradation years (EDY) calculation performed by the licensee.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and the results of the licensee's Mode 3 containment walkdown inspections performed during the Spring 2008 outage, including generated CRs and their subsequent engineering evaluations.

The inspectors conducted an independent walkdown of the reactor building to evaluate compliance with licensee's BACC program requirements and verify that degraded or non-conforming conditions, such as boric acid leaks identified during the Mode 3 containment walkdown, were properly identified and corrected in accordance with the licensee's BACC and CAP.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

No steam generator tube inspection activities were conducted during this outage.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

Resident Inspector Quarterly Review

a. Inspection Scope

On June 24, 2008, the inspectors observed performance of senior reactor operators and reactor operators on the plant simulator during licensed operator requalification training. The scenario (LOR-SA-015B) involved a feedwater pump trip and feedwater pump master controller failure from 100 percent RTP followed by a loss of all secondary heat sink condition. The inspectors assessed overall crew performance, communications, oversight of supervision, and the evaluators' critique. The inspectors verified that any significant training issues were appropriately captured in the licensee's CAP.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors evaluated two equipment issues described in the CRs listed below to verify the licensee's effectiveness of the corresponding preventive or corrective maintenance associated with SSCs. The inspectors reviewed maintenance rule (MR) implementation to verify that component and equipment failures were identified, entered, and scoped within the MR program. Selected SSCs were reviewed to verify proper categorization and classification in accordance with 10 CFR 50.65. The inspectors examined the licensee's 10 CFR 50.65 (a)(1) corrective action plans to determine if the licensee was identifying issues related to the MR at an appropriate threshold and that corrective actions were established and effective. The inspectors' review also evaluated if maintenance preventable functional failures (MPFF) or other MR findings existed that the licensee had not identified. The inspectors reviewed the licensee's controlling procedures, i.e., engineering services procedure (ES)-514, "Maintenance Rule Implementation," and the Virgil C. Summer "Important To Maintenance Rule System Function and Performance Criteria Analysis," to verify consistency with the MR requirements.

- CR-08-00292, Malfunction of "C" steam generator feedwater flow control valve IFV00498 causes steam generator water level transient and manual reactor trip; and,
- CR-08-01191, "B" EDG low oil pressure alarm relay failure to actuate during calibration.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated, as appropriate, for the five selected work activities listed below: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforeseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and, (4) that emergent work problems were adequately identified and resolved. The inspectors evaluated the licensee's work prioritization and risk characterization to determine, as appropriate, whether necessary steps were properly planned, controlled, and executed for the planned and emergent work activities listed below:

- Work Week 2008-15: risk assessment for scheduled maintenance and/or testing on “A” RHR pump, “A” CCW pump, “A” control room ventilation, “A” EDG turbine-driven emergency feedwater pump, “C” SW pump and “C” chiller;
- Work Week 2008-16: risk assessment for scheduled maintenance and/or testing on safety-related snubbers, “A” motor-driven emergency feedwater pump, “A” centrifugal charging pump, electric fire service pump, “A” EDG, and emergent maintenance on “C” feedwater control valve;
- Work Week 2008-17: risk assessment for scheduled maintenance and/or testing on the turbine-driven emergency feedwater pump, “B” control room ventilation, safety-related snubbers, alternate AC source line relocation, “B” CCW pump, “C” main steam safety valves, and down power to 85 percent;
- Review of refueling outage shutdown risk and contingency plans for RCS inventory at nine inches below the reactor vessel flange prior to core offload with reactor coolant pump seal work ongoing; and,
- Review of refueling outage shutdown risk and contingency plans for reactor core offload, single train of offsite power source and engineered safety features equipment available, and alternate power to the “B” spent fuel pool pump.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed five operability evaluations affecting risk significant mitigating systems to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether operability was properly justified and the subject component or system remained available, such that no unrecognized increase in risk occurred; (3) whether other existing degraded conditions were considered; (4) that the licensee considered other degraded conditions and their impact on compensatory measures for the condition being evaluated; and, (5) the impact on TS limiting conditions for operations and the risk significance in accordance with the Significance Determination Process (SDP). Also, the inspectors verified that the operability evaluations were performed in accordance with station administrative procedure (SAP)-209, “Operability Determination Process,” and SAP-999, “Corrective Action Program.”

- CR-08-01234, “C” feedwater control valve position alarms received;
- CR-08-01393, Reactor building alternate inlet purge supply isolation valve XVG06056-HR exceeded its maximum limiting stroke time;
- CR-08-01443, “B” train RHR piping resting on steel wall cutout opening between adjacent room;
- CR-08-01526, “C” main steam safety relief valve XVS02806K-MS lift setpoint exceeded TS limit; and,
- CR-08-02304, “A” EDG voltage limit of 7770 volts exceeded during loss of offiste power testing.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

a. Inspection Scope

The inspectors evaluated one equipment change that was considered a temporary modification and two engineering change request (ECR) packages for permanent modifications to evaluate the changes for adverse effects on system availability, reliability, and functional capability. Documents reviewed included procedures, engineering calculations, modification design and implementation packages, WOs, site drawings, corrective action documents, applicable sections of the FSAR, supporting analyses, TS, and design basis information. The inspectors witnessed aspects of each modification implementation and observed aspects of post-modification testing of both permanent modifications to verify adequate testing of the changes. Documents reviewed are listed in the Attachment to this report.

The temporary modification reviewed included the installation of temporary alternate power to the "B" spent fuel pool cooling pump (XPP0032B) while supply power from its normal safety-related electrical bus was out of service. The inspectors evaluated the change documents and associated 10 CFR 50.59 reviews against the system design basis documentation and FSAR to verify that the changes did not adversely affect the safety function of safety systems.

The two permanent modifications and the associated attributes reviewed are as follows:

ECR 50567, Addition of service water vacuum relief valves and replacement of reactor building cooling unit (RBCU) service water return valves XVG03107A/B-SW

- Licensing Basis
- Failure Modes
- Energy Needs
- Control Signals
- Timing
- Plant Document Updating
- Operations
- Flowpaths
- Implementation
- Post Modification Testing

ECR 50466, "A" EDG governor replacement

- Licensing Basis
- Failure Modes
- Energy Needs
- Control Signals

- Timing
- Plant Document Updating
- Operations
- Flow paths
- Implementation
- Post Modification Testing

The inspectors also reviewed selected CRs associated with modifications to confirm that problems were identified at an appropriate threshold, were entered into the CAP, and appropriate corrective actions had been initiated.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

For the six maintenance activities listed below, the inspectors reviewed the associated post-maintenance testing (PMT) procedures and either witnessed the testing and/or reviewed test records 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) test 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 verified that these activities were performed in accordance with general test procedure (GTP)-214, "Post Maintenance Testing Guideline."

- PMT for "B" EDG quarterly preventive maintenance (WOs 0525493, 0606511, 0711404, 0711412, 0715589, 0716996, and 0802012);
- PMT for "B" RHR pump and valve preventive maintenance (WOs 0610131, 0712407, 0714084, and 0716987);
- PMT for "C" feedwater control valve gag installation and replacement of the positioner pilot valve, air filter, and air tubing (WO 0803916);
- PMT for RBCU service water return valves XVG03107A/B-SW following implementation of ECR 50567;
- PMT for Parr Hydro Station alternate AC cable replacement using surveillance test procedure (STP)-125.021, "Periodic Testing of the Alternate AC Power Supply," and STP-125.022, "Parr Hydro Timed Blackout Recovery Test;" and,
- PMT for digital rod position indication repair following surveillance test failure (WO 0805565).

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Refueling Outage RF-17

a. Inspection Scope

On April 25, 2008, the unit was shutdown to commence RF-17. The 50 day outage was completed on June 14. The inspectors used inspection procedure 71111.20, "Refueling and Outage Activities," to complete the inspections described below.

Prior to and during the outage, the inspectors reviewed the licensee's outage risk assessments and controls for the outage schedule to verify that the licensee had appropriately considered risk, industry experience and previous site specific problems, and to confirm that the licensee had mitigation/response strategies for losses of any key safety functions.

In the area of licensee control of outage activities, the inspectors reviewed equipment removed from service to verify that defense-in-depth was maintained in accordance with applicable TS and that configuration changes due to emergent work and unexpected conditions were controlled in accordance with the outage schedule and risk control plan.

The inspectors reviewed selected components which were removed from service to verify that tagouts were properly installed and that associated equipment was appropriately configured to support the function of the clearance.

During the outage, the inspectors reviewed and/or observed the following:

- RCS pressure, level, and temperature instruments to verify that those instruments were installed and configured to provide accurate indication;
- The status and configuration of electrical systems to verify that those systems met TS requirements and the licensee's outage risk control plan. The inspectors also evaluated if switchyard activities were controlled commensurate with their risk significance and if they were consistent with the licensee's outage risk control assessment assumptions;
- Spent fuel pool operations to verify that outage work was not impacting the ability of the operations staff to operate the spent fuel pool cooling system during and after core offload. The inspectors also reviewed the licensee's calculation results of spent fuel pool and reactor vessel heatup rates in case of a potential loss of cooling event;
- The control of containment penetrations and containment entries to verify that the licensee controlled those penetrations and activities in accordance with the appropriate TS and could achieve/maintain containment closure for required conditions; and,

- All accessible areas inside the reactor building prior to reactor startup to verify that debris had not been left which could affect the performance of the containment sumps.

The inspectors reviewed the following activities for conformance to applicable procedural and TS requirements:

- Plant shutdown activities;
- Decay heat removal system operations;
- Inventory controls and measures to provide alternate means for inventory addition;
- Reactivity controls;
- Reactor vessel defueling and refueling operations; and,
- Reactor heatup, mode changes, initial criticality, startup and power ascension activities.

The inspectors reviewed various problems that arose during the outage to verify that the licensee was identifying problems related to outage activities at an appropriate threshold and was entering them in the CAP.

b. Findings

No findings of significance were identified.

.2 Control of Heavy Loads

a. Inspection Scope

In response to operational experience concerns regarding reactor vessel head lifts (NRC Operating Experience Smart Sample FY2007-03), the inspectors reviewed Virgil C. Summer's programs and procedures to determine whether past and current practices were within the licensing basis, and consistent with guidance in NUREG-0612, "Control of Heavy loads at Nuclear Power Plants," and the Nuclear Energy Institute's (NEI) formal initiative to ensure that heavy load lifts were conducted safely. In addition, the inspectors observed the heavy load lifts for the reactor vessel head removal and reinstallation during RF-17.

b. Findings

In a previous NRC inspection of licensee commitments and basis documentation governing the design and control of outage related reactor vessel head lifts (documented in IR 000395/2007003), inspectors identified that the licensee had failed to incorporate a heavy load lift analysis (regarding load weight, load height, medium present during lift, and bounding reactor vessel head load drop analysis results) into their FSAR. Failure to update the FSAR to reflect aspects of heavy load lifts involving the reactor vessel head and include information from a reactor vessel head drop analysis was a violation of 10 CFR 50.71(e).

The NRC has found industry uncertainty regarding the licensing bases for handling of reactor vessel heads, and as a result issued EGM 07-006, “Enforcement Discretion for Heavy Load Handling Activities,” on September 28, 2007. NEI has informed NRC of industry approval of a formal initiative that specifies actions each plant will take to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices. The NRC staff believes implementation of the initiative will resolve uncertainty in the licensing bases for heavy load handling, and enforcement discretion related to the uncertain aspects of the licensing basis is appropriate during the implementation of the initiative.

Prior to RF-17, the licensee contracted with Westinghouse Corporation to perform a new plant-specific reactor vessel head drop analysis. Phase 1 of this analysis was completed using the methodology and assumptions in WCAP-9198, “Reactor Vessel Head Drop Analysis.” The results of this “realistic” evaluation and formulation of a safe load path were documented in Design Calculation (DC) 039020-026, Revision 0. The calculation provided a bounding maximum head lift height of 35 feet above the reactor vessel flange through air only. The inspectors verified that the new calculated maximum load weight, load height, and safe load path was incorporated into appropriate outage lift procedures. The licensee planned to complete Phase 2 of this analysis using finite element analysis methods prior to the next refueling outage in the Fall of 2009. The inspectors determined that there were no previous documented head lifts that exceeded the new 35 foot bounding height limitation. Based on review of DC 039020-026, vessel head lift procedures and controls, and observation of aspects of the actual vessel head lift activities during RF-17, the inspectors determined that the licensee adequately implemented interim actions in accordance with EGM 07-006 prior to the specified lifts during RF-17, thereby meeting the criteria to warrant enforcement discretion.

Therefore, consistent with EGM 07-006, we are exercising enforcement discretion for the above violation in accordance with Section VII.B.6 of the NRC Enforcement Policy and are not issuing enforcement action for the violation.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed the six surveillance test procedures (STPs) listed below to verify that TS surveillance requirements were followed and that test acceptance criteria were properly specified to ensure that the equipment could perform its intended safety function. The inspectors verified that proper test conditions were established as specified in the procedures, that no equipment preconditioning activities occurred, and that acceptance criteria were met.

In-Service Tests:

- STP-401.002, “Main Steam Line Code Safety Valves ASME OM Code Test” (STP performed for “C” loop main steam safety valves).

Containment Isolation Valve (CIV):

- STP-215.003A, “Containment Isolation Valve Leakage Test for the CVCS, ND, RC, SF, SI, SP, and WL Systems” (for Reactor Building Spray sump penetration 328).

Other Surveillance Tests:

- STP-454.002, “Control Room Emergency Air Cleanup System Performance Test;”
- STP-124.010/011, “Integrated Safeguards Test, Train A/B;”
- STP-125.004A/B, “Diesel Generator A/B Load Rejection Test;” and,
- STP-125.017/018, “Diesel Generator A/B Loss of Offsite Power Test.”

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Controls The inspectors evaluated licensee performance in controlling worker access to radiologically significant areas and monitoring jobs in-progress associated with the 2008 refueling outage. The inspectors directly observed implementation of administrative and physical radiological controls; evaluated radiation worker (radworker) and health physics technician (HPT) knowledge of and proficiency in implementing radiation protection requirements; and assessed worker exposures to radiation and radioactive material.

During facility tours, the inspectors directly observed postings and physical controls for radiation areas, high radiation areas (HRAs), and potential airborne radioactivity areas established within the radiologically controlled area (RCA) of the reactor building, auxiliary building, and radioactive waste (radwaste) processing and storage locations. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys for selected RCA areas. Results were compared to current licensee surveys and assessed against established postings and radiation work permit (RWP) controls. Licensee key control and access barrier effectiveness were evaluated for selected locked high radiation area (LHRA) and very high radiation area (VHRA) locations. Changes to procedural guidance for LHRA and VHRA controls were discussed with health physics (HP) supervisors. Controls and their implementation for storage of irradiated material within the spent fuel pool (SFP) were reviewed and discussed. Established radiological controls were evaluated for selected refueling outage tasks including pressurizer weld overlay, reactor vessel head lift, control rod drive mechanism (CRDM) ventilation ductwork maintenance, and radwaste processing and storage. In addition, licensee controls for areas where dose rates could change significantly because of plant shutdown and refueling operations were reviewed and discussed.

For selected tasks including CRDM maintenance, reactor vessel head lift, and entry into the regenerative heat exchanger room, the inspectors attended pre-job briefings and reviewed RWP details to assess communication of radiological control requirements to workers. Occupational workers' adherence to selected RWPs and HPT proficiency in providing job coverage were evaluated through direct observations and remote monitoring via closed-circuit television. For the selected jobs, electronic dosimeter (ED) alarm set points and worker stay times were evaluated against area radiation survey results.

The inspectors evaluated the effectiveness of radiation exposure controls, including air sampling, barrier integrity, engineering controls, and postings through a review of both internal and external exposure results. Worker exposure as measured by ED and by licensee evaluations of skin doses resulting from discrete radioactive particle or dispersed skin contamination events during current refueling outage activities were reviewed and assessed. For HRA tasks involving significant dose rate gradients, e.g. work in the refueling cavity "sandboxes", the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure. The inspectors also reviewed and discussed selected whole-body count analyses conducted during the current refueling outage.

Radiation protection activities were evaluated against the requirements of FSAR Section 12; TS Sections 6.8 and 6.12; 10 CFR Parts 19 and 20; and approved licensee procedures. Records reviewed are listed in the report Attachment.

Problem Identification and Resolution: Licensee CAP documents associated with access control to radiologically significant areas were reviewed and assessed. This included review of selected CRs related to radworker and HPT performance. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure SAP-999, "Corrective Action Program," Revision (Rev.) 3. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in the report Attachment.

The inspectors completed 21 of the required line-item samples described in Inspection Procedure (IP) 71121.01.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA): Guidance and implementation of the licensee's ALARA program during the 2008 RF-17 were observed and evaluated by the inspectors. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for outage work tasks expected to incur the maximum collective exposures. Reviewed activities included removal of spent fuel bundles for video inspection and re-arrangement of fuel, the removal of shielding and insulation over the reactor vessel head to perform an inspection and repairs of the reactor vessel level instrumentation system (RVLIS) line.

While performing this review the inspectors directly observed drilling of the hydra-nuts on the reactor vessel head, head stud plug installation, RCP motor uncoupling, sand box cover installation, grinding on pressurizer spray nozzle penetrations, cavity inspection and head lift. During the observations, the inspectors evaluated the licensee’s use of engineering controls, low-dose waiting areas, and on-the-job supervision. Incorporation of planning, established work controls, expected dose rates, and dose expenditure into the ALARA pre-job briefings and RWPs for those activities were also reviewed. The inspectors observed several RWP ALARA briefings conducted by the licensee, and the staff interactions with employees and contractors.

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

Trends in individual and collective personnel exposures at the facility were reviewed. Records of year-to-date individual radiation exposures sorted by work groups were examined for significant variations of exposures among workers. The inspectors examined the dose records of all declared pregnant workers during 2007 and 2008 to evaluate total or current gestation dose. Applicable procedures were reviewed to assess licensee controls for declared pregnant workers. Trends in the plant’s three-year rolling average collective exposure history, outage, non-outage and total annual doses for selected years were reviewed and discussed with licensee representatives.

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

Problem Identification and Resolution: The inspectors reviewed the corrective action program documents listed in Section 2OS2 of the report Attachment that were related to the licensee’s ALARA program. The inspectors assessed the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with SAP-999, "Corrective Action Program," Revision 3.

The inspectors completed 15 of 15 required samples.

b. Findings

No findings of significance were identified.

2PS1 Radioactive Gaseous And Liquid Effluent Treatment And Monitoring Systems

a. Inspection Scope

Groundwater Protection: The inspectors reviewed the decommissioning files for events which could have contributed to groundwater contamination. The review did not identify any significant spills or leaks. The review did identify documentation of minor events that had been remediated. The results of sampling wells distributed across the site and the site of the proposed new units identified that two locations had measurable amounts of tritium. The levels detected at these locations were a fraction of the Environmental Protection Agency (EPA) drinking water limit of 20,000 pico-curie per liter (pCi/L). The detection of tritium at these locations was readily explained. The first location was immediately adjacent to a liquid waste holding pond. The second location was approximately ½ mile from the site in the location planned for Unit 3. The source of the tritium was condensate polisher resin that had been tilled into the soil in accordance with a South Carolina land disposal permit in the mid 1990's. Although the condensate polisher resin was from a radiologically clean system, a small amount of tritium had penetrated the steam generator tubes and was entrained in the water that was trapped in the resin. The levels of tritium were near the minimum required level of detection for a shallow well sample and about a quarter of that level from a deeper sample. None of the samples taken in the sampling wells surrounding the above sampling location had elevated tritium nor did the other sampling wells on the existing plant site other than in the immediate vicinity of the liquid waste hold up pond. No other radionuclides were detected in groundwater samples from the site. The inspectors completed 1 sample related to ground water from inspection procedure 7112201.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization: During inspector walk-downs, accessible sections of the liquid and solid radioactive waste (radwaste) processing systems were assessed for material condition and conformance with system design diagrams. Inspected equipment included floor drain tanks; resin transfer piping; resin and filter packaging components; and abandoned evaporator equipment. The inspectors also observed processing of potentially contaminated bagged waste. The inspectors discussed component function, processing system changes, and radwaste program implementation with licensee staff.

The 2006 Annual Effluent Report and radionuclide characterizations from 2006 - 2007 for each major waste stream were reviewed and discussed with radwaste staff. For primary resin and dry active waste (DAW) the inspectors evaluated analyses for hard-to-detect nuclides, reviewed the use of scaling factors, and examined comparison results between licensee waste stream characterizations and outside laboratory data. Waste stream mixing and concentration averaging methodology for powdered resin was evaluated and discussed with radwaste technicians. The inspectors also reviewed the licensee's procedural guidance for monitoring changes in waste stream isotopic mixtures.

Radwaste processing activities and equipment configuration were reviewed for compliance with the licensee's process control program (PCP) and FSAR, Chapter 11. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 61, and guidance provided in the Branch Technical Position on Waste Classification and Waste Form. Reviewed documents are listed in the report Attachment.

Transportation: The inspectors directly observed preparation activities for a shipment of contaminated laundry. The inspectors noted package markings and placarding, performed independent dose rate measurements, and interviewed shipping technicians regarding Department of Transportation (DOT) regulations.

Five shipping records were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors reviewed emergency response information, DOT shipping package classification, radiation survey results, and evaluated whether receiving licensees were authorized to accept the packages. Licensee procedures for opening and closing Type A packages were compared to recommended vendor protocols. In addition, training records for selected individuals currently qualified to ship radioactive material were reviewed.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 71, 49 CFR Parts 172-178; as well as the guidance provided in NUREG-1608. Type B shipments were compared to applicable Certificate of Compliance (CoC) requirements. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution: Selected CRs and self-assessments in the area of radwaste/shipping were reviewed in detail and discussed with licensee personnel. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SAP-999, Corrective Action Program, Rev. 3. The inspectors also evaluated the scope of the licensee's internal

audit program and reviewed recent assessment results. Documents reviewed for problem identification and resolution are listed in the report Attachment.
The inspectors completed 6 of 6 samples as required by inspection procedure 71122.02.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Reactor Safety: Barrier Integrity Cornerstone

a. Inspection Scope

The inspectors verified the accuracy of the licensee's PI submittals listed below for the period April 2007 through March 2008. The inspectors used the performance indicator definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, 'Regulatory Assessment Performance Indicator Guideline," Revision 5, and licensee procedure SAP-1360, "NRC and INPO/WANO Performance Indicators," to check the reporting of each data element. The inspectors sampled licensee event reports (LERs), operator logs, plant status reports, CRs, and performance indicator data sheets to verify that the licensee had properly reported the PI data. Also, the inspectors discussed the PI data with licensee personnel associated with the performance indicator data collection and evaluation.

- RCS Specific Activity; and,
- RCS Identified Leak Rate.

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety and Public Radiation Safety Cornerstones

a. Inspection Scope

The inspectors sampled licensee submittals for the PIs indicated below for the period from January 2007 through March 2008. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used to verify the basis in reporting for each data element.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspectors reviewed CR records generated from January 2007 through March 2008 to ensure that radiological occurrences were properly classified per NEI 99-02 guidance. The inspectors also reviewed electronic dosimeter alarm logs, radioactive material intake records, and monthly PI reports for calendar year 2007 and the first 4 months of 2008. In addition, licensee procedural guidance for classifying and reporting PI events was evaluated. Reviewed documents are listed in the report Attachment.

Public Radiation Safety Cornerstone

- RETS/ODCM Radiological Effluents Occurrence

The inspectors reviewed and evaluated selected radiological liquid and gaseous effluent release data, abnormal release results, cumulative and projected doses to the public, and selected CRs for the period of January 2007 through March 2008. Documents reviewed are listed in the report Attachment.

The inspectors completed two of the required samples for IP 71151, “Performance Indicator Verification,” one for the occupational radiation safety PI and one for the public radiation safety PI.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Review 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 either attending daily screening meetings that briefly discussed major CRs, or accessing the licensee’s computerized corrective action database and reviewing each CR that was initiated.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

The inspectors performed a review of the licensee’s corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The review was focused on repetitive equipment issues, but also considered trends in human performance errors, the results of daily inspector corrective action item screening discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The review nominally considered the six-month period of January 2008 through June 2008. Documents reviewed included licensee monthly and quarterly corrective action trend reports, engineering system health reports, maintenance rule documents, department self-assessment activities, and quality assurance audit reports.

b. Assessment and Observations

The inspectors identified one adverse trend involving personnel failures to follow foreign material exclusion (FME) procedure requirements. Noteworthy examples identified by the inspectors included:

- CR-08-01868: Loss of FME integrity in the "FME Area-High" immediately surrounding the Spent Fuel Pool following fuel offload during Refueling Outage 17.
- CR-08-01183: Loss of FME integrity while an "FME Area-High" was established during the cylinder maintenance and inspection portion of the extended maintenance outage of "B" EDG.
- CR-08-00989: Loss of FME integrity while an "FME Area" was established during an extended licensee effort to replace Circulating Water Pump Screens.

In addition to the NRC-identified items above, the inspectors' trend review revealed the following 2008 licensee-identified CRs that support the conclusion of an adverse trend in FME control: CR-08-00218, CR-08-00306, CR-08-00802, CR-08-01146, CR-08-01618, CR-08-01631, CR-08-01669, CR-08-01747, CR-08-01760, CR-08-01798, CR-08-01799, CR-08-01801, CR-08-01808, CR-08-01880, CR-08-01887, CR-08-01888, CR-08-01894, CR-08-01921, CR-08-01955, CR-08-01969, CR-08-01980, CR-08-02087, CR-08-02092, CR-08-02414, CR-08-02481, CR-08-02580, and CR-08-02630.

The above trend was observed despite a licensee initiative to improve existing procedures and enhance training in anticipation of RF-17. The licensee has not yet incorporated these trend comments into their CAP.

.3 Annual Sample Review

a. Inspection Scope

The inspectors reviewed one issue in detail to evaluate the effectiveness of the licensee's corrective actions for important safety issues documented in CR-08-00292 and CR-08-00301. This review was associated with the malfunction of a main feedwater regulating valve (FRV) that caused a steam generator water level transient requiring a manual reactor trip on January 24, 2008. The inspectors assessed whether the issue was appropriately identified; documented accurately and completely; properly classified and prioritized; adequately considered extent of condition, generic implications, common cause, and previous occurrences; adequately identified root causes/apparent causes;

and identified appropriate corrective actions. Also, the inspectors verified the issue was processed in accordance with SAP-999, "Corrective Action Program."

b. Findings and Observations

Introduction: A Green self-revealing finding was identified for the failure to implement effective and timely corrective actions to prevent malfunction of "C" FRV IFV00498 that resulted in a manual reactor trip.

Description: On January 24, 2008, a manual reactor trip was inserted by the operators due to rapidly decreasing steam generator level following malfunction of the "C" FRV and the inability to manually control the valve position. The reactor trip was uncomplicated and all safety systems responded appropriately. The licensee documented this event in LER 2008-001-00, "Manual Reactor Trip Due to Low Steam Generator Level Caused by Feedwater Flow Control Valve Malfunction."

The licensee's Root Cause Analysis of the "C" FRV failure determined that it was caused by malfunction of the pilot valve stem in the Bailey AV-1 pneumatic positioner due to fretting of the pilot valve stem and foreign material intrusion. Licensee inspections following the trip identified both the evidence of fretting on the pilot valve stem guides and the stem body, as well as the existence of particulate (brass) from upstream air supply components (mainly from the pilot check valve due to vibration induced component wear). Either of these conditions alone could have caused sticking of the pilot valve stem resulting in erratic FRV operation, valve failure to respond to demand or uncontrolled valve movement. The Root Cause Analysis identified two other causal factors for the FRV malfunction including: 1) the scope of a previous Root Cause Analysis 05-03640 (for similar "C" FRV positioner malfunctions noted in September 2005) was too narrowly focused to identify root causes and corrective actions to prevent recurrence; and, 2) the design of the FRV control system (i.e., instrument air) did not meet vendor recommendations for air quality at the positioner. Noteworthy items that formed the basis for the conclusion that corrective actions for previous failures were ineffective included the following:

- CR-04-00884 documented a failure of the positioner for the "C" FRV that resulted in an automatic reactor trip in March 2004 due to fretting of the pilot valve stem. All three FRV positioners were replaced and a preventive maintenance schedule was setup to replace the positioners every refueling outage. As part of the long term corrective actions, ECR 50583 was initiated in October 2004 to replace all three positioners with a different manufacturer design during the next refueling outage; however, this ECR was later deferred.

- CR-04-03772 documented anomalous "C" FRV response due to fretting of the positioner pilot valve in December 2004. All three FRV positioners were replaced. The CR referenced the need to replace the positioners with a different design under ECR 50583.

- CR-05-03640 documented erratic behavior of the "C" FRV positioner in September 2005 due to sticking of the positioner pilot valve from foreign material intrusion of contaminants. Investigations concluded that most likely, the source of the contaminants were present in the positioners when installed and not from the air supply. The pilot valves were replaced and a preventive maintenance activity was setup to perform quarterly valve inspections until ECR 50583 could be implemented.

- Quarterly FRV positioner pilot valve visual inspections conducted on 3/29/07, 6/22/07, 9/7/07, and 9/27/07, identified minor traces of residue and/or small particulate composed of brass, aluminum, and silica. The brass was subsequently determined to originate from vibration induced wear of the upstream pilot check valve and the aluminum possibly from instrument air system desiccant.

- Numerous external operating experience issues were identified since 1996 involving similar feedwater flow anomalies caused by sticking pilot valve positioners due to pilot valve stem fretting or foreign material intrusion. In December 2004, Topical Report 4-42, "Review of Air-Operated Valve Related Events," was issued dealing with air-operated valve performance problems, including issues related to FRV positioner pilot valve fretting and foreign material intrusion. Engineering evaluation of this topical report took over 22 months to complete and the extent of the recommendations involved reference to implementing ECR 50583.

During RF-17, the licensee completed modification ECR 50478 on all three FRVs in order to minimize the vulnerability to further positioner pilot valve malfunction due to fretting or foreign material intrusion until the current positioner models can be replaced with a new design. This modification replaced all control air tubing with stainless steel, relocated control air components to reduce the potential for vibration induced service wear from the positioner pilot valve check valve, and installed filters to improve supply air quality to the positioners.

<u>Analysis</u>: The licensee's CAP procedure (SAP-999) defines a corrective action to prevent recurrence (CAPR) as a condition requiring actions to prevent recurrence by addressing the root cause. The inspectors determined that the failure to conduct adequate root cause analysis and implement adequate and timely actions to prevent recurrence of a reactor trip by a known FRV degradation previously identified in the licensee's CAP database with CAPRs via CR-04-00884 and CR-05-03640 was a performance deficiency. This finding is greater than minor because it is associated with the Initiating Event Cornerstone attribute of Equipment Performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at power operations. The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available.

The cause of this finding was directly related to the aspect of appropriate and timely corrective action in the cross-cutting area of Problem Identification and Resolution (Corrective Action component) because actions to address previously identified “C” FRV malfunction due to positioner pilot valve fretting and foreign material intrusion were not implemented in a timely manner (P.1.d).

Enforcement: No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because the performance deficiency involved non-safety related equipment. Since this finding was entered into the licensee’s corrective action program as CR-08-00292, and was determined to be of very low safety significance, it will be tracked as Finding (FIN) 05000395/2008003-01, Untimely Corrective Actions To Resolve Feedwater Regulating Valve Malfunction Resulted In Reactor Trip.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted the following observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors’ normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 (Discussed) Temporary Instruction (TI) 2515/172: Reactor Coolant System Dissimilar Metal Butt Welds (DMBWs).

a. Inspection Scope

The inspectors reviewed the licensee’s activities related to the inspection and mitigation of dissimilar metal butt welds in the RCS to ensure that the licensee activities were consistent with the industry requirements established in the Materials and Reliability Program (MRP) document MRP-139, “Primary System Piping Butt Weld Inspection and Evaluation Guidelines,” dated July 2005. The inspectors’ activities took place during RF-17 and covered the following: a) documentation and direct observation of the weld overlay process on the Safety/Relief Valve (SRV) 4C Nozzle; and b) documentation and direct observation of portions of the volumetric examination of the nozzles of the PORV, SRV 4A, and SRV 4B DMBW after completion of the full structural weld overlay (FSWOL). The inspectors also reviewed the qualifications of the welders as well as those of the personnel conducting the ultrasonic testing. The inspectors also observed the calibration of the ultrasonic equipment (e.g., phased array) prior to its use in the field. The inspectors only implemented portions of TI-172 that corresponded to the available activities during the aforementioned outage. The remaining TI inspection activities are scheduled to be completed prior to the end of 2008.

Volumetric Examinations:

1) For each examination inspected, was the activity performed in accordance with the examination guidelines in MRP-139, Section 5.1, for unmitigated welds or mechanical stress improved welds and consistent with NRC staff relief request authorization for overlaid welds?

Pressurizer Nozzles for the PORV, SRV 4A, and SRV 4B Dissimilar Metal Butt Weld (DMBW) After Mitigation by Full Structural Weld Overlay (FSWOL)

Yes, the volumetric examinations of the presurizer nozzles for the PORV, SRV 4A, and SRV 4B DMBW after completion of the FSWOL were performed in accordance with a qualified procedure for UT examination, consistent with MRP-139 requirements.

The procedure was qualified in accordance with ASME Section XI, Appendix VIII, as implemented through the Electrical Power Research Institute (EPRI) Performance Demonstration Initiative (PDI) Program. The licensee utilized phased array UT technology to perform the examination using procedure PDI-UT-126, “Phased Array UT Procedure.” The UT examiners scanned the FSWOL in two axial and two circumferential directions. The licensee was able to obtain 100% coverage in the UT examination performed to detect fabrication flaws in the FSWOL.

2) For each examination inspected, was the activity performed by qualified personnel?

Pressurizer Nozzle for the PORV, SRV4A, and SRV 4B DMBW After Mitigation by FSWOL

Yes, the personnel involved in the UT examinations of the pressurizer PORV, SRV 4A, and SRV 4B FSWOL were qualified in accordance with MRP-139 requirements. The examiners were qualified Level II in the UT method as required by the UT procedure and in accordance with the vendor’s written practice for NDE personnel. The UT examiners were also PDI qualified for the specific UT procedure they implemented. The final examination report was reviewed by a vendor’s Level III in the UT method and a licensee’s Level III in the UT method.

3) For each examination inspected, was the activity performed such that deficiencies were identified, dispositioned, and resolved?

Pressurizer Nozzles for the PORV, SRV 4A, and SRV 4B DMBW After Mitigation by FSWOL

Yes, the inspectors reviewed documentation and directly observed field work to verify that deficiencies were identified, dispositioned, and resolved. However, based on the inspection activities, no deficiencies were identified.

Weld Overlays:

1) For each weld overlay inspected, was the activity performed in accordance with ASME Code welding requirements and consistent with NRC staff relief request authorizations? Has the licensee submitted a relief request and obtained NRR staff authorization to install weld overlays?
Pressurizer SRV 4C Nozzle DMBW FSWOL

Yes, the licensee installed the pressurizer SRV 4C nozzle DMBW FSWOL in accordance with the applicable sections of the ASME Boiler and Pressure Vessel Code (ASME Code). The licensee sought approval for a proposed alternative to certain ASME Code requirements through Relief Request (RR)-III-05, based on ASME Code Case N-740. Approval for the relief request was obtained and an NRC safety evaluation report (SER) was issued on February 14, 2008; ADAMS Accession Number ML080460036.

The inspectors reviewed welding procedure specifications, procedure qualification records, weld wire certifications, and the in-process welding process control sheets for compliance to ASME Section IX requirements and adherence to the SER. The inspectors also evaluated a number of the licensee's corrective action program documents (condition reports), and third party contractor corrective action process issue reports regarding weld overlay quality issues.

2) For each weld overlay inspected, was the activity performed by qualified personnel?

Pressurizer SRV 4C Nozzle DMBW FSWOL

Yes, welding personnel were qualified in accordance with the requirements identified in ASME Code Section IX. The inspectors reviewed the welder performance qualification test records and compared them with the requirements of QW-300. The in-process welding process control sheets were reviewed for compliance with the proposed alternative and ASME Code Section IX requirements.

3) For each weld overlay inspected, was the activity performed such that deficiencies were identified, dispositioned, and resolved?

Pressurizer SRV 4C Nozzle DMBW FSWOL

Although WSI had a program in place to identify, disposition, and resolve deficiencies encountered during the FSWOL work, no deficiencies were identified.

.3 (Closed) Severity Level III Violation 05000395/2007502-01: Emergency Action Level (EAL) Changes Resulted in Decreases in Effectiveness and a Non-Standard EAL Scheme.

The inspectors evaluated the licensee's corrective action program responses to the October 12, 2007, Notice of Violation associated with NRC Emergency Preparedness Inspection Report 05000395/2007502, for issues regarding multiple changes to the licensees EALs, between the years of 1980 and 2006, without prior NRC approval. The inspectors reviewed the corrective actions the licensee described in its correspondence dated November 12, 2007 entitled, "Reply to a Notice of Violation: EA-07-079 NRC Inspection Report 2007502" and held interviews with appropriate personnel. Immediate corrective actions have been implemented and the long-term corrective action to prevent recurrence of converting to an EAL scheme based on NEI-99-01, "Methodology for Development of Emergency Actions Levels" Revision 4, has a projected completion date of October 2008. NRC staff determined that the long-term corrective action to prevent recurrence does not need to be complete prior to closing this violation. This violation is closed.

b. Findings and Observations

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Jeffrey Archie and other members of the licensee staff on July 10, 2008. The licensee acknowledged the results. The inspectors confirmed that inspection activities discussed in this report did not contain proprietary material.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Archie, Vice President, Nuclear Operations
L. Bennett, Manager, Plant Support Engineering
L. Blue, Manager, Nuclear Training
M. Browne, Manager, Quality Systems
A. Cribb, Supervisor, Nuclear Licensing
G. Douglass, Manager, Nuclear Protection Services
M. Fowlkes, General Manager, Engineering Services
D. Gatlin, General Manager, Nuclear Plant Operations
R. Justice, Manager, Maintenance Services
D. Lavigne, General Manager, Organizational / Development Effectiveness
G. Lippard, Manager, Operations
M. Mosley, Manager, Chemistry Services
P. Mothena, Manager, Health Physics and Safety Services
J. Nesbitt, Manager, Materials and Procurement
D. Shue, Manager, Planning / Outage
W. Stuart, Manager, Design Engineering
B. Thompson, Manager, Nuclear Licensing
S. Zarandi, General Manager, Nuclear Support Services

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Opened and Closed

05000395/2008003-01	FIN	Untimely Corrective Actions To Resolve Feedwater Regulating Valve Malfunction Resulted In Reactor Trip (Section 4OA2.3)

Closed

05000395/2007502-01	VIO	EAL Changes Resulted in Decreases in Effectiveness and a Non-Standard EAL Scheme. (Section 4OA5.3)

Discussed

2515/172	TI	Reactor Coolant System Dissimilar Metal Butt Welds (DMBWs) (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures and Drawings
SOP-118, Component Cooling Water
FSAR 9.2.2, Component Cooling Water System
D-302-611/612/613/614, Component Cooling System Flow Diagram
SOP-115, Residual Heat Removal
E-302-641, Residual Heat Removal System Flow Diagram
SOP-306, Emergency Diesel Generator

Detailed Equipment Alignment – RHR System
FSAR Chapters 5 and 6
TS Sections 3.5.2, 5.3, 5.4, 3.9.7.1, 3.9.7.2
SOP- 115, Residual Heat Removal
AB-7, Residual Heat Removal System Description
Design Basis Documents for RHR
E-302-641, Residual Heat Removal System
E-302-693, Safety Injection System
CR data base search and review of RHR system CRs from 01/01/07-05/01/08, (10 CRs reviewed)
STP-230.007, RHR Pump A/B and Check Valve full flow test - 05/17/08
STP-205.004, RHR Pump and Valve Operability Test - 04/10/08 and 03/27/08

Section 1R08: In-service Inspection Activities

Procedures/Calculations/Engineering Documents
SAP-1100, “Boric Acid Corrosion Control Program,” Revision 1
PTP-151.001A, “Inspection For Boric Acid Corrosion,” Revision 1
PPSEG-19, “Boric Acid Corrosion Evaluation,” Revision 0
Performance Demonstration Initiative (PDI) – UT-2, PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds
PDI Protocol SI-UT-126, Phased Array UT Procedure
Qualification and Certification of 4 Quality Control Inspection Personnel
Certificate of Compliance/Conformance for 2”-1500# Edward Forged Pressure Combo Valve
Certified Material test Report for 2” S/160 Smls Pipe, SA312 Type 304, Tubacex, Heat # 34913
Calculation Number DC04010-001, Reactor Vessel Head Effective Degradation Years, Revision 3
Laboratory Testing, Inc., Certified Test Report for Ultragel II – 07125
Certified Material Test Report by Special Metals for Inconel filler metal 52M, Heat/Lot # NX0T86TK, reviewed and accepted by Welding Services Inc.

Corrective Action Documents
CR-06-03321, Relevant indication identified on fillet weld
CR-06-03349, Dry boric acid on valve XVT08056 vent cap
CR-06-03360, Non-minor inactive boric acid leak on XVT08877A-SI end cap
CR-06-03382, Boric acid at the drain for the ECCS test check valves

CR-06-03954, SAP-1100 BACCP procedure improvements
CR-06-04220, AC Sys Industrial CLR outlet isolation valve external corrosion
CR-07-00555, Increased RCS unidentified leakage
CR-07-00570, Boric acid leakage on safety injection system
CR-07-01142, Evaluation of the overall ISI program health requirement
CR-07-01234, Boric acid around RHR pump “A” casing bolts and pump supports
CR-07-01907, Boric acid leakage on spent fuel heat exchanger “A” outlet valve
CR-07-01908, Boric acid leakage on RB spray pump full-flow test isolation valve
CR-07-03205, Boric acid on hydro test pump discharge header relief valve
*CR-08-01950, Boric Acid Program enhancements identified by NRC

* Corrective Action documents created as a result of this inspection.

<u>Other</u>
Welding Services Inc., 2008 Lessons Learned document (proprietary information -returned prior to exit)
In-service Inspection Report #14, dated 2-13-2007, (which includes the R16 In-service Examination Evaluation)
Work Order 0711229-010, visual examination (VT-2) of reactor vessel upper head, as required per MRP-139
Structural Integrity Associates, Inc., Procedure Number SI-UT-126, Procedure for the Phased Array Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds, Revision 3
Structural Integrity Associates, Inc., Phased Array Ultrasonic Examination Record for Pressurizer SRV 4A 4501-36 OL Nozzle Weld Overlay
Structural Integrity Associates, Inc., Phased Array Ultrasonic Examination Record for Pressurizer SRV 4B 4501-35 OL Nozzle Weld Overlay
Structural Integrity Associates, Inc., Phased Array Ultrasonic Examination Record for Pressurizer PORV 4502-24 OL Nozzle Weld Overlay
Welding Services, Inc., Liquid Penetrant Inspection Report Number 103669-PT-003, base metal PT of weld 4502-24 (OL)
Welding Services, Inc., Liquid Penetrant Inspection Report Number 103669-PT-007, PT after 3rd bead of weld 4502-24 (OL)
Welding Services, Inc., Welding Procedure Specification WPS 03-08-T-804-Bottom, Revision 2
Washington Group Calibration Data Sheet for welds 2-2501-53 and 2-2501-54
Washington Group Calibration Data Sheet for welds 2-2525-7 and 2-2525-8, and 2-2525-9
Sonaspection Int’l Structural Integrity Report, dated 10/24/2006, for calibration block SI-8-AX-02 (which was used for the phased array examination of the pressurizer nozzles), which included the material certificate, ultrasonic report, mechanical inspection report, and QA release note
Calibration Certificate for Omniscan phased array equipment used by Structural Integrity Associates for UT of pressurizer weld overlays SCE&G Relief Request RR-III-05, dated 06/01/2007

Section 1R18: Plant Modifications
Procedures and Drawings
SOP-117, Service Water System
SOP-123, Spent Fuel Cooling System
SOP-125, Industrial Cooling Water System
SOP-306, Emergency Diesel Generator
STP-125.004A, Diesel Generator "A" Load Rejection Test
STP-125.017, Diesel Generator "A" Loss of Offsite Power Test
EMP-100.004, Installation of Temporary Alternate Feed Cable for Spent Fuel Pool Pumps
ICP-180.013, Set-Up and Adjustment of Diesel Generator "A" Governor
Fire Emergency Procedures FEP-2.0, 3.0, and 4.0
D-302-651, Spent Fuel Pool Cooling System Flow Diagram
D-302-222, Service Water System Flow Diagram
B-208-024, sheet DG32A/B, Diesel Generator "A" Governor Control

Work Orders
WOs 0705865 and 0711188
WO 0603544
WO 0603989

Other Documents
TS 3/4.6.2.3
FSAR 6.2.2, 9.1.3.3, and 9.2.1
Design Basis Documents for Spent Fuel Pool and Service Water Systems
Tagout 08-0131
TR 06610-002, FMEA and Set-up of Woodward 2301A Governor and Digital Reference unit
Regulatory Guide 1.9, Rev 0 and Rev 1, Selection of Diesel Generator Set Capacity for Standby Power Supplies
Design Basis Document for Diesel Generator Engine Support and Control Systems
FSAR 8.3.1.1.2

Section 2OS1: Access Control to Radiologically Significant Areas
Procedures, Guidance Documents, and Manuals
HPP-160, Control and Posting of Radiation Control Zones, Rev. 11
HPP-517, Multiple Whole Body and Extremity Badging Exposure Calculations, Rev. 8
SAP-140, Plant Key Control, Rev. 6
SPP-210, Security Lock and Key Control, Rev. 9
SAP-999, Corrective Action Program, Rev. 3

Records and Data
RWP No. 08-71, CRDM Ductwork, Rev. 3
RWP No. 08-83, "C" RCP Main Flange Gasket Replacement, Rev. 1
RWP No. 08-97, Pressurizer Weld Overlay, Rev. 3
Radiological Survey, Reactor Head Near Shield Doors, 4/29/08
Radiological Survey, "B" RCP Removed From Casing, 5/4/05
Radiological Survey, Pressurizer H/L Surge Line, 5/4/08 and 5/5/08

Personnel Contamination Event Tracking Log, 4/28/08 – 5/9/08, and selected Whole Body Count records

CAP Documents
CR 07-2123, Self-assessment of HP Field Operations, 7/19/07
CR 07-1960, Discovered HRA key to Excess Letdown Heat Exchanger missing during annual inventory, 7/2/07
CR 08-1331, Dose rate alarm not properly reported to HP, 4/8/08
CR 08-1517, Radiological posting found underneath insulation, 4/22/08
CR 08-1706, Dose rate alarm received in pressurizer cubicle, 4/29/08

Section 2OS2: ALARA Planning and Controls
Procedures, Instructions, Guidance Documents, and Operating Manuals
HPP-151, Use Of The Radiation Work Permit And Standing Radiation Work Permit, Rev. 8
HPP-154, Issuance And Control Of Respiratory Protection Equipment, Rev. 12
HPP-155, Control of Airborne Radiation Exposure (DAC-HRS), Rev. 11
HPP-0160, Control and Posting of Radiation Control Zones, Rev. 11
HPP-0401, Issuance, Termination and Use of RWPS and SRWPS, Rev. 18
HPP-0402, Radiological Survey Requirements and Controls for Reactor Building and Incore Pit Entries, Rev. 11
HPP-0403, Radiological Controls for Nuclear Work Activities, Rev. 10
SAP-500, Health Physics Manual, Rev. 11
Exposure Reduction 5 Year Plan - 2008
Refuel 16 Outage Report
ALARA Planning Reports (APR)
APR 08-014, All Manual Filter Change outs for 2008 LHRA
APR 08-051, Operations RCS Leakage Inspections and Valve Lineups Inside and Outside the Reactor Building
APR 08-061, All Work Associated With LCV00459, LCV00460, and XVG 08085
APR 08-070, Reactor Vessel Head Work to Support Refueling
APR 08-071, RVLIS, CRDM Ductwork, Nozzle Covers, NI Covers, Cavity Seal Ring, Missile Shields, Detension and Tension Studs
APR 08-081, Lower Vessel Inspections

Records and Data
RWP 06-052, All Scaffolding inside the Reactor Building
RWP 06-071, RVLIS, CRDM Ductwork, Nozzle Covers, NI Covers, Cavity Seal Ring, Missile Shields, Detension and Tension Studs
RWP 06-079 Volumetric Head Inspection
RWP 06-096, Reactor Building Sump Clogging Modification Install RHR/SI Sump Strainers
RWP 06-104, Remove/Replace Shield, Shroud & Insulation From The Reactor Head
V.C. Summer 2007 Annual ALARA Report
ALARA Committee Meeting Minutes 4th Quarter 2006, 12/12/2006
ALARA Committee Meeting Minutes 1st Quarter 2007, 3/28/2007
ALARA Committee Meeting Minutes 2nd Quarter 2007, 6/27/2007
ALARA Committee Meeting Minutes 3rd Quarter 2007, 9/26/2007

ALARA Committee Meeting Minutes 4th Quarter 2007, 12/12/2007

Corrective Action Program Documents
CR 07-02579, Removal and Inventory of Used Incore Flux Thimbles was difficult because they were stuck and required cutting of disposal tubes. 65 mrem used.

Section 2PS1: Radioactive Gaseous And Liquid Effluent Treatment And Monitoring Systems
Records
10 CFR 50.75(g) log with CR numbers
10 CFR 50.75(g) investigation report for leak at CST sample valve
10 CFR 50.75(g) investigation report for tritium found at new plant site.
Annual Effluent Radioactive Release Report, January – December 2006
Annual Radiological Environmental Operating Report, January – December 2006
Self Assessment Report, VCS Environmental Monitoring Program, March 18, 2008
Self Assessment Report, VCS Site Count Room Program, March 18, 2008

Procedures
HPP-1020, Environmental Sample Collection, Rev. 3
HPP-1022, Environmental Sampling and Analytical Requirements, Rev.5A
HPP-1024, Groundwater Monitoring Well Sampling, Rev.3

Section 2PS2: Radioactive Material Processing and Transportation
Procedures, Manuals, and Guides
HPP-0703, Shipping Radioactive Material, Rev. 17
HPP-0712, Classification of Radioactive Materials, Rev. 9
HPP-0716.014, CNS 14-215H Cask Handling, Rev. 0
HPP-0717, Sample Collection, Preparation, and Analysis Techniques for Assuring Compliance with 10CFR61, Rev. 7
PCP-001, Process Control Program for Processing Wet Waste, Rev. 11A
SAP-999, Corrective Action Program, Rev. 3

Shipping Records and Radwaste Data
Shipment 06-08, Radwaste Resin, 2/16/06
Shipment 06-88, Vessel Ex-core Dosimetry, 11/16/06
Shipment 07-05, Contaminated Trash (DAW), 1/16/07
Shipment 07-40, Iridium-192 Radiography Source, 8/14/07
Shipment 08-13, Radwaste Resin, 2/28/08
10 CFR Part 61 Radioactive Waste Stream Analysis Reports, DAW (2007), DAW, Filters, and Resins (2006)
CoC No. 9269, Model 650L Iridium-192 Source Changer

CAP Documents
SA07-HP-02, Self-assessment of Radwaste Group, 4/30/07 – 8/13/07
CR 07-663, Pressure gauge used for resin dewatering not included in station calibration program, 2/21/07

CR 07-1771, Provide training to OPs on how to respond to rad material shipping accidents if emergency phone call is received, 6/11/07
CR 08-583, Explosive detector containing a Nickel-63 source shipped as non-rad, 2/11/08
CR 08-1846, Resin transfer overflowed the spent resin storage tank, 5/7/08

Section 4OA1: Performance Indicator Verification
Records
PI Submittals from January 2007 to April 2008

Procedures
HPP-242, Reporting of NRC Performance Indicators, Rev. 0
SAP-1360, NRC and INPO/WANO Performance Indicators, Rev.0

Condition Reports Initiated for NRC Identified Issues
CR-08-01385, Yellow painted rubber shoe covers found outside RCA
CR-08-01443, “B” train RHR piping resting on steel wall cutout opening between adjacent room
CR-08-01605, QC boron walkdown found boron on FT-434-LD-RC
CR-08-01609, Snubber testing not incorporated 10 CFR 50.65(a)(4)
CR-08-01612, Quality Control (QC) identified boron on XVT-18836-SI
CR-08-01613, QC identified boron on XVT-11-ST in Penetration 317
CR-08-01618, Duct tape found during QC walkdown on “B” RCP drain line
CR-08-01623, During QC walkdown found boron on XVT-8056-RC
CR-08-01625, During QC walkdown found broken glass to standpipe “B” acc
CR-08-01690, Steam generator water level allowed to be less than procedure requirements
CR-08-01705, Reactor building inspections for as-found boron leakage not conducted at expected reactor coolant system pressure
CR-08-01756, RB spray pump control board switch tagged with wrong colored hold tag
CR-08-01767, Concern for possible preconditioning of main steam safety valve testing
CR-08-01855, NRC inspector found badge broken from employee neckstrap in PA
CR-08-01868, Loss of foreign material exclusion (FME) controls in spent fuel pool area
CR-08-02580, Reactor building FME items during closeout
CR-08-02602, Failure to follow procedure results in 8706 error in breaker
CR-08-02652, Wet boron found on XVT8363B-CS
CR-08-02828, Simulator not modified prior to operator requalification training following RF-17 RBCU modification changes

LIST OF ACRONYMS

AB	Auxiliary Building
ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
BPVC	Boiler and Pressure Vessel Code
CAP	Corrective Action Program
CB	Control Building
CCW	Component Cooling Water
CIV	Containment Isolation Valve
CoC	Certificate of Compliance
CR	Condition Report
CRDM	Control Rod Drive Mechanism
CFR	Code of Federal Regulations
DAW	Dry Active Waste
DC	Design Calculation
DG	Diesel Generator
DMBW	Dissimilar Metal Butt Weld
DOT	U.S. Department of Transportation
ECR	Engineering Change Request
ED	Electronic Dosimeter
EDG	Emergency Diesel Generator
EDY	Effective Degradation Years
FME	Foreign Material Exclusion
EPA	U.S Environmental Protection Agency
EPRI	Electrical Power Research Institute
ES	Engineering Service
FIN	Finding
FME	Foreign Material Exclusion
FRV	Feedwater Regulating Valve
FSAR	Final Safety Analysis Report
FSWOL	Full Structural Weld Overlay
GTP	General Test Procedure
HRA	High Radiation Area
HPT	Health Physics Technician
HVAC	Heating Ventilation and Air Conditioning
IB	Intermediate Building
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	In-Service Inspection
LER	Licensee Event Report
LHRA	Locked High Radiation Area
MPFF	Maintenance Preventable Functional Failures
MR	Maintenance Rule
mrem/hr	millirem/hour

MRP	Materials and Reliability Program
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OAP	Operations Administrative Procedure
OOS	Out of Service
pCi/L	pico-Curies per Liter
PCP	Process Control Program
PDI	Performance Demonstration Initiative
PI	Performance Indicator
PMT	Post-Maintenance Testing
PORV	Power Operated Relief Valve
PT	Liquid Penetrant
radwaste	Radioactive Waste
RBCU	Reactor Building Cooling Unit
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
Rev.	Revision
RF-17	Seventeenth Refueling Outage
RHR	Residual Heat Removal
RP	Radiation Protection
ROP	Reactor Oversight Process
RR	Relief Request
RTP	Rated Thermal Power
RWP	Radiation Work Permit
RVLIS	Reactor Vessel Level Indication System
SAP	Station Administrative Procedure
SCE&G	South Carolina Electric and Gas
SDP	Significance Determination Process
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SG	Steam Generator
SOP	System Operating Procedure
SRV	Safety Relief Valve
SSC	Structures, Systems, or Components
STP	Surveillance Test Procedure
SW	Service Water
SWPH	Service Water Pump House
TI	Temporary Instruction
TS	Technical Specification
UT	Ultrasonic Testing
VT-2	Visual Examination
VUHP	Vessel Upper Head Penetration
WO	Work Order
VHRA	Very High Radiation Area