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

January 26, 2004

South Carolina Electric & Gas Company
ATTN: Mr. Stephen A. Byrne
Senior Vice President, Nuclear Operations
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT NO. 05000395/2003005

Dear Mr. Byrne:

On December 27, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Virgil C. Summer Nuclear Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 8, 2004, with Mr. Jeff Archie, and other members of your staff.

The inspections 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. The inspections also examined activities that support the application for a renewed license for the Summer Station.

Based on the results of this inspection no findings of significance were identified. However, a licensee-identified violation determined to be of very low safety significance is listed in Section 4OA7 of this report. If you contest this non-cited violation (NCV) in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Virgil C. Summer Nuclear Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of

NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

Sincerely,

/RA/

Kerry D. Landis, Chief
Reactor Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report No. 05000395/2003005
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-395
License No.:	NPF-12
Report No.:	05000395/2003005
Licensee:	South Carolina Electric & Gas (SCE&G) Company
Facility:	Virgil C. Summer Nuclear Station
Location:	P. O. Box 88 Jenkinsville, SC 29065
Dates:	September 28 - December 27, 2003
Inspectors:	M. Widmann, Senior Resident Inspector M. King, Resident Inspector J. Lenahan, Senior Reactor Inspector, RII (Sections1R08, 4AO3.2, 4OA5) R. Hamilton, Radiation Specialist, RII (Sections 2OS1, 4OA1.2)
Approved by:	K. D. Landis, Chief Reactor Projects Branch 5 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000395/2003005; 09/28/2003 - 12/27/2003; Virgil C. Summer Nuclear Station; Routine Integrated Report, and Other Activities.

The report covered a three month period of inspection by resident inspectors; and announced inspections by one regional senior reactor inspector, and a Radiation Specialist. No findings of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self Revealing Findings

None

B. Licensee-Identified Violation

- One violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. This violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period at 100 percent power. On October 8, power was reduced to 85 percent to support planned main steam safety valve testing. On October 11, the unit completed a planned shutdown (Mode 3) to commence the fourteenth refueling outage (RF-14). Following RF-14, criticality and Mode 1 were achieved on November 23 and 25, respectively. The unit was returned to 100 percent power on November 29 and remained at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors performed a seasonal weather related condition inspection for adverse weather protection to evaluate implementation of adverse weather procedure Operations Administrative Procedure OAP-109.1, “Guidelines for Severe Weather.” The inspectors reviewed the licensee’s plant cold weather preparation for the sodium-hydroxide storage tank, condensate storage tank, reactor makeup storage tank and refueling water storage tank instrumentation. This review was performed to assess the risk of weather related initiating events, and whether measures taken, adequately protected accident mitigation systems from adverse weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Availability of Redundant Equipment

a. Inspection Scope

The inspectors verified through plant walkdowns that with a train of equipment removed from service, the opposite train of equipment was correctly aligned, available and operable. The following systems / components were verified:

- B train control room ventilation while A train was removed from service during emergent maintenance activities on the control room dampers;
- B emergency diesel generator (EDG) while the A EDG was out of service for scheduled maintenance overhaul;
- Verification of containment integrity and capability of equipment hatch closure during RF-14.

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 inspection included review of outstanding maintenance work requests (MWRs) and related condition evaluation reports (CERS) to verify that the licensee had properly identified and resolved equipment alignment problems that could impact mitigating system availability. Specific procedures and documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.2 Semiannual Inspection: Residual Heat Removal (RHR) System

a. Inspection Scope

The inspectors performed a detailed review and walkdown of the RHR system and related support systems to identify any discrepancies between the current operating system equipment lineup and the designed lineup. This walkdown included accessible areas inside the containment during the refueling outage and outside containment following unit return to power. In addition, the inspectors reviewed completed surveillance procedures, outstanding maintenance work requests, leakage assessments and RHR system related CERs to verify that the licensee had properly identified and resolved equipment problems that could affect the availability and operability of the RHR system. The inspectors also reviewed the work package, contingency planning, risk assessments and actual field work during implementation of engineering change request ECR-500079, RHR Pump A - Split Coupling Retro Fit Modification, performed during the RF-14 outage.

The inspectors reviewed dominant cutsets for the RHR system fault tree and results of the low head injection system and normal RHR subsystem vulnerability studies for insight into system failures. Based on this insight, the inspectors reviewed selected surveillance test procedures, including refueling water storage tank low-low level safeguards actuation signal surveillance and selected RHR system check valves maintenance histories. The inspectors also reviewed Electrical Maintenance Procedure (EMP)-100.002, “Emergency Installation of Cable for the RHR System,” to verify this procedure could adequately support installation of emergency power to the B RHR pump and selected RHR motor operated valves upon loss of normal power supply. Other specific procedures and documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed recent CERs, MWRs, and impairments associated with the fire suppression 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 twelve areas:

- 1DA switchgear room (fire zone IB-20);
- 1DB switchgear rooms and heating, ventilation and air conditioning (HVAC) rooms (fire zones IB-16, IB-17, IB-22.2);
- Relay room solid state protection system (SSPS) instrumentation and inverter (fire zones CB-6, CB-10, CB-12);
- Control room (fire zone CB-17.1);
- Control building, OSC and DRCB-103 cable spread (fire zones CB-1.1, CB-1.2, CB-2, CB-5);
- Control building cable spreading rooms (fire zones CB-4, CB-15);
- Diesel generator rooms A and B (fire zones DG-1.1/1.2, DG-2.1/2.2);
- HVAC chilled water pump rooms A and B (fire zones IB-7.2, IB-9, IB-23.1);
- Charging pump rooms A, B and C (fire zones AB-1.5, AB-1.6, AB-1.7);
- RHR, reactor building (RB) spray pump rooms (fire zones AB-1.2, AB-1.3);
- Reactor building (fire zone RB-1);
- Service water pump house (fire zones SWPH 1, 3, 5.1/5.2).

Under the problem identification and resolution section of this inspection attachment, the inspectors also reviewed CER 0-C-03-3393, “two of nine temperature elements in the reactor building used for compensatory fire protection actions were inoperable for two days.”

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

The inspectors reviewed heat exchanger testing programs for the component cooling water heat exchangers (CCW Hxs) and the licensee’s bio-fouling controls for the service water system. The CCW system is ranked by the licensee as the highest risk significant system based on importance with the service water system ranked fifth. This review verified that the frequency of testing was sufficient and established acceptance criteria was appropriate to detect any potential CCW Hx deficiencies. The inspectors specifically examined the test program for masked or degraded performance and to assess the adequacy of the licensee implemented service water system bio-fouling controls. The review also verified whether heat sink performance problems were adequately identified and entered into the licensee's corrective action program.

Trending analysis, test frequency, and future testing plans for the CCW Hxs were discussed with the system engineer responsible for monitoring heat exchanger performance. Additionally, visual inspections of the initial opening and post-cleaning of the CCW Hxs were performed by the inspectors during RF-14. Specific documents reviewed related to bio-fouling monitoring and other documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

.1 Inservice Inspection (ISI)

a. Inspection Scope

The inspectors observed in-process ISI work activities, reviewed ISI procedures, and reviewed selected ISI records. The observations and records were compared to the TS and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1989 Edition, no Addenda, to verify compliance. Ultrasonic examinations (UT) on weld numbers 1-4302-2 and 1-4302-3 on the residual heat removal inlet to hot leg piping were observed.

Qualification and certification records for examiners, and equipment for selected examination activities were reviewed. In addition, samples of ISI issues in the licensee's corrective action program were reviewed for adequacy. Work records documenting results of inservice inspection activities were also reviewed. The records were compared to the TS, license amendments and applicable industry established performance criteria to verify compliance.

b. Findings

No findings of significance were identified.

.2 Repair and Replacement Activities

a. Inspection Scope

The inspectors observed in-process work activities, and reviewed work procedures and records for repair and/or replacement of components covered by the ASME Section XI Code. The inspectors performed a review of records documenting repairs to the seal injection piping on the C reactor coolant pump (RCP). A through wall leak was discovered in this piping while performing the boric acid inspection program at the start of the RF-14. This condition was documented in licensee event report (LER) 50-395/2003-004-00, Reactor Coolant Pump Seal Injection Nozzle Leakage. Records examined included a chronology of previous leaks and repairs to seal injection piping on all three RCPs (A, B, & C), non-destructive examination results (liquid penetrant and UT), nonconformance notice (NCN) 03-3436, repair procedures, 900281-01 and 900281-02, and repair records. The repair records included material certifications for the replacement spool piece and nozzle assembly and weld filler material, weld process travelers, weld procedure qualification record, welder and inspector qualification records, and results of visual inspection and liquid penetrant examination of the completed welds. The inspectors examined the completed repairs and observed UT of the new nozzle assembly and completed welds performed by licensee ISI personnel using a specialized technique established to detect flaws in these types of configurations. The inspectors also reviewed records documenting repair and replacement of a three-inch service water system valve, XVG03142B.

b. Findings

No findings of significance were identified with the repair and replacement activities. Refer to Section 4OA3.2, Event Followup, for LER 50-395/2003-004-00 closeout and additional discussion related to seal injection nozzle leakage on the C RCP.

.3 IWE Containment Vessel Inspection

a. Inspection Scope

The inspectors examined interior portions of the concrete containment building and reviewed selected records. The observations and records were compared to the TS, ASME Boiler and Pressure Vessel Code, Article IWE of Section XI, 1992 Edition and 1992 Addenda, and 10 CFR 50.55a. The inspectors examined the interior surfaces of the containment liner and the moisture barrier at the intersection of the liner and interior concrete floor area. The inspectors also reviewed records documenting visual inspections performed on the containment building in July through November 2000, and augmented inspections performed to examine the tendon access gallery, RHR and spray guard pipes, and the containment liner and moisture barrier during the April 2002, and October 2003 refueling outages to satisfy applicable requirements of the TS and ASME Section XI.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On December 17, 2003, the inspectors observed senior reactor operators' and reactor operators' performance on the plant simulator during licensed operator requalification training. The training scenario involved a failure of pressurizer level transmitter LT-459 resulting in letdown isolation followed by a reactor trip with two stuck rods and a loss of coolant accident (LOR-SA-023). The inspectors reviewed written examination material and job performance measures given to an operator as part of operator license reactivation. The inspectors verified that training included risk-significant operator actions, implementation of emergency classification and the emergency plan. The inspectors assessed overall crew performance, communication, oversight of supervision and the evaluator's critique. The inspectors verified training issues were appropriately captured in corrective action program.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the equipment issues described in the CERs listed below, the inspectors evaluated the licensee's effectiveness of the corresponding preventive or corrective maintenance associated with structures, systems or components (SSCs). The inspectors reviewed maintenance rule (MR) implementation to verify that component and equipment failures were identified, entered, and scoped within the Rule. Selected SSCs were reviewed to verify proper categorization and classification as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors examined (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. Inspectors reviewed the licensee's controlling procedures 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.

- CER 0-C-03-1617, criteria reset for instrument air compressor due to failure of pressure controller which impacted the capability of the atmospheric discharge bypass valve from operating;
- CER 0-C-03-2348, LCV-115D failed to stroke due to tripped thermal overloads affecting ability to provide a flow path for the B train of high head safety injection.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's assessments of the risk impacts of removing from service those components associated with emergent work items. The inspectors evaluated the selected SSCs listed below for: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (4) that emergent work problems were adequately identified and resolved. The inspectors evaluated the licensee's work prioritization and risk determination to determine, as appropriate, whether necessary steps were properly planned, controlled, and executed for the planned and emergent work activities listed below:

- A train EDG 24-hour run per STP-125.008, diesel considered available per station order allowing equipment out of service (EOOS) monitoring program to remain green versus yellow during testing;
- Review of shutdown risk and contingency plan for reactor coolant system (RCS) inventory at nine inches below the reactor vessel flange;
- Shutdown risk to the plant regarding protecting the spent fuel pool cooling power supply while solar magnetic field event occurred;
- CER 0-C-03-3073, modification to snubber RHH4006 being performed during 72-hour TS action statement that did not account for cure time of baseplate grout;
- CER 0-C-03-4348, turbine driven emergency feedwater pump unable to control speed during surveillance test STP-220.002, causing emergent work and EOOS risk assessment adjustment.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolution

a. Inspection Scope

This inspection evaluated operators' preparations and response for the listed non-routine plant evolution to ensure they were appropriate and in accordance with the required procedures. The inspectors also evaluated performance and equipment problems to ensure that they were entered into the corrective action program.

- Failure of RHR letdown valve HCV00142 during refueling activities that required operator action in accordance with Abnormal Operating Procedure (AOP)-115.4, “Loss of RHR While Refueling,” (reference CER 0-C-03-3497).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected 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) where compensatory measures were involved, whether the compensatory measures were in place, would work as intended, and were appropriately controlled; and (5) the impact on TS limiting conditions for operations and the risk significance in accordance with the SDP. The inspectors reviewed the following five CERs / NCNs, issues and evaluations:

- CER 0-C-02-3174 and Technical Work Record LC-13109, for Westinghouse Letter 99-CG-G-0032, fuel protective grid flow passages smaller than sump screen dimensions;
- CER 0-C-03-3130, A EDG fuel oil storage tank level during 24-hour EDG run;
- CER 0-C-03-4094, reactor building sump screens identified deficient conditions including proper screen installation;
- NCN 03-3099 and CER 0-C-03-4288, train A K507 solid state protection relay failure; 10 CFR Part 21 report issued and associated LER;
- NCN 03-3500, A EDG fuel supply pipe broken at the threads; use of fire service water instead of service water during maintenance runs.

b. Findings

No findings of significance were identified. Closeout of the LER 50-395/2003-005-00 for NCN 03-3099 is documented in Section 4OA3.3, Event Followup.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated design change packages for one modification ECR, to evaluate the modification for adverse effects on system availability, reliability, and functional capability. The modification and the associated attributes reviewed are as follows:

ECR-50453, installation of service water discharge piping relief valve XVR-13142-SW;

- Materials/components compatibility, functionality and consistency with design bases;
- Field installation;
- Post modification performance;
- Plant procedure, critical drawing, design basis information, FSAR updating.

For the selected modification package, the inspectors observed the as-built configuration. Documents reviewed included procedures, engineering calculations, modifications design and implementation packages, work orders (WO), site drawings, corrective action documents, applicable sections of the living FSAR, supporting analyses, TSs, and design basis information.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (PMT)

a. Inspection Scope

For the post-maintenance tests listed below, the inspectors reviewed the test procedure and witnessed either the testing and/or reviewed test records to determine whether the scope of testing adequately verified that the work performed was correctly completed and demonstrated that the affected equipment was functional and operable:

- MWR 010007 and CER 0-C-03-3106, following regulator replacement for control room ventilation A train PMT per STP-124.003;
- MWR 215562, repack main steam isolation valve XVM-2801A-MS and retest per STP-130.004D, “Main Steam Isolation Full Stroke Test;”
- MWR 216221 following turbine disassembly and inspection and retested per STP-222.002, “Turbine Driven Emergency Feedwater Pump and Valve Test,” (Sections 6.1-4 and 6.8, 6.9 manual actuation start / inservice testing);
- MWR 314493, rework of B train component cooling water outlet pipe to reactor coolant pump XPP0030B per NCN 03-3693;
- MWR 315031, recalibration of Agastat relays on B train EDG and retest per STP-125-018, “Diesel Generator B Loss of Offsite Power Test;”
- MWR 316298, troubleshooting and retest of digital rod position indication card to address failure of control rod M4 to move with the control bank.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The unit began a refueling outage on October 11 which ended on November 26. 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 control plan for the RF-14 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 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 commensurate with the outage risk control plan for key safety functions and applicable technical specifications, and that configuration changes due to emergent work and unexpected conditions were controlled in accordance with the outage risk control plan.

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

During the outage, the inspectors:

- Reviewed RCS pressure, level, and temperature instruments to verify that those instruments were installed and configured to provide accurate indication and that instrumentation error was accounted for;
- Reviewed 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 safety and if they were consistent with the licensee’s outage risk control plan assumptions;
- Observed 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 full core offload. The inspectors also compared these operations to FSAR commitments and TS requirements;
- Observed licensee control of containment penetrations to verify that the licensee controlled those penetrations in accordance with the refueling operations TSs and could achieve containment closure for required conditions;
- The inspectors examined the spaces and cubicles inside the reactor building prior to reactor startup to verify that debris had not been left which could affect performance of the containment sumps.

The inspectors also reviewed the following activities related to RF-14 for conformance to applicable procedural and TS requirements:

- monitoring of shutdown activities;
- decay heat system operations;
- inventory control and measures to provide alternative means for inventory addition, including during conditions of reduced inventory;
- reactivity controls including locked valve dilution controls;
- refuel handling operations (inspection, insertion, and tracking of fuel assemblies through core reload);
- 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 refueling outage activities at an appropriate threshold and entering them in the corrective action program. The CERs that were specifically reviewed by the inspectors are listed below. The CERs identified below were initiated during the refueling outage and were considered significant.

- 0-C-03-3099, During STP-345.037 the master relay K507 did not actuate, NCN and 10 CFR 21 Part 21 notices, and LER 50-395/2003-005-00 issued;
- 0-C-03-3196, Steam generator power operated relief valves were sluggish to respond to increasing RCS temperature during plant shutdown with RCS heat-up of approximately 55 degree F per hour; resulted in B main steam safety valve lifting;
- 0-C-03-3201, Control rods F-8 and E-11 failed to withdraw from the bottom, during rod testing these rods did not move when their respective banks were pulled out;
- 0-C-03-3208, Intermediate Range Channel N-35 spiking during plant shutdown, detector declared inoperable;
- 0-C-03-3634, Reactor coolant pump seal injection nozzle leakage discovered, LER 50-395/2003-004-00 issued;
- 0-C-03-3822, Following core barrel replacement and RHR system was restarted the refueling cavity clarity degraded significantly;
- 0-C-03-3988, During performance of STP-125.017, upon initial breaker closure, the bus voltage dropped below the 5400 volts TS required voltage, operability review and extent of condition review required;
- 0-C-03-3989, During performance of STP-125.017 reactor building sump leakage increased by approximately three gallons per minute (gpm), suspect reactor building cooling units service water relief valve lifted;
- 0-C-03-4172, Digitial rod position indication failure of both channels resulted in a manual sub-critical reactor trip being performed.

b. Findings

No findings of significance were identified. LERs as identified above are closed in Section 4OA3, Event Followup.

1R22 Surveillance Testing

a. Inspection Scope

For the six surveillance tests listed below, the inspectors examined the test procedure and either witnessed the testing and/or reviewed test records to determine whether the scope of testing adequately demonstrated that the affected equipment was functional and operable:

- STP-122.003, “Component Cooling Train B Valve Operability Test;”
- STP-125.008, “Diesel Generator A Refueling Operability Test;”
- STP-125.011, “Integrated Safeguards Guards Testing,” train B;
- STP-215.001B, “Reactor Building Personnel Escape Airlock Test;” including 30 month containment escape airlock pressurization;
- STP-230.006, “ECCS / Charging Pump Operability Testing (Refueling);”
- STP-401.002, “Main Steam Line Code Safety Valves ASME Section XI Test,” (including XVS-2806 F, G and H).

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification and related CER to assess the impact on risk-significant SSC parameters, such as, availability, reliability and functional capability. The inspectors verified the temporary modification had not adversely affected safety function of the required system.

- Tygon hose and temporary level indication system during drain down nine inches below the flange; CER 0-C-03-3430, inaccuracies in Mansell level monitor system.

b. Findings

No findings of significance were identified.

2OS1 Access Controls To Radiologically Significant Areas (71121.01)

a. Inspection Scope

Access Controls. Licensee program activities for monitoring workers and controlling access to radiologically-significant areas and tasks were inspected. The inspector evaluated procedural guidance, directly observed implementation of administrative and established physical controls, assessed worker exposures to radiation and radioactive material, and appraised radiation worker and technician knowledge of, and proficiency in implementing Radiation Protection (RP) program activities.

During the onsite inspection, radiological controls for refueling outage activities were observed and discussed. The inspector identified five jobs that were expected to have the highest cumulative radiation exposure and reviewed the associated ALARA packages. This review included incorporation of industry experience, the use of temporary shielding, airborne radioactivity and contamination controls. The Radiation Work Permits (RWPs) were reviewed by the inspector for consistency with the planning documentation and logical task breakdown. The inspector discussed with ALARA and Chemistry supervision Plant collective exposure trends and source terms.

Occupational workers' adherence to selected RWPs and Health Physics Technician proficiency in providing job coverage were evaluated by the inspector through direct observations, review of selected exposure records and investigations, and interviews with licensee staff. Occupational exposure data associated with direct radiation, potential radioactive material intakes, and from discrete radioactive particles were reviewed and assessed independently.

RP program activities were evaluated by the inspector against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, and J; Updated Final Safety Analysis Report (UFSAR) details in Section11, Waste Disposal and Radiation Protection System, UFSAR Section 12.3 Health Physics Program and approved licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in Section 2OS1 of the report Attachment.

Condition Evaluation Reports. Licensee CER Reports associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with responsible licensee representatives. The inspector assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with licensee procedures listed in the report Attachment.

b. Findings

There were no findings of significance.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Reactor Safety Cornerstone

a. Inspection Scope

To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element. The inspectors reviewed a selection of station logs, RCS leak rate surveillance procedures test procedures, TS requirements, RCS activity surveillance test records, computer trend data, LERs, power history curves, corrective action program database, the monthly operating reports, and PI data sheets to verify data reported. In addition, the inspectors also interviewed licensee personnel associated with the PI data collection, evaluation and distribution. During the inspection period the inspectors also observed the performance of the surveillance activity (STP-144.02, “Operational Leakage Test”) that determined RCS identified leak rate to verify reported data accuracy. The inspectors sampled data for the following three PIs:

- Unplanned Power Changes, (Cornerstone: Initiating Events) - data reviewed for the period of September 2002 through September 2003;
- Reactor Coolant System Leak Rate, (Cornerstone: Barrier Integrity) - data reviewed for the period of September 2002 through September 2003;
- Reactor Coolant System Activity, (Cornerstone: Barrier Integrity) - data reviewed for the period of September 2002 through September 2003.

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety Cornerstone and Public Radiation Safety Cornerstones

a. Inspection Scope

The inspector sampled licensee November 1, 2002 through September 30, 2003 submittals for the Performance Indicator (PI) criteria listed below for the period from November 2002 through September 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Rev. 1, were used to verify the basis in reporting for each data element.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness PI

Listings of Plant Issues were reviewed to determine if any appeared to exceed the reporting threshold for occupational radiation safety performance indicators. The inspector interviewed the personnel responsible for initial screening of plant issues for performance indicator reporting. The interview was to determine the criteria that would have to be exceeded before a plant issue was flagged as a performance indicator hit. The screening process was discussed in detail.

Public Radiation Safety Cornerstone

- Radiological Control Effluent Release Occurrence PI

For the review period, the inspector reviewed data reported to the NRC and evaluated selected radiological liquid and gaseous effluent release data, out-of-service process radiation monitor and compensatory sampling data, abnormal release results, and selected condition reports records documented in the Sections 2OS1, and 4OA1 of the report Attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 NRC-identified issue with a Technical Specification Information / Relocation (TSR)

a. Inspection Scope

CER 0-C-03-3179, described as “TSR-1020 exceeds the intended scope of a TS,” was selected for review. The inspectors reviewed the licensee’s actions to address an NRC-identified issue with a TS interpretation, TSR-1020, Revision 3, which interpreted TS 3.7.9, “Area Temperature Monitoring” limiting condition for operation (LCO) action statement. The inspectors conducted a review of the licensee’s problem identification and resolution activities to ensure they included:

- Complete and accurate identification of the problem in a timely manner commensurate with its significance;
- Evaluation and disposition of performance issues associated with maintenance effectiveness, including maintenance practices, work controls and risk assessment;
- Consideration of extent of condition, common cause and previous occurrences;
- Identification of root and contributing causes of the problem;
- Identification of corrective actions which are appropriately focused to correct the problem;
- Completion of corrective actions in a timely manner commensurate with the safety significance of the issue.

b. Findings and Observations

No findings of significance were identified. However, inspectors raised issues with TSR-1020, specifically that it could allow inappropriate removal of both trains of Service Water Pump House (SWPH) ventilation without a specified time limit. The TSR was not removed from use or modified consistent with the scope of TS 3.7.9 in a timely fashion. This TSR could have resulted in support equipment (SWPH ventilation) not being available for post-accident conditions.

During the previous inspection period, on August 22, 2003, the inspectors identified an issue with licensee’s TSR-1020 which exceeded the intended scope of the TS 3.7.9. This TS interpretation as issued allowed for removal of both trains of SWPH ventilation equipment for maintenance without time restriction and without a supporting risk assessment per 10 CFR 50.65. For this TS, room temperature area monitoring was limited to 102 degrees Fahrenheit (° F) to ensure the equipment can meet design requirements. The TS basis was established to ensure equipment qualification and long term operability. TSR-1020 exceeded that scope because it provided an unlimited SWPH ventilation system outage time.

Because the licensee did not implement a dual train SWPH ventilation equipment outage during the inspection period (or at any time in the last two years based on inspector reviews), no violation of NRC regulatory requirements was identified. The significance of the issue was minimized because additional temperature monitoring and compensatory actions are directed by the TSR-1020 that would have reduced the potential risk to the service water mitigating system had the TSR been implemented (for a dual train SWPH ventilation outage). Additionally, 10 CFR 50.65 “requirements for monitoring the effectiveness of maintenance at nuclear power plants,” requires before performing maintenance activities the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities, which would apply to any dual train SWPH ventilation maintenance outage. Corrective actions taken or planned by the licensee for this issue has been entered into the licensee’s corrective action program under CER 0-C-03-3179.

.2 Daily and Semi-Annual Review of the Licensee’s Correction Action Program (CAP)

a. Inspection Scope

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 attending CER screening meetings and accessing the licensee’s computerized database.

The inspectors also performed a semi-annual review of the licensee’s CAP to access trends that might indicate the existence of more significant safety issues. This semi-annual review included a review of the licensee’s system health reports, self assessment reports, and the CER data base.

Plant issues associated with mis-positioning events reviewed by the inspectors included the following CERs:

- 0-C-03-4170, XVT02813-MS steam supply header drain trap isolation valve found mis-positioned during Turbine Driven Emergency Feedwater Pump post maintenance run resulting in exhaust boot ripping with steam entering the room;
- 0-C-03-4258, two valves on the auxiliary condenser were mis-positioned resulting in unexpected auxiliary condenser pressure increase when A Feedwater pump was started; A feedwater pump had to be secured;
- 0-C-03-4264, valve XVA11508-SC for main generator stator cooling was found a mis-positioned, normally open valve found closed;
- 0-C-03-4273, chill water valve mis-positioning resulted in high room temperature, A CRDM room cooling fan found running with chill water aligned to the B fan.

b. Findings and Observations

There were no findings of significance identified. However, the inspectors observed that there was an increase in the number of mis-positioning events that occurred in operations near the end of the refueling outage due mainly to human performance problems. The inspectors also observed that the licensee was pro-actively taking measures to resolve the mis-positioning problems including the use of operations shift stand-downs. The inspectors noted that the licensee had not generated a negative trend assessment CER in accordance with SAP-1131, "Corrective Action Program," Enclosure 7.6. The licensee was responsive to the inspectors observation and initiated a trending CER.

4OA3 Event Followup

.1 (Closed) LER 50-395/2003-003-00: Control Room Ventilation Boundary Breached During Maintenance

The inspectors reviewed the subject LER and CER 0-C-03-2819 to assess the cause and licensee actions taken for discovery of control room ventilation boundary being breached during maintenance activities on September 8, 2003. This issue had previously been reviewed under Sections 1R15 and 4OA7.4 of NRC IR 05000395/2003004 and a licensee-identified violation of very low significance (Green) was documented.

The licensee is still completing corrective actions including root cause review of this condition. The licensee currently plans to submit a LER supplement to document the additional corrective actions. No new issues or additional findings of significance were identified during the LER closeout review. The ongoing root cause and corrective actions are being tracked by CER 0-C-03-2819.

.2 (Closed) LER 50-395/2003-004-00: Reactor Coolant Pump Seal Injection Nozzle Leakage

The inspectors reviewed the subject LER and CER 0-C-03-3436 to assess the cause and licensee actions taken following discovery of RCP seal injection nozzle leakage on October 19, 2003. A licensee-identified violation involving reactor coolant pressure boundary leakage while in Mode 1 was reviewed. The inspectors reviewed the corrective actions and repairs to verify they were reasonable to address this issue. The inspectors assessed the condition for any other potential violations and safety significance.

On October 19, during a refueling outage (Mode 6) the licensee discovered boron around the seal injection nozzle and on the thermal barrier casing of the C reactor coolant pump. The licensee reported this condition under Event Notification EN # 40261 on October 20 in accordance with 10 CFR 50.72(b)(3)(ii)(A). Review of the corrective actions and repairs were completed under Section 1R08, Inservice Inspection Activities, of this report. No findings of significance with the nozzle repairs were identified.

Preliminary determination for the root cause of the leak indicates it was likely a result of overloading or mechanical vibration. Based on the licensee's investigation, they determined that pressure boundary leakage existed while in Mode 1 (power operation) prior to its discovery in RF-14. This condition affected the Barrier Integrity cornerstone (pressure boundary leakage) and potentially could affect the Initiating Event cornerstone (a unit transient initiator). The issue is more than minor because if left uncorrected it would become a more significant safety concern due to greater pressure boundary leakage and resulting in a plant transient or a reactor trip. There was minimal consequence to this condition, however, because the RCS leak rates during the previous cycle were well below the unidentified leakage TS limit of 1.0 gpm. Typical leak rates during the past cycle were less than 0.2 gpm.

The potential safety consequence of this nozzle leakage was evaluated by the inspectors and a regional Senior Reactor Analyst and determined to have very low safety significance. This conclusion is based on assuming when the small leak increased to greater than 1.0 gpm then the daily RCS leak rates would have detected the leakage and TSs would have directed a normal plant shutdown (transient initiator). Assuming the one and half inch nozzle line had failed completely, then the resulting leak would be bounded by FSAR accident analysis and previous Westinghouse analysis which indicates a leak of less than charging pump capacity would occur. This would also result in a unit shutdown or a reactor trip (transient initiator) in accordance with annunciator response and abnormal response procedures. The Phase II Significance Determination Process Notebook was the tool used with the Transient worksheet completed. The resulting numerical values indicated the violation was of very low safety significance (< 1E-6).

During power operation (Mode 1), TS 3.4.6.2 requires no pressure boundary leakage. The licensee assessed this condition in NCN 03-3436 and CER 0-C-03-3511 which stated that the leak existed for more than six hours while in Mode 1 prior to discovery and, therefore constituted a violation of TS. This issue has been entered into the licensee's corrective action program under NCN 03-3436, CER 0-C-03-3511 and the subject LER. A "boat" sample of the failed weld was removed for further metallurgical analysis. Root cause of this condition will be determined based on metallurgical analysis and provided in an LER supplement. For enforcement disposition of this issue see Section 4OA7.1.

.3 <u>(Closed) LER 50-395/2003-005-00</u>: Failure of Master Relay K507 During Solid State Protection System Testing

The inspectors reviewed the subject LER and NCN 03-3099 to assess the cause and licensee actions taken for discovery of a failure of Master Relay K507 discovered during solid state protection system (SSPS) testing on October 3, 2003. The licensee at that time appropriately declared the A train SSPS out of service and replaced the failed relay in the TS allowed outage time. Root cause of the relay failure analysis later concluded the failure was due to an open circuit in the relay coil lead wire resulting from corrosion caused by off-gassing of chlorine contamination from the paper used to wrap the coil, or the adhesive used to attach the paper to the coil. An additional relay in A train also showed evidence of the same type of corrosion but not to the point of failure. Therefore, this failure was considered to be a manufacturing defect in a basic component, which could cause a loss of safety function. The licensee appropriately reported this event in accordance with 10 CFR 50.73 (a)(2)(vii) and 10 CFR 21.21 for a potential significant safety hazard.

The licensee replaced all potentially affected relays with a different manufacturer's relay (samples of which were also tested and shown to free of the defect). The redundant train relay had a different type (manufactured) relay and was operable. The manufacturer of the subject master relay (MidTex) is no longer in business. Failure of the subject relay was determined by the licensee to have occurred at the time of SSPS testing. No performance deficiency on the part of the licensee was identified by this review. No findings of significance or violations of NRC regulatory requirements were identified.

4OA5 Other Activities

.1 Visual Inspection of Plant Systems, Structures, and Components in Containment - Licensee Renewal Inspections

a. Inspection Scope

The inspectors performed visual inspections of the interior of the Virgil C. Summer containment on October 29 and 30, 2003, during RF-14. This included observation of accessible portions of plant systems, structures, components, instrumentation lines, and electrical cables inside the containment to observe material condition and inspect for aging conditions that might not have been previously recognized and addressed in the License Renewal Application. The following is a partial list of equipment observed:

Main steam and feedwater systems pipe supports;
Personnel and equipment hatches;
Steam generators “A”, “B”, and “C” supports;
Reactor building spray headers and piping;
Ventilation ducting;
Electrical cables and supports;
Instrumentation lines, instrumentation, and supports;
“A”, “B”, and “C” reactor coolant pump cubicles / loop rooms;
Containment electrical penetrations;
Reactor coolant drain collection tank and piping;
Pressurizer relief tank;
Containment pipe penetration area;
Containment sump area;
Containment liner and coatings.

b. Findings

The observations of general material conditions included: inspection of piping components for evidence of leaks or corrosion, inspection of coatings (piping, tanks, and structural components), and inspection of electrical cables and instrumentation lines for indications of deterioration. The material condition at Virgil C. Summer was good and no significant aging management issues were identified. The inspection of the application for licensee renewal was documented in NRC Inspection Report (IR) 05000395/2003009.

.2 (Closed) Temporary Instruction (TI) 2515/150, Revision 2, “Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009)”

a. Inspection Scope

The inspectors observed activities relative to inspection of the Unit 1 reactor pressure vessel (RPV) head penetrations in response to NRC Bulletin 2002-02 and Order EA-03-009. The guidelines and criteria for the inspection were provided in NRC TI 2515/150, “Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Bulletin 2002-02).” The program review included observations of portions of the remotely monitored examinations, review of qualifications for examination personnel, and review of licensee procedures and documents.

b. Findings and Observations

No findings of significance were identified. All inspection activities associated with TI 2515/150 are complete. Specific questions contained within the TI are discussed below.

1. For each of the examination methods used during the outage, was the examination:

   a. Performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

      The examination was performed by qualified and knowledgeable individuals that were qualified as Level II non-destructive Visual Testing (VT)-1, 2 and 3 examiners. The vendor performing the remote camera work, Everest Visual Inspection Technology were Level III VT-1, 2, and 3 qualified.

   b. Performed in accordance with demonstrated procedures?

      The examination was performed per Quality Systems Procedure (QSP)-216, Revision 0, “Boric Acid Corrosion Inspection.” The results were documented in a report and CER 0-C-03-3615. The inspectors also reviewed QSP-505, “Visual Examination,” which was the governing procedure for station and vendor personnel for the reactor vessel head inspection. Performance of the visual VT-2 examination was conducted in accordance with established procedures.

c. Able to identify, disposition, and resolve deficiencies?

The examination was able to adequately identify, characterize and resolve deficiencies.

d. Capable of identifying the primary water stress corrosion cracking (PWSCC) and/or RPV head corrosion phenomena described in Order EA-03-009?

The examination was capable of identifying the corrosion phenomena described in the order. In instances where boron residue was identified the licensee was able to ascertain that the source was outside of the insulation at the top of the control rod drive mechanism (CRDM) housings flowing downward towards the vessel head. No indications were noted that would be representative of PWSCC and/or RPV head corrosion phenomenon.

2. What was the physical condition of the reactor vessel head (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

The upper head was generally clean with five nozzles (#27, 47, 51, 55, and 59) identified that exhibited dried boron residue from conoseals, vessel head vent assembly and/or reactor vessel level instrumentation connections. For the CRDM penetrations identified the respective areas were cleaned. Most small debris identified on the head was removed, however some materials remained and were considered by the licensee to pose no threat to the integrity of the head. There were no viewing obstructions impeding the inspection. There was no evidence of leakage, corrosion or other degradation originating from the reactor head surface.

3. Could small boron deposits, as described in the Bulletin 2001-01, be identified and characterized?

The examination was capable of identifying small boron deposits; however, none were identified that could be attributed to leaks in the areas of concern.

4. What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?
No CRDM or the reactor head vent penetration required repair. In addition, there were no material deficiencies requiring any immediate repairs. The dried boron residue found on nozzles # 27, 27, 51, 55, and 59 were documented in the licensee's corrective action program under CER 0-C-03-3615 and cleaned revealing no deficiencies.

5. What, if any, impediments to effective examinations, for each of the applied methods, were identified (e.g., centering rings, insulation, thermal sleeves, instrumentation, nozzle distortion)?

VT-2 examination for the detection of boron leakage was not limited by any impediments.

6. What was the basis for the temperatures used in the susceptibility ranking calculation, were they plant-specific measurements, generic calculations (e.g., thermal hydraulic modeling, instrument uncertainties), etc.?

   Virgil C. Summer (VCS) used a Westinghouse generic calculation for the basis of their susceptibility ranking. This calculation assumed that the RPV head temperature has been a constant 557° F for the life of the plant. VCS does not have installed instrumentation to directly measure RPV head temperature. It is assumed that RPV head temperature is equal to the hottest cold leg temperature, which is conservatively assumed to be the best estimate zero load RCS temperature. WCAP 13714, "Replacement Steam Generator/Uprating Engineering Report, June 1994 (Section 2.0 and Table 2.0-1), stated that the best estimate zero load RCS temperature is 557° F. This value was validated against the plant computer data available for the period between 1996 through 2003 which showed that the average cold leg temperature was below 557° F (reference Virgil C. Summer susceptibility ranking calculation, DC04010-001, Rev. 0, Attachment III).

7. During non-visual examinations, was the disposition of indications consistent with the guidance provided in Appendix D of this TI? If not, was a more restrictive flaw evaluation guidance used?

   Not applicable, all examinations were done visually. No other techniques were used.

8. Did procedures exist to identify potential boric acid leaks from pressure-retaining components above the RPV head?

   The licensee has QSP-216, “Boric Acid Corrosion Inspection,” and QSP-505, “Visual Examination,” in place to identify and evaluate any boric acid leaks above the RPV head.

9. Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components above the RPV head?

   The licensee did not performed follow-on examinations (e.g., dye penetrant testing or other inspections techniques) on nozzles identified with dried boron resin. The licensee concluded that all leakage was from sources other than the CRDMs or reactor head vent penetration. There was no evidence of leakage, corrosion or other degradation originating from the reactor head surface.

.3 <u>(Closed) TI 2515/152, Revision 1, “Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02)”</u>

a. <u>Inspection Scope</u>

The inspectors observed activities relative to inspection of the Unit 1 RPV lower head penetrations in response to NRC Bulletin 2003-02. The guidelines and criteria for the inspection were provided in NRC temporary inspection (TI) procedure TI 2515/152, Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2002-02).

b. Findings and Observations

No findings of significance were identified. All inspection activities associated with TI 2515/152 are complete. Specific questions contained within the TI are discussed below.

1. For each of the examination methods used during the outage, was the examination:

   a. Performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

      The examination was performed by qualified and knowledgeable individuals that were qualified as Level II non-destructive Visual Testing (VT)-1, 2 and 3 examiners. The vendor performing the remote camera work, Everest Visual Inspection Technology were Level III VT-1, 2, and 3 qualified.

   b. Performed in accordance with demonstrated procedures?

      The examination was performed per Quality Systems Procedure (QSP)-216, Revision 0, “Boric Acid Corrosion Inspection.” The results were documented in CER 0-C-03-3313. The inspectors also reviewed QSP-505, “Visual Examination,” which was the governing procedure for station and vendor personnel for the reactor vessel head inspection. Performance of the visual VT-2 examination was conducted in accordance with established procedures.

   c. Able to identify, disposition, and resolve deficiencies?

      The examination was adequately completed using a robotic camera with resolution that met the visual acuity as defined by IWE/IWL specifications, ASME Section XI Code and QSP-216. There were no deficiencies identified. The licensee provided copies of the video tapes recorded during the examination which were reviewed by the resident inspectors, in addition to, observations during portions of the actual examination.

   d. Capable of identifying pressure boundary leakage as described in the bulletin and/or RPV lower head corrosion?

      The examination was capable of identifying pressure boundary leakage as described in the bulletin and/or RPV lower head corrosion.

2. Could small boric acid deposits representing RCS leakage, as described in the Bulletin 2003-02, be identified and characterized, if present by the visual examination method used?

The examination was adequate to satisfy the Bulletin requirements. Boric acid deposits identified were dispositioned as “wash-down” from leakage of the RCS A train hot leg pipe to vessel weld originally identified during refueling outage RF-12 (October 2000). Inspections of that repair effort are documented in NRC IR 05000395/2000008.

3. How was the visual inspection conducted (e.g., with video camera or direct visual by the examination personnel)?

   All examinations were conducted via remote video camera.

4. How complete was the coverage (e.g., 360° around the circumference of all the nozzles)?

   The vendor conducting the examination, Everest Visual Inspection Technology, was able to attain 360° coverage around the circumference of all lower head nozzles.

5. What was the physical condition of the RPV lower head (e.g., debris, insulation, dirt, deposits from other sources, physical layout, viewing obstructions)? Did it appear that there are any boric acid deposits at the interface between the vessel and the penetrations?

   The examination revealed a small amount of boron deposits, rust and associated stains over portions of the bottom head. The licensee dispositioned the boron and rust as a result of the water introduced during “wash-down” activities to remove boron buildup below the A train hot leg pipe between the vessel wall and the vessel. Otherwise, the general physical condition of the RPV lower head area was generally clean with no indications of boric acid leaks from other sources. A platform of mirror insulation several feet below the lower head provided a level surface for the robotic camera to perform the survey. There were no obstructions or obstacles in the licensee completing a comprehensive examination of the lower head. There was a minor collection of dust on the platform with a few loose ‘balls’ of metal wool insulation used to fill the annulus area between the incore tubes and mirror platform. The licensee placed this issue of the boron deposits in their corrective action program under CER 0-C-03-3313 with actions to address where boron had built up around the instrument tube due to the wash-down effect.

6. What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

   The examination determined that there were no identified deficiencies requiring repair. The examination did identify light to moderate staining and surface rust seen around the nozzle annular areas and on the bare head surface at the instrument tube interfaces. The areas where rust and boron had built up were subsequently cleaned and reinspected with no evidence of degradation or evidence of leakage originating from the reactor lower head surface or instrument interface.

7. What, if any, impediments to effective examinations, for each of the applied non-destructive examination methods, were identified (e.g., insulation, instrumentation, nozzle distortion)?

   The licensee had to remove some mirror insulation to gain access to the lower head area for the robotic camera and to stage a stationary camera. However, a full examination of the nozzles was still accomplished.

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

   There were no additional follow-on examinations required due to the absence of any indications of boric acid leaks from the surface of the lower head.

9. Did the licensee take any chemical samples of the deposits? What type of chemical analysis was performed (e.g., Fourier Transform Infrared (FTIR)), what constituents were looked for (e.g., boron, lithium, specific isotopes), and what were the licensee's criteria for determining any boric acid deposits were not from the RCS leakage (e.g., Li-7, ratio of specific isotopes, etc.)?

   Yes, chemical samples were taken using an in-house protocol method modeled after the EPRI methodology developed for aging deposits found on the Bottom Mounted Instrument nozzles at the South Texas Project. Radionuclide activity was determined for each sample taken using gamma spectroscopy. Boric acid samples taken were looking for specific isotopes; Cobalt-58, Cobalt-60, Cesium-134, Cesium-137, and Zinc-65. The licensee used the ratio of Co-58/Co-60 considering that the two isotopes behave identically chemically, so the ratio of the activities should reflect the ratio of the original source decay corrected to the time of removal from the source. Cesium-134 was not detected in any sample, therefore, the Ce-134/Ce-137 ratio was not used. Based on samples collected the licensee determined that decay-corrected curve for Co-58/Co-60 versus time indicated that the samples were 2.86 to 3.04 years old. These results would correlate to the A train hot leg pipe weld crack event. Analysis of all samples collected were consistent with these results.

10. Is the licensee planning to do any cleaning of the head?

    Yes, the licensee completed implementation of a head cleaning plan during refueling outage RF-14 (October - November 2003). Areas identified were reinspected after cleaning with no evidence of leakage between the instrumentation penetration to bottom vessel head interface.

11. What are the licensee's conclusions regarding the origin of any deposits present and what is the licensee's rationale for the conclusions?

    The licensee concluded that boron deposits noted originated from the A train hot leg pipe to vessel weld crack and that there was no evidence of boric acid leakage from the bottom mounted instrumentation penetrations on the reactor vessel lower head. Based on chemistry analysis of samples taken the age of the boron deposits correlated to the occurrence of the hot leg leakage. The A train weld between the pipe to vessel weld was replaced during refueling outage RF-12.

.4 (Open) TI 2515/153, "Reactor Containment Sump Blockage (NRC Bulletin 2003-01)"

a. Inspection Scope

The inspectors performed a preliminary review of the licensee's activities in response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors (PWRs)," in accordance with NRC TI 2515/153, "Reactor Containment Sump Blockage (NRC Bulletin 2003-01)," dated October 3, 2003.

Visual inspections of the containment sumps, sump screens and flow paths were performed by the inspectors during the refueling outage. The inspectors also reviewed the licensee's completed and proposed compensatory measures submitted in accordance with Bulletin 2003-01, Option 2, to verify they have been implemented or are planned and scheduled for implementation consistent with the licensee's response.

b. Observations and Findings

No findings of significance were identified for the preliminary review conducted. Pending NRC (NRR) review of the licensee's response letter RC-03-0164, dated August 6, 2003, TI 2515/153 will remain open.

Specific inspection results for each question in the TI are discussed below.

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

   Yes, Virgil C. Summer entered a refueling outage during this past quarter on October 11 and returned to power on November 26, 2003. A containment walkdown was conducted by the licensee to quantify potential debris sources during the RFO.

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

   Not applicable, Virgil C. Summer is not currently in a RFO.

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

   Not applicable, Virgil C. Summer has entered a RFO since September 1, 2002.

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

Yes, Virgil C. Summer walkdowns conducted during RF-14 (October 11 through November 26, 2003) checked for gaps in the sumps' screened flowpath and for major obstructions in containment upstream of the sumps. Issues with sump screen gaps, sump closeout inspection and reactor building foreign material exclusion were captured in the licensee's corrective action program under CERs 0-C-03-3857, 4023, 4025, 4094, 4108, 4112, and 4143. Issues identified were corrected or evaluated as acceptable prior to returning to power.

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

Virgil C. Summer sump evaluations and analysis have not yet been completed. In response to the NRC Bulletin 2003-01, the licensee chose Option 2 and described interim compensatory measures that have been implemented or will be implemented. These efforts will attempt to reduce the risk which may be associated with the potentially degraded or nonconforming ECCS and sump recirculation functions while complex evaluations to determine compliance proceed. No specific plant modifications were included in the set of interim compensatory actions to address potential recirculation performance issues.

The licensee was prepared to make sump screen related repairs or minor modifications based on the inspection results identified during RF-14. The licensee did make modifications to the sump design for air venting under ECR-50527, RB Sump Vent Addition, as a result of these inspections. Issues with long narrow openings (5 to 12 inches long by 1/8 to 1/4 inch wide) on the sump access hatches which were identified during the licensee inspection were repaired under NCN 03-3857. Additional issues with the sump screens were captured under CERs 0-C-03-4025, 0-C-03-4094 and evaluated or corrected.

No plans were developed by the licensee for major sump related modifications during RF-14. Depending on the results of the complex sump evaluations and anticipated further generic communications from the NRC and the industry, advanced preparations for modifications are anticipated for future refueling outages. The licensee is investigating plans to increase the sumps' screen area and modifying the design to increase the available design margin. These reviews and proposed actions are currently being tracked under CER 0-C-03-1897.

The inspectors performed visual inspections of the containment sumps, sump screens and flow paths to the sumps during the refueling outage and verified no major obstructions exist in the containment flowpath upstream of the sumps. The inspectors also performed a Mode 4 containment closeout inspection. Issues identified by the inspectors (i.e., screen fasteners without full thread engagement, missing screen hold down bar, conduit holes and foreign material exclusion) were appropriately captured in the licensee corrective action program.

The inspectors reviewed the NRC Bulletin 2003-01, Option 2, interim compensatory measures implemented or planned. These actions appeared to be being effectively implemented. The actions were reasonable with the intent to reduce the potential risk of emergency core cooling system and reactor building spray recirculation degradation. Additionally, the licensee compensatory actions included operator training on indications and potential recovery responses should sump clogging occur. No findings or violation of NRC regulatory requirements were identified. Documents reviewed by the inspectors including the condition evaluation report data base (1999-2003) for issues related to the reactor building sumps and NRC Bulletin 2003-01 are listed in the Attachment.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Jeff Archie, and other members of the licensee's staff on January 8, 2004 and on January 23, 2004. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violation

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

.1 TS 3.4.6.2, Reactor Coolant Operational Leakage, requires, in part, no pressure boundary leakage. Contrary to this requirement, the licensee identified during RF-14, on October 19, 2003 that a pressure boundary leak condition had existed while in Mode 1 and therefore constituted a violation of TS. This finding was of very low significance because had the leakage increased, the TS unidentified leakage requirements would have directed a normal unit shutdown, or the condition would have resulted in a shutdown / reactor trip bounded by the FSAR and previous Westinghouse analysis for an RCS leak less than charging pump capacity (see Section 4OA3.2). This issue was entered into the licensee corrective action program under CERs 0-C-03-3436, 0-C-03-3511 and LER 50-395/2003-004-00.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Archie, General Manager, Nuclear Plant Operations
F. Bacon, Manager, Chemistry Services
L. Blue, Manager, Health Physics Services
R. Clary, Manager, Nuclear Licensing and Operating Experience
M. Findlay, Manager, Nuclear Protection Services
M. Fowlkes, General Manager, Engineering Services
S. Furstenberg, Manager, Nuclear Operations Training
D. Gatlin, Manager, Operations
D. Goldston, Operations Superintendent
D. Lavigne, General Manager, Organization Effectiveness
K. Nettles, General Manager, Nuclear Support Services
W. Stuart, Manager, Plant Support Engineering
A. Torres, Manager, Planning / Scheduling and Project Management
R. White, Nuclear Coordinator, South Carolina Public Service Authority
S. Zarandi, Manager, Maintenance Services

NRC

W. Rogers, Region II Senior Reactor Analyst
R. Bernhard, Region II Senior Reactor Analyst
S. Sparks, Region II Senior Enforcement Specialist

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Temporary Instruction TI 2515/153	Reactor Containment Sump Blockage (NRC Bulletin 2003-01)(Section 4OA5.4)

Closed

LER 50-395/2003-003-00	Control Room Ventilation Boundary Breached During Maintenance (Section 4OA3.1)
LER 50-395/2003-004-00	Reactor Coolant Pump Seal Injection Nozzle Leakage (Section 4OA3.2)
LER 50-395/2003-005-00	Failure of Master Relay K507 During Solid State Protection System Testing (Section 4OA3.3)

TI 2515/150, Revision 2	Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009) (Section 4OA5.2)
TI 2515/152, Revision 1	Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02) (Section 4OA5.3)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section 1R04 - Equipment Alignment

FSAR Sections 8.3.1
SOP-306, “Emergency Diesel Generator”
SOP-307, “Diesel Generator Fuel Oil System”
TS Sections 3.7.1.2, 3.7.11, and 3.8.1
D-302-351, “Diesel Generator - Fuel Oil”
D-302-353, “Diesel Generator - Miscellaneous Services”

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, 3.9.3
Engineering Change Request ECR-500079, RHR Pump A - Split Coupling Retro Fit Modification package and Outage Work Script
EMP-100.002, “Emergency Installation of Cable for the RHR System,”
SOP- 115, “Residual Heat Removal,”
STP-375.001, (2,3,4), “Refueling Water Storage Tank Level Instrument ILT00990-(993), “ loop calibration procedures for the last three years
STP-395.054, (5, 6, 7), “Refueling Water Storage Tank Level Instrument ILT00990-(993) Operational Test,” for the last three years
SD-CGE-283, “Residual Heat Removal System Description”
Design Basis Documents for ECCS, RHR and CCW systems
D-302-641, “Residual Heat Removal System
D-302-693, “Safety Injection System”
CER Data Base search and review of RHR system CERs for 2003, (43 CERs reviewed)

Completed General and Surveillance Test Procedures reviewed - licensee completion dates
GTP-302, Remote Position Valve verification (for RHR system valves) - 11/18/03
STP-230.007, RHR Pump A and Check Valve full flow test - 10/3/03
STP-215.003A, Leakrate test on 8811B RHR suction from RB recirc sump - 11/2/03
STP-130.004B, RHR valve Operability Test - 11/2/03
STP-230.007, RHR B Pump and check valve full flow test - 11/3/03
STP-250.019, RHR GTP-006, Leakage Assessment -11/11/03
STP-215.008, RHR and SI (ECCS check valve testing) - 11/23/03

STP-205.004, RHR A, B Pump and Valve Operability Test - 11/3/03; 11/17/03; 12/01/03; and 12/17/03

Section 1R07 - Heat Sink Performance

Bio-fouling Reports Reviewed
Service Water Study for V.C. Summer Nuclear Station, Betz Industrial dated 6/6/88
GE Betz Inorganic Analysis Report, ID 84949.3, date 9/25/03
GE Betz Metullugical Lab Report CCW Heat Exchangers, Reference Number 2003-0284, dated 5/19/03 and Reference Number 2003- 0635, dated 9/13/03
GE Betz Microbiological Analysis, Lab ID 85806, dated 9/29/03 and 85631, dated 9/21/03
CER data base search for bio-fouling / clam issues 1999-2003
Various computer data and trending analysis for bio-fouling (clamtrol for control of clams, various chemistry levels in the service water system)

Section 2OS1 Access Control To Radiologically Significant Areas

Procedures, Guidance Documents, and Manuals

- Health Physics Procedure (HPP)-151, Use of the Radiation Work Permit and Standing Radiation Work Permit, Revision (Rev.) 8
- HPP-152, Radiation Control Area Access Control, Rev. 9
- HPP (Health Physics Procedure)-160, Control and Posting of Radiation Control Zones, Rev. 10
- HPP-401 Issuance, Termination and Use of RWPs and SRWPs, Rev. 16
- HPP-402, Radiological Survey Requirements and Controls for Reactor Building and Incore Pit Entries, Rev. 10
- HPP-403, Radiological Controls for Nuclear Work Activities, Rev. 8
- HPP-410, Health Physics Routine Surveys, Rev. 8
- HPP-413, Diving Operations, Rev. 2
- HPP-419, Electronic Dosimeter Alarm Setpoint Determination and Alarm Response Action, Rev. 0
- HPP-505, Issuance and Termination of Personnel Dosimetry, Rev. 16
- Station Administrative Procedure (SAP)-500, Health Physics Manual, Rev. 11
- SAP-1131, Corrective Action Program, Rev. 4

ALARA Packages/Radiation Work Permits (RWPs)

(ALARA Packages Consisted of All Issued Revisions to RWP, Surveys, Air Sample Data, ALARA In-Progress Reviews, Pre-Job Briefing Forms and Rosters and RP Job Turnover Sheets)
- ALARA Package: RWP 03-00045/001-4, Perform ISI to Include All Support Work (Insulation, Scaffolding, Weld Prep, Etc.), Revs. 1-4
- ALARA Package: RWP 03-00050/001, All Snubber Testing I/S RB, Rev. 1
- ALARA Package: RWP 03-00059/001, All Test Unit Activities Including STP'S 215.008, 230.006A,B,C,D &E, ALL 250 Series, Fire Service, STB 212.006, Rev. 1
- ALARA Package: RWP 03-00071/001,RVLIS, CRDM, Detension/Tension, Nozzle Covers,

Etc. (Head Work) , Rev. 1
- ALARA Package: RWP 03-00080/004, Reactor Vessel Head Boron Inspection, Revs. 1, 3, and 4

Corrective Action Program (CAP)/Primary Identification Program (PIP) Documents

- PIP 0-C-03-0075, 52 Task Observations Have Been Performed by HP Supervision and Management During the Fourth Quarter of 2002. These Observations Included Plant Walkdowns, Training Evaluations, and Both Specific and Generic Task Observations
- PIP 0-C-03-0520, Inadequate Supply of Neutron Instrumentation On Site to Support the Rx Bldg Entry
- PIP 0-C-03-0666, During Audit QA-AUD-200302 the Station Controls for Access to High, and Very High Radiation Areas Were Compared to Those Identified in RG8.38 "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants" and Two Gaps Were Identified.
- PIP 0-C-03-1353, Discrepancy Between Neutron TLD Results and Calculated Neutron Exposure.
- PIP 0-C-03-1682, Plant Transient and Fuel Leakage Resulted in an Unposted Radiation Area in AB 388-17, "B" Waste Gas Compressor Room
- PIP 0-C-03-2726, Expected 70 ft^3 of old BTRS Cation Resin (Low to No Dose). Received Only 25 ft^3 Higher Dose Resin Fines During Primary Spent Resin Transfer (PSRST to Radwaste).
- PIP 0-C-03-3583, A Worker Performing ISI Inspections Near "B" RCP Received an Uptake of Radioactive Material Which Was Discovered When He Exited the RCA.

Section 40A1 Performance Indicator Verification

Procedures

- HPP-242, Reporting NRC Performance Indicators, Rev. 0
- SAP-1131, Corrective Action Program, Rev. 4

Records

- Listings of Corrective Action Reports (PIP's) Covering Radiation Protection Related Corrective Action Documents for CY 2003.
- Monthly Performance Indicator Reports for November 2002-November 2003

Section 4OA5 - Temporary Instruction 2515/153

Condition Evaluation Reports

0-C-00-1336, mechanic dropped wrench in the RB sump;
0-C-01-0261, RB Sump grating found standing on end, not secured in Mode 3;
0-C-01-0341, CERs related to RB sump / minor insulation not being processed correctly;
0-C-02-1307, Temporary equipment ID labels hung in the reactor building;

0-C-02-3174, Debris filter bottom nozzle and protective grid on fuel may have passage dimensions smaller than those of the sump screen (Westinghouse letter 99CG-G-0032);
0-C-02-3228, NPSH for the RHR pumps during containment recirculaton design calculation does not model inlet losses and used incorrect dimensions for the reducer (no operability concern since available NPSH is 26 ft and required is 15 ft; with 11 ft of margin);
0-C-02-3284, Percentage of core cladding assumed to undergo metal-water reaction in hydrogen generation analysis may not meet regulatory guide 1.7 requirements (ECR 50461, RB scaffold storage modification affected);
0-C-03-1897, NRC Bulletin 2003-01: “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors.” Virgil C. Summer evaluation required;
0-C-03-2905, OE 16678 - Potential ECCS Sump Blockage due to Boron Accumulation;
0-C-03-3191, NRC Bulletin 2003-01 response contained commitments for future actions that were not captured in the CER 0-C-03-1897;
0-C-03-3639, Prior to core reload a flow path from the RB spray sump to the RWST via one RB Spray pump with its power supply breaker racked down should be available for outage defense in depth contingencies;
0-C-03-3857, Inspection of the RB recirculation sumps identified openings along the hinge side of the sump access hatches. The openings are 5 inches and 12 inches long(1/8 to 1/4 inch wide) on each access hatch, non-conformance notice;
0-C-03-3869, Grout at the base plates for the scaffold rack installed in the reactor building per ECR 50461 has cracked;
0-C-03-3942, Contamination detected both internal and external on worker who had been working in the RHR sump;
0-C-03-4023, Various articles found in the A RHR sump while performing cleanliness inspection (including an large FME barrier still taped on the suction pipe);
0-C-03-4025, Inspection of the RHR pump and spray pump sumps with the NRC resident inspector identified a number of concerns with the sump screen structure;
0-C-03-4094, Screen fastners discovered without full thread engagement and missing in the RHR and SI recirculation sump during inspections with the NRC resident inspector;
0-C-03-4108, Confusion with NCN dispositions / repair of RB sumps where openings in the wire mesh were larger than allowed, affected QA post-work review process;
0-C-03-4112, NRC inspector pointed out Tig wire remaining in the B spray sump during final closeout inspection, foreign material exclusion event;
0-C-03-4113, Inspection of reactor building spray and RHR sumps were performed prematurely, create a PM task prior to final STP inspection of the sumps;
0-C-03-4143, Several small debris items and boron on the seal table were identified in the Reactor Building during the NRC Mode 4 reactor building inspection during RF-14;
0-C-03-4341, NRC Reg Guide 1.82: Water Sources for Long-Term Recirculation Cooling Following a Loss-of Coolant Accident. This NRC document needs to be reviewed for Virgil C. Summer applicability and possible FSAR revision.

<u>Other Documents Reviewed</u>

NRC Bulletin 2003-01: Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors, dated June 9, 2003;

RC-03-0164, dated August 6, 2003, V.C. Summer Nuclear Station Response to NRC Bulletin 2003-01, Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors;
NRC Perspectives on GSI-191 PWR Sump Performance and NEI PWR Sump Performance Workshop Industry Meeting Presentations, July 2003;
Technical Work Record, TWR KL40152, Proposed Response to NRC Bulletin 2003-01;
CGSS-03-0168, dated October 22, 2003, RF-14 Paint Remediation Plan (plan to inspect/ repair reactor building paint and coatings, commitment under letter RC-03-0164 to the NRC in response to Bulletin 2003-01);
Plant modification performed during RF-14; ECR 50461 Permanent Storage of Scaffold Material / Shielding in the Reactor Building;
Plant Modification performed during RF-14; ECR 50527, RB Sump Vent Addition;
NEI 02-01 Revision 1, Condition Assessment Guidelines: Debris Sources Inside PWR Containments, September 2002;
Virgil C. Summer Nuclear Station Emergency Operating Procedures;
Virgil C. Summer Final Safety Analysis Report;
Virgil C. Summer Technical Specifications;
QSP-522, Reactor Building and Refueling Cavity Inspections;
SAP- 363, Foreign Material and Debris Control;
STP-109.001, Reactor Building Closeout Inspection;
STP-728.030, Reactor Building Fire Barrier Inspections.

Procedures (for Sections 1R08 and 4OA5.1)

- Washington Group Procedure SCEG-VT-89-1, Visual Examination, VT-1, Rev. 1, dated 12/18/01
- Washington Group Procedure SCEG-VT-89-3, Visual Examination for Mechanical and Structural Components, Rev. 1, dated 12/21/01
- Washington Group Procedure SCEG-MT-89-1, Magnetic Particle Examinations of Welds and Bolting, Rev. 2, dated 1/7/02
- Washington Group Procedure SCEG-PT-89-1, Liquid Penetrant Examination - Solvent Removable, Visible Dye Technique, Rev 2, dated 1/7/02
- SCE&G Procedure QSP-504, Magnetic Particle Examination, Rev. 6
- SCE&G Procedure QSP-505, Visual Examination, Rev. 9, and Change A, dated 1/6/03
- SCE&G Procedure QSP-506, IWE and IWL Examination, Rev. 1
- SCE&G Procedure UT-12, Manual Ultrasonic Examination of Austenitic Piping Welds, Rev. 0, dated 1/7/02
- SCE&G Procedure UT-CP-2, Procedure for Inspection System Performance Checks, Rev. 1, dated 10/30/01
- Procedure PDI-UT-2, PDI Generic Procedure for the Ultrasonic Examination of Austenitic Piping Welds, Rev. C, dated 3/15/01
- SCE&G Station Administrative Procedure SAP-1100, Boric Acid Corrosion Control Program, Rev. 0
- SCE&G Quality Systems Procedure QSP-216, Boric Acid Corrosion Inspection, Rev. 0

Other Documents

- Nonconformance Notice number 033654, Cracks in weld in RCP C seal injection nozzle
- Responsible Engineer Evaluation Report for Containment Inservice Inspection, IWE and IWL, dated ½/01
- Augmented Inspection reports for containment tendon access gallery, containment IWE, and RHR and Spray Guard pipes, completed during RFO-13 (April, 2002) and RFO-14 (October, 2003)
- PCI Project Instruction PI-900281-01, Fabrication of Reactor Coolant Pump “C” Injection Line Nozzle, Rev. 0
- PCI Project Instruction PI-900281-02, Installation of Seal Injection Nozzle to Reactor Coolant Pump “C” Thermal Barrier, Rev. 0
- Primary Identification Program 0-C-03-3511, Cracks in weld in RCP C seal injection nozzle
- Visual inspection (VT-3) reports for component cooling system support numbers CCH-0303, -0304, -0306, -0378, -0385, and -0386
- Magnetic particle examination report for reactor vessel upper head to flange weld, weld number 1-1300-1
- Ultrasonic examination reports for weld 2-3000-21 on the containment spray piping, weld number 1-4101A-15 on the safety injection piping, and weld number 1-4107A on the letdown system piping
- Ultrasonic examination reports and visual inspection records for reactor pressure vessel studs 1 - 58