July 1, 1996

South Carolina Electric & Gas Company
ATTN: Mr. Gary J. Taylor
Vice President, Nuclear Operations
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

SUBJECT: NRC INSPECTION REPORT NO. 50-395/96-07 AND NOTICE OF VIOLATION

Dear Mr. Taylor:

This report refers to the inspection conducted on April 21 through June 1, 1996, at the V. C. Summer Nuclear Station. The purpose of the inspection was to determine whether activities authorized by the license were conducted safely and in accordance with NRC requirements. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress.

Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it is described in detail in the subject inspection report. The violation is of concern because the practice of placing contaminated receptacles outside the posted contaminated area invites the spread of contamination.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to prevent recurrence. If the correspondence adequately addresses the required response, your response may reference or include previous docketed correspondence. After reviewing your response to the Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be placed in the NRC Public Document Room (PDR). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information, so that it can be placed in the PDR without redaction.

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The response directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

Sincerely,

Original signed by George A. Belisle

George A. Belisle, Chief
Reactor Projects Branch 5
Division of Reactor Projects

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

Enclosures: 1. Notice of Violation
2. NRC Inspection Report

cc w/encls:
R. J. White
Nuclear Coordinator (Mail Code 802)
S.C. Public Service Authority
c/o Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

J. B. Knotts, Jr., Esq.
Winston and Strawn
1400 L Street, NW
Washington, D. C. 20005-3502

Chairman
Fairfield County Council
P. O. Drawer 60
Winnsboro, SC 29180

Virgil R. Autry, Director
Radioactive Waste Management
Bureau of Solid and Hazardous
Waste Management
S. C. Department of Health
and Environmental Control
2600 Bull Street
Columbia, SC 29201

cc w/encls continued: See page 3

cc w/encls: Continued
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R. Gibbs, RII
P. Fillion, RII
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W. Stansberry, RII
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NRC Resident Inspector
U.S. Nuclear Regulatory Commission
Route 1, Box 64
Jenkinsville, SC 29065

*See previous concurrence

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRS	RII:DRS	RII:DRS
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cc w/encls: Continued
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cc w/encls: Continued

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OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRS	RII:DRS	RII:DRS
SIGNATURE	TRF for					
NAME	RAiello alt	JStaretos	LGarner	WMiller	WKleinsorge	ETesta
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NAME	FWright					
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NOTICE OF VIOLATION

South Carolina Electric & Gas Company
V. C. Summer Nuclear Station

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

During an NRC inspection conducted on April 21 through June 1, 1996, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," NUREG 1600, the violation is listed below:

> Technical Specification 6.8.1 requires, in part, that written procedures be established, implemented, and maintained covering the activities referenced in the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, dated February 1978. Paragraph 7.e of Appendix A to Regulatory Guide 1.33 states that the licensee have written radiation protection procedures.
>
> Station Administrative Procedure SAP-500, Health Physics Manual, revision 8, dated December 9, 1993, Section 6.4.L, Monitoring and Control of Surface Contamination, subsection 1 states, "Contaminated surfaces of permanent structures within the Radiation Control Area are controlled and posted if Beta/gamma emitting loose surface contamination levels exceed 1,000 dpm/100 cm^2."
>
> Health Physics Procedure HPP-158, Contamination Control for Areas, Equipment and Materials, revision 7, dated April 3, 1996, Section 5.1, Contamination Control of Areas/Equipment within the Radiological Controlled Area (RCA), subsection 1 states, "Areas and equipment within the RCA are controlled and posted if the smearable contamination levels exceed 1000 dpm/100 cm^2 Beta-Gamma or 100 dpm/100 cm^2 Alpha."
>
> Contrary to the above, on April 30, 1996, on elevation 436′ in the Hot Machine Shop, the licensee failed to follow the procedural requirements for posting and controlling contaminated areas. The NRC identified contamination levels 15 times greater than the procedural limits for Beta-Gamma outside the posted contamination area.

This is a Severity Level IV violation (Supplement IV).

Pursuant to the provisions of 10 CFR 2.201, South Carolina Electric & Gas Company is hereby required to submit a written statement or explanation to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector, V. C. Summer Nuclear Station, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full

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compliance will be achieved. If the correspondence adequately addresses the required response, your response may reference or include previous docketed correspondence. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information, so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Dated at Atlanta, Georgia
this 1st day of July 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report No.: 50-395/96-07

Licensee: South Carolina Electric & Gas Company
Columbia, SC 29218

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

Facility Name: Virgil C. Summer Nuclear Station

Inspection Conducted: April 21 through June 1, 1996

Inspectors: L. W. Garner for 7-1-96
R. F. Aiello, Senior Resident Inspector Date Signed

J. Starefos, Resident Inspector
L. Garner, Project Engineer (Paragraphs 3.1.1, 3.1.2, 3.2.1, 3.8, 4.2)
W. Miller, Reactor Inspector (Paragraphs 2.7, 4.1, 5.7.1)
W. Kleinsorge, P.E., Reactor Inspector (Paragraph 3.3)
E. Testa, Reactor Inspector (Paragraphs 5.4, 5.7.2, 5.7.3)
F. Wright, Reactor Inspector (Paragraphs 5.4, 5.7.2, 5.7.3)
A. Johnson, Project Manager, NRR (Paragraph 7.0)

Approved by: George A. Belisle 7/1/96
George A. Belisle, Chief Date Signed
Reactor Projects Branch 5
Division of Reactor Projects

SUMMARY

Scope:

This routine integrated inspection was conducted on site in plant operations which included plant status, component cooling water pump speed switch opened while energized, refueling water storage tank flow path to reactor building emergency sump, flow control valve leakage past seat, containment walkdown, main control board instrumentation off normal indicators, core alteration, and close out issues; maintenance which included power shield testing, emergency feedwater pump turbine steam supply flow control valve maintenance, Type C testing, rod testing, inservice inspection, steam generator eddy current testing, control rod drag testing, main steam isolation valve closure due to personnel error, high flux at shutdown alarm setpoint too high, and close out issue; engineering which included 10 CFR 50.59 screening and safety evaluations, emergency feedwater steam line supports, and close out issue; and

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plant support which included fire protection, NRC Bulletin 96-02, health physics, chemistry and radiological controls, industrial safety initiatives, refueling 9 exposure control, and close out issues. In addition, reviews of Final Safety Analysis commitments and an Institute of Nuclear Power Operations Evaluation Report were also performed.

Results:

Plant Operations

A Non-Cited Violation was identified due to operators inadvertently opening an energized Component Cooling Water (CCW) pump speed switch that resulted in equipment damage and minor personnel injury (Paragraph 2.3).

A Non-Cited Violation was identified due to operators inadvertently creating a direct flow path from the refueling water storage tank to the B containment recirculation sump that resulted in an unwanted transfer of approximately 11,000 gallons of water (Paragraph 2.4).

The licensee made a four hour report regarding un-analyzed Emergency Feedwater (EFW) flow control valve leakage (Paragraph 2.5).

An Unresolved Item (URI) regarding the misapplication of paint used inside containment was identified (Paragraph 2.6).

A number of instrument meters on the main control board were reading outside the green band. These green bands were installed on the meters to indicate the normal operating zone. The plant has operated in this manner for several years. This item was identified as an URI (Paragraph 2.7).

The inspectors identified an URI with respect to the licensee conducting core alterations without containment integrity (Paragraph 2.8).

A violation for failure to comply with a system operating procedure was closed after reviewing corrective actions (Paragraph 2.9.1).

Air inleakage caused no damage to the Turbine Driven Emergency Feedwater (TDEFW) pump (Paragraph 2.9.2).

Maintenance

The inspectors did not identify any significant problems while reviewing power shield testing, emergency feedwater pump turbine steam supply flow control valve maintenance, 10 CFR 50 Appendix J, Type C tests, and the rod position indication test (Paragraphs 3.1.1, 3.1.2, 3.2.1, 3.2.2).

The inspectors reviewed documents and records, and observed work activities to determine whether Inservice Inspection (ISI) was being conducted in accordance with applicable procedures, regulatory requirements, and licensee commitments.

During this outage, the inspectors reviewed certification records of Non-Destructive Examination (NDE) equipment and materials, and reviewed NDE personnel qualifications for personnel who had been utilized in the ISI examinations. The inspectors identified no concerns (Paragraph 3.3).

The licensee made a 10 CFR Part 21 event notification on May 20 and a clarifying update on May 22 regarding the reactor coolant pump seal injection line and the CCW thermal barrier welds (Paragraph 3.3).

The inspectors did not identify problems while inspecting steam generator eddy current testing or control rod drag testing (Paragraphs 3.4 and 3.5).

The licensee made a four hour report for an inadvertent engineered safety feature actuation resulting in main steam isolation valve closure. The inspectors identified this as a Non-Cited Violation (Paragraph 3.6).

The licensee identified that the source range high flux at shutdown was set too high due to a failure to follow procedures. The inspectors identified this as a Non-Cited Violation (Paragraph 3.7).

An Unresolved Item associated with minimum monthly operating times for engineered safety feature filter trains was closed (Paragraph 3.8).

Engineering

A review of 24 modifications and procedure changes indicated that the licensee's Final Safety Analysis Report (FSAR) screenings were thorough. Furthermore, the licensee adequately identified the applicable FSAR and Technical Specification sections related to the proposed changes (Paragraph 4.1).

As found settings on five emergency feedwater steam line supports were correct (Paragraph 4.2.)

The inspectors reviewed actions involving a partially submerged reactor building tendon (Paragraph 4.3).

The inspectors identified positive findings in the facility's review of their FSAR commitments (Paragraph 7.0) and communications with the NRC resident inspectors (Paragraph 4.0).

Plant Support

The inspectors identified a Non-Cited Violation concerning the failure to report a minor fire in the Turbine Building (Paragraph 5.2).

The inspectors did not identify any concerns during the review of the licensee's actions for NRC Bulletin 96-02 (Paragraph 5.3).

The licensee submitted their annual Effluent and Waste Disposal Report for 1995. The results were similar to previous reports (Paragraph 5.4.1).

The programs for classification, packaging and shipment of radioactive materials were reviewed. No problems were identified (Paragraph 5.4.2).

Radiological effluent monitoring instrumentation was reviewed. No problems were identified (Paragraph 5.4.3).

In the Health Physics, Chemistry and Radiological controls areas, the licensee was collecting and reporting the required meteorological data, and maintaining the meteorological monitoring instrumentation in an operable condition (Paragraph 5.4.4).

The licensee had also implemented water chemistry control programs in accordance with the Technical Specification requirements and the Electric Power Research Institute guidelines for Pressurized Water Reactors primary and secondary water chemistry (Paragraph 5.4.5).

Housekeeping and the control of contaminated and radioactive material within the licensee's auxiliary, radioactive waste, containment, warehouses, scrap storage areas, fuel handling, and intermediate buildings, were acceptable. However, one violation for failure to follow procedural requirements for posting and controlling contaminated areas was identified (Paragraph 5.4.6.3).

The inspectors identified two positive findings, one in the area of industrial safety (Paragraph 5.5) and one in Refueling Outage 9 exposure control (Paragraph 5.6).

The licensee successfully completed actions to resolve issues involving the fire system, and free standing water and high water content in resins shipped to the disposal site (Paragraphs 5.7.1, 5.7.2 and 5.7.3).

REPORT DETAILS

Acronyms used throughout this report are defined in paragraph 10.

1.0 PERSONS CONTACTED

Licensee Employees

*Bacon, F., Manager, Chemistry Services
Blue, L., Manager, Health Physics
*Browne, M., Manager, Design Engineering
*Byrne, S., General Manager, Nuclear Plant Operations
Derrick, J., Supervisor, Procurement Engineering
*Fields, C., Manager, Materials and Procurement
Fipps, S., Independent Safety Evaluation Group
*Fowlkes, M., Acting Manager, Operations
Franchuk, T., Supervisor, Administration, Facilities, Document Control
Furstenberg, S., Manager, Maintenance Services
*Hunt, S., Manager, Quality Systems
Kammer, M., Supervisor, Design Engineering
*Kelley, V., Coordinator, Emergency Services
Lavigne, D., General Manager, Nuclear Safety
*Lippard, G., Acting Manager, Nuclear Licensing and Operating Experience
*Loignon Jr., G., Project Coordinator
McAlister, T., Supervisor, Quality Control
*Moffatt, G., Manager, Planning and Scheduling
*Mothena, P., Supervisor, Health Physics
Nesbitt, J., Manager, Technical Services
*Nettles, K., General Manager, Strategic Planning and Development
*O'Quinn, H., Manager, Nuclear Protection Services
*Proper, J., Supervisor, Nuclear Licensing and Operating Experience
*Taylor, G., Vice President, Nuclear Operations
*Taylor, T., Manager, Engineering Services
*Waselus, R., Manager, Systems and Component Engineering
Wasieczko, J., Supervisor, Security Operations
White, R., Nuclear Coordinator, South Carolina Public Service Authority
Williams, G., Associate Manager, Operations

Other licensee employees contacted included office, operations, engineering, maintenance, chemistry/radiation, and corporate personnel.

2.0 PLANT OPERATIONS (71707, 92901, 40500, 37550)

2.1 Plant Status

At the start of this inspection period, the Unit was in RF9 which commenced on April 15, 1996. The plant entered Mode 4 on May 15 at 9:12 pm; Mode 3 on May 18 at 12:03 pm; Mode 2 on May 20 at 6:30 pm when the reactor went critical; and Mode 1 on May 22 at 4:10 pm. The plant was operating at 100 percent power at the end of the inspection period.

2.2 General

The inspectors conducted frequent Control Room (CR) tours to verify proper staffing, operator attentiveness, and adherence to procedures in order to maintain awareness of overall facility operations. The inspectors reviewed operator logs to verify operational safety and compliance with Technical Specifications (TS), attended daily plant status meetings, and shift turnovers. The inspectors periodically reviewed instrumentation and safety system lineups from CR indications to assess operability. The inspectors conducted plant tours to observe equipment status and housekeeping.

In order to assure that potential safety concerns were properly reported and resolved, the inspectors reviewed Off Normal Occurrences (ONOs). The inspectors routinely attended plan of the day meetings where management discussed the details of selected ONOs and the proposed actions to resolve the issues.

During the outage, the licensee provided a white board with information for the operators in the control room. The board included the outage safety system bar chart, a list of Abnormal Operating Procedures (AOP) which could be necessary during refueling operation (e.g., AOP-115.4, Loss of Residual Heat Removal (RHR) While Refueling), and other operator aids such as the boration flow path and time to boil in the Spent Fuel Pool (SFP) and in the reactor cavity. The inspectors determined that this was a good initiative to enhance the operator's response capability.

The official outage length was determined by the licensee to be approximately 39 days. The inspectors verified by periodic observation that the licensee successfully transitioned through each operational mode change. The licensee carefully assured that all applicable TS Surveillance Test Procedures (STPs) were adequately and completely performed.

2.3 Component Cooling Water Pump Speed Switch Opened While Energized

On April 25, 1996, an electrical fault was caused on the speed switch for the B CCW pump by operators performing switching operations. The fault occurred when the operators attempted to open the B CCW pump 7200 volt fast speed switch under load. The operators were directed to manipulate the de-energized C CCW pump speed switch but inadvertently entered the B room and opened the energized switch for the running B pump. The fault was interrupted when the B CCW pump's breaker opened as expected with this condition. The A CCW pump was out of service at the time. The control room placed the C CCW pump in service to compensate for the loss of the B CCW pump. SFP cooling was interrupted until the C CCW pump was started. The licensee determined that the temperature in the SFP increased approximately 0.3° F. Although the event did not result in a fire, the B CCW speed switch was damaged. Both operators suffered minor injuries.

An incident investigation by Quality Assurance/Independent Safety Engineering Group (QA/ISEG) was performed to determine the root cause of the event. The team concluded that the first root cause was that procedural guidance was not used; the second was that the operators were distracted; and, the third was attributed to inadequate operator knowledge.

The licensee held an Management Review Board meeting on April 30, 1996, to address this event. The licensee plans to take the following actions:

Evaluate effective physical barriers to preclude future inadvertent operation.

Review Station Administrative Procedure (SAP)-123, SAP-200, station policies, and procedures for clarity of expectations.

Evaluate data presented and determine if trends exist.

Evaluate AI-600 based on conditions identified to assure appropriate measures are established to insure proper plant conditions for this and similar events.

Establish a plan for the performance of a human factors evaluation of the demarcation of rooms and doors for the plant.

Establish a policy for pre-job briefings for evolutions in operations.

The nuclear safety significance was minor since there was no fuel in the RV and loss of cooling to the SFP was only momentary until the C CCW pump could be racked in and started. The licensee determined the root cause and took measures to minimize recurrence. This licensee identified violation for failure to follow procedure is being treated as a Non-Cited Violation (NCV) consistent with Section VII.B.1 of the NRC Enforcement Policy. This is identified as NCV 50-395/96-07-01, CCW Speed Switch Event.

2.4 Refueling Water Storage Tank Flow Path to Reactor Building Emergency Sump

On May 9, 1996, at approximately 5:10 pm, the licensee inadvertently created a direct flow path from the RWST to the B containment recirculation sump. The licensee estimated that approximately 11,000 gallons of RWST water flowed to the sump prior to the flow path being secured by a control room supervisor.

Operations was performing integrated safeguards testing when the event occurred. During the test, the licensee de-energized valves between the sump and the suction of the RHR and CS pumps. The transfer of water resulted when de-energized valves in the RHR and CS lines from the B RHR and B CS sump were energized while a locked-in safety injection signal was present. This locked-in safety injection signal caused the valves to open creating a direct flow path from the RWST to the sump. No inventory was lost from the reactor coolant system. The opposite RHR cooling train was maintained throughout the event. The majority of the water in the RHR and CS sump was transferred to the primary relief tank, processed and then transferred back to the RWST. The remainder of the water was transferred out through the Reactor Building sump system or cleaned up by decontamination personnel.

Procedural steps would have prevented this transfer of water. However, the master control copy of the procedure was issued from Document Control with a page missing. Document Control Procedure 101, Document Control, revision 2, requires that controlled documents will be complete with all pages and attachments. The failure to page check this document is a procedural violation and is being treated as an NCV consistent with Section VII.B.1 of the NRC Enforcement Policy. This is identified as NCV 50-395/96-07-02, RWST Water in RHR and CS Emergency Sump.

2.5 Flow Control Valve Leakage Past Seat

During integrated safeguards testing conducted on May 12, 1996, operations personnel observed that steam generator levels were increasing. This was the result of leakage past the closed motor driven EFW pump FCVs. Upon further investigation, it was determined that flow past these closed valves had the potential to result in an un-analyzed condition as related to the reactor building steam line break analysis. The concern was also extended to the turbine driven EFW pump flow control valves. The licensee's evaluations indicated that with the faulted steam generator isolated, some leakage was acceptable at the onset of the accident. After approximately 1800 seconds, emergency feedwater to the faulted steam generator was assumed to be terminated. The current leakage past the valves exceeded the leakage limits as stated in the FSAR. A four hour report was made on May 14, 1996. The licensee has modified four of their Emergency Operating Procedures to direct operators to shut both the upstream and downstream isolation valves on the faulted steam generator's EFW FCVs within 30 minutes following an event. The inspectors considered this change to be acceptable.

2.6 Containment Walkdown

The inspectors conducted a walkdown of containment with QC personnel on May 13 and an independent walkdown on May 15, 1996. The inspectors visually inspected containment housekeeping, component and instrument conditions, storage of equipment and material, pipe hanger and seismic restraints, breaker and instrument covers, and the reactor cavity. The inspectors also performed a detailed visual inspection of the RHR and CS sump area. The inspectors identified one safety concern. Several fasteners were missing from the RHR and CS sump debris screens. Most of the fasteners were inaccessible. Therefore, design engineering performed a design calculation to analyze this condition. This calculation evaluated the consequences of the missing screen fasteners, angle support nuts and fasteners with limited thread engagements. The calculation demonstrated, to the inspectors' satisfaction, that the condition of the screen fasteners would not prevent the screens from performing their design basis functions during recirculation.

Subsequent to the containment walkdown, the inspectors became aware of an issue regarding the painting performed inside containment during the outage. Floors in the reactor building were painted with a single coat instead of the two coats as described in CMP-500.001, Application of Protective Coatings to Concrete Surfaces Inside the Reactor Building, revision 4. Until the inspectors perform additional reviews of the adequacy of one coat of paint, this item will be tracked as Unresolved Item (URI) 50-395/95-07-03, Paint Coat in Containment Does Not Meet Application Specification.

2.7 Main Control Board Instrumentation Off Normal Indicators

During an August 1995 NRC inspection, an inspector noted that many of the principal vertical meter indicators on the main control board were reading significantly outside of their normal operating range as indicated by green identification bands. However, none of these measured parameters were in alarm nor were any of the indicators reading in excess of the identified red band. These instruments included pressurizer surge line temperature, RCS flow, main steam and main feedwater flow, main feedwater temperature, RCS T-hot, and refuel cavity temperature. The licensee informed the inspector that the green bands had been installed in the early 1980s to indicate the normal operating band for the measured parameters, but the normal operating indication for these meters had been reading outside of the green bands for several years. This information is documented in NRC Inspection Report No. 50-395/95-15, Section 3.d.

Following the August 1995 inspection, the licensee initiated design package MRF-90102C, Main Control Board (MCB) Indicators. This design modification revised the normal instrumentation indication for a number of the instrument meters on the main control board to support the power uprate (effective at the beginning of the fuel cycle which started in May 1996) and to correct the existing instrumentation meters which were reading outside of the green or normal operating band.

In May 1996, the inspectors reviewed design package MRF-90102C and noted that the green operating bands for 84 instruments were to be revised. Of this total, 24 were revised during the Spring 1996 refueling outage. Completion of the revision to the green normal operating bands for all meters was scheduled to be completed prior to the end of the Fall 1997 refueling outage. After this date, all applicable meters will be provided with a green band identifying the normal operating range.

From 1980 through 1982, NRC licensing human factor engineering reviews were performed on the Summer facility. During these reviews, the licensee committed to perform a number of modifications to the control room to meet the guidelines of NRC NUREGs 0737 and 0700 concerning TMI

Action Items. The results of these reviews were documented in the Summer Safety Evaluation Report, Supplement 4, dated August 1982. One of the licensee's commitment modifications included the installation of indicator markings for a number of vertical meters on the main control board. The licensee was to provide a green band to indicate the normal operating range, a yellow mark to indicate caution alarm limits, and a red mark to indicate danger or high/low interlock alarm points. This commitment was documented in enclosures sent to the NRC by the licensee in letters dated January 15, 1981 and March 26, 1982. The commitment was also included in the enclosure sent to the NRC by the licensee's letter dated April 15, 1985.

In August 1995, an NRC inspector noted that the normal meter indication for a number of meters was reading outside of the green indication zone. Apparently, the plant had operated in this manner for several years. Pending additional regulatory review, this is identified as an URI 50-395/96-07-04, Control Room Meters Reading Outside Green Indication Zone. This condition is scheduled to be corrected upon the completion of modification MRF-90102C.

2.8 Core Alteration

On April 21, 1996, the licensee was performing FHP-604, Functional Testing of Fuel Handling Systems, revision 10, Change A. The trolley was indexed over the reactor vessel and the gripper was lowered within the pressure vessel. Currently, in TS, core alteration is defined to be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. In the past, the licensee conducted functional testing of fuel handling system components without TS conditions such as containment integrity being established for core alterations. Pending further NRC review, the item is identified as URI 50-395/96-07-05, Conducting Core Alteration Without Containment Integrity.

2.9 Close Out Issues

2.9.1 (Closed) VIO 50-395/94-19-01, Failure to Comply with a System Operating Procedure. On August 12, 1994, the licensee was performing a planned release of Waste Gas Decay Tank (WGDT) G. During the release, the licensee noted that the inservice WGDT A was also being released as indicated by decreasing pressure in the tank. The contents of both tanks were being monitored for high radiation levels during the release and no setpoints were exceeded. The licensee stopped the release of both A and G tanks and performed a valve lineup for the waste gas system. Two valves, XVD07881-WG and XVD07883A-WG, were found to be open rather than closed as required by System Operating Procedure (SOP)-119, Waste Gas Processing System Operating Procedure. These two mispositioned valves accounted for the simultaneous release of the A and G WGDTs. The licensee's evaluation of the inadvertent release found that even with the release of the contents of tank A, the resulting offsite doses were within the limits established in the Offsite Dose Calculation Manual. The setpoint established on the discharge effluent monitor was conservative and would have stopped the release if the combined activity of the tanks had exceeded the instantaneous release rate limits. The inspectors verified that SOP-119 was revised to verify proper position of system valves prior to initiating a tank release. This change provided additional assurance that an unintentional release will not occur in the future. In addition, the licensee placed a copy of this violation response in the Operations Department required reading file. Their review of this response emphasized the importance of procedural compliance and completion of work activities.

2.9.2 (Open) IFI 50-395/95-20-01, Air in Service Water (SW) Piping to Suction of Turbine Driven Emergency Feedwater (TDEFW) Pump. On December 11, 1995, the TDEFW pump was removed from service for planned maintenance. The maintenance included work on the pump casing vent valve and the discharge piping vent valve. The work required draining the suction header. Following maintenance, the suction header was filled and vented. When the TDEFW pump was started for post maintenance testing, operators in the control room received low suction pressure alarms. In addition, the pressure indicators on the suction and discharge sides of the TDEFW pump fluctuated abnormally. The TDEFW pump was secured and an investigation into the abnormal indications was initiated. In May 1996, the inspectors reviewed the engineering assessment and the disassembled TDEFW pump. There was no evidence of damage as a result of air inleakage as identified in NRC Inspection Report No. 50-395/95-20. Pending review of the licensee's root cause determination for the air inleakage, this item remains open.

In the area of operations, three unresolved items and two NCVs were identified.

3.0 MAINTENANCE (62703, 55050, 61726, 73753)

3.1 Maintenance Observations

3.1.1 Power Shield Testing

During this refueling outage, a program had been established to inspect and, if necessary, resolder an electrical connection in breaker power shields with certain model numbers. More than 50 safety-related and non safety-related breakers were included in the program.

The inspectors observed post maintenance testing of a refurbished power shield for the breaker installed in cubicle number 05C of switchgear XSW-1DB2. During the test, the technicians discovered that the long time trip test was not being performed with the selector plug in its design location. The plug was repositioned and the test was resumed. EMSI-NCN 5358, Power Shield Replacement Per Non-Conformance Notice 5358, revision 0, Change B, contained sufficient instructions to preclude a breaker from being returned to service with a power shield incorrect plug setting. Specifically, after test equipment removal, step 7.8 required all plug settings to be returned to their original values and to be verified correct by a second person. The inspectors verified that the plug settings were correct and the procedure's acceptance criteria was met.

When the plug setting was changed, a new meter scale was selected on the test equipment. The inspectors noted that the technician obtaining the data experienced difficulty in determining what the small divisions represented. With assistance from his foreman, the correct values were obtained. The observation was discussed with maintenance supervision. The licensee reviewed other power shield work records performed by the technician and found no discrepancies.

3.1.2 EFW Pump Turbine Steam Supply FCV Maintenance

On May 6, 1996, the inspectors observed the reassembly of IFV02030-MS, EFW Pump Turbine Steam Supply Flow Control Valve. The work was properly performed in accordance with procedures and drawings. The reassembled parts were undamaged and suitable for use. Foreign material exclusion controls associated with the open valve were satisfactory.

Two items noted by the inspectors were subsequently identified by maintenance management as not meeting management's expectations. Disconnected instrument air tubing to the pressure regulator and solenoid valves for the valve's actuator were left uncovered. This condition was subsequently corrected. Also, prior to performing the first torque pass on the bonnet-to-body nuts, a numbering scheme different than the recommended one in the procedure was established. Although the numbering scheme would have provided the correct torquing sequence, a standardized numbering scheme had been established to ensure consistency between crews and to reduce the potential for confusion and human errors. When the inspectors pointed out that the numbering scheme was different from the one in the procedure, the nuts were relabeled in accordance with the numbering scheme in the procedure.

3.2 Surveillance Observations

3.2.1 Type C Testing

The inspectors observed 10 CFR 50 Appendix J, Type C, tests being performed on containment penetration numbers 408 and 302 on May 2 and 7, respectively. Containment penetration number 408 test was turned over to the oncoming test group with the piping around the outboard valve, MVG-8102A, as having been drained. The oncoming test group decided to verify that the piping was completely drained. Upon opening a drain valve adjacent to this valve, approximately one and one-half pints of water drained from the line. This observation was later discussed with maintenance supervision.

The inboard valve, XVC08368A-CS, for containment penetration number 408 was located in an approximately three foot pipe section that could not be gravity drained. Compressed air was used to blow the water in this section of pipe out a vent line. The method used to remove the water and the piping configuration around the inboard valve was discussed with the NRC regional Appendix J specialist. For this containment penetration, the inspectors determined that, without a plant modification, the licensee did what was practical to remove water from around the inboard valve and that the test was valid as performed.

At the time of this inspection, 99 valves had been successfully tested, three had failed, and 33 remained to be tested. The inspectors verified that the failed valves had either been repaired and successfully retested or were scheduled for repair and retest.

The inspectors reviewed the completed test documentation for 15 containment penetrations tested this outage, the test summary results for all Type C containment penetrations since 1991, the Third Periodic Reactor Containment Building Integrated Leakage Rate Test Report dated March 12, 1993, and FSAR section 6.2.6. Some minor documentation problems, such as data blocks not being checked, were noted. In one instance, on May 1, 1996, the test results for valve MVG-8888B were recorded on the incorrect data form under valve MVG-8889. These observations were discussed with the cognizant supervisor. The licensee subsequently entered the MVG-8888B data on the correct form and attached it to the completed procedure. The review did not disclose any adverse trends in containment isolation valve performance which had not already been noted and actions initiated to correct. Testing was being performed as described in the FSAR. The Type C test results indicated that the containment isolation valves were being well maintained.

3.2.2 Rod Testing

The inspectors observed that STP-106.002, Rod Position Indication Operational Test, revision 1, Change G was performed satisfactorily.

3.3 Inservice Inspection (ISI)

The inspectors reviewed documents and records, and observed activities to determine whether ISI was being conducted in accordance with applicable procedures, regulatory requirements, and licensee commitments. The applicable code for ISI is the ASME B&PV Code, Section XI, 1989 Edition with no Addenda (89NA). The plant completed the ninth fuel cycle, and is in the second outage (RF9), of the first 40 month period, of the second ten year ISI interval (I2,P1,O2). The plant received its operating license August 6, 1982, and commenced commercial operations on January 1, 1984. The licensee, SCE&G Company, has contracted with RC&E to perform PT, MT, VT, and UT examinations under the umbrella of the RC&E QA program. Some MT, PT, VT, and UT examinations were performed by SCE&G, under their own QA program. ET examination of steam generator tubing was conducted by Framatome Technologies under the Framatome QA program.

The inspectors reviewed two documents relating to the ISI program to determine whether the plan had been approved by the licensee and to assure that procedures and plans had been established for the applicable activities. The documents were also reviewed for technical content. The documents were appropriate for the intended application.

The inspectors reviewed 12 Non-Destructive Examination (NDE) procedures to determine whether the procedures were consistent with regulatory requirements and licensee commitments. The procedures were also reviewed for technical content. Relative to procedure QSP-501, Solvent Removable Liquid Penetrant (SCE&G), revision 3, the inspectors noted that the procedure did not require the light meter used (to measure Code specified ultraviolet intensity) to be calibrated. The inspectors did note that florescent PT examinations had not been performed for quite some time. The meter was verified to be calibrated. Additionally, the procedure did not implement the ASME Section XI paragraph T-646.2 (a) required five minute stay time in the darkened area prior to the performance of florescent PT examinations. Except as noted above, the documents were well written and appropriate for their intended application. The licensee indicated that they would make appropriate procedural corrections.

The inspectors observed work and work activities, reviewed certification records of NDE equipment and materials, and reviewed NDE personnel qualifications for personnel who had been utilized in the ISI examinations during this outage.

The inspectors observed/reviewed records for 23 PT, 2 MT, 22 UT, and 12 VT examinations. The observations were compared with the applicable procedures and the Code. The examinations were performed satisfactorily.

The inspectors reviewed certification documentation for the following NDE equipment and consumables: a PT cleaner, two PT developers, a PT penetrant, an MT 10 Lbs. test weight, an MT yoke, an MT powder, three UT instruments, 10 UT transducers, one UT couplant, five UT reference blocks, three remote data acquisition units and eight ET calibration standards. The inspectors also reviewed the certification, qualification, and visual acuity documentation for 13 PT, 10 MT, 12 UT, 4 VT and 6 ET examiners.

TS require ET examination of a minimum of three percent of the total SG tube population. The licensee allowed themselves an inspection window of 36 hours. They set a goal of tube end-to-end bobbin coil ET examination of 27 percent of the tube population, concentrated at the periphery with the balance throughout the bundle. The licensee actually examined 22 percent of the tubes in SG A and 16 percent of the tubes in SG B or approximately 13 percent of the total tube population.

The inspectors concluded that ISI examinations were satisfactorily conducted by properly qualified personnel in accordance with appropriate procedures. Records reflected the accomplishment of examinations in accordance with procedural requirements and regulatory commitments.

The inspectors reviewed procedures, observed work activities and reviewed selected records to evaluate the licensee's welding program. The applicable Code for welding on safety-related piping structures and components is ASME B&PV Code Section XI 89NA. The applicable Code for welding procedure and performance qualification is ASME B&PV Code Section V (the latest at the time of qualification). Because South Carolina is a Non ASME Code state, there are no regulatory requirements associated with non safety-related welding. The licensee uses their safety-related welding program for non safety-related work whenever possible.

The inspectors observed welding projects in progress. These included seal injection and CCW nozzles to RCP thermal barrier welds (ASME Class 1) and installation of the closed cooling water system (non safety-related).

In May 1987, the C RCP seal injection nozzle was discovered leaking in the socket welded connection of the nozzle to the thermal barrier. At that time, the weld was partially excavated and rewelded. In December 1994, during RF8 the C RCP seal injection nozzle, again, had a through wall leak on the same weld. This time the crack was in the lower portion of the weld. This weld was also partially excavated and replaced. The licensee contacted the Electric Power Research Institute (EPRI) to develop an NDE technique to identify cracks of the type experienced on the C RCP seal injection nozzle socket weld. EPRI developed an ultrasonic go/no-go gage to inspect the RCP thermal barrier nozzles. The remaining eight nozzles were examined in the last week of April 1996, during RF9. The licensee detected indications associated with five welds. The five nozzle assemblies (flange and pipe) were replaced. The inspectors observed defect removal, welding, and NDE activities associated with the nozzle replacements and examined completed welds. The licensee made a 10 CFR Part 21 event notification on May 20 and a clarifying update on May 22, 1996, regarding RCP seal injection line and CCW thermal barrier welds. The subject condition appears to be a generic hardware defect in manufacturing of a basic component which has the potential to cause a loss of safety function and a major reduction in the degree of protection provided to public health and safety. There was a potential for a simultaneous failure of the nozzle weld defects in the seal injection line and the CCW connections at the RCP thermal barrier flange. Should this occur there would be no cooling to the RCP seal or lower pump bearing. Both would have been subjected to RCS cold leg temperature in excess of 550°F. This would have had the potential to initiate a RCP seal failure induced loss of coolant accident.

The licensee was in the process of adding a cooling tower to the turbine building cooling system. This is discussed further in NRC Inspection Report No. 50-395/96-05. As a result of welding problems discovered in March 1996, associated with the closed cooling water system, the licensee initiated an inspection program implemented by the site QC organization. This program included visual examinations of welds at the fit-up, root and final stages of completion for the remaining pipe welds. The inspectors observed weld related work activities associated with the closed cooling water system. In addition, the inspectors conducted a walkdown inspection of the piping system and examined selected completed welds.

Approximately two dozen weld rod stubs were found adjacent to the closed cooling water piping at the cooling tower (located outside of security area). One stub was noted in the turbine building in a small pile of debris. No stubs were identified in the reactor building. The stubs outside the security area are indicative of poor housekeeping practices but are not a nuclear safety concern as the rods could not be used in a safety related application. The stub in the turbine building should have been placed in the locked rod stub disposal receptacle. Except as noted, all welding filler material was controlled consistent with the Code. All electrode ovens containing covered electrodes were energized. Welds selected by the inspectors for visual examination were acceptable.

Welders were properly identified by welder qualification cards, or picture identification (e.g., Drivers License) that could be cross indexed with the welder qualification documentation. The inspectors reviewed records attesting to 11 welder qualifications and welder qualification maintenance, 5 welding filler material certifications and receiving inspection documentation, 3 welding procedure specifications, and 10 of their supporting procedure qualification records.

The licensee needed extra welding personnel on short notice to work on the closed cooling water system. They transferred some personnel from their fossil plants to meet this need. As South Carolina is not a ASME Code state, there is no requirement for compliance with the ASME Code for fossil application. There is no licensee committed welding Code for non safety-related applications. The fossil plant welders had proper documentation supporting appropriate ASME Section IX initial qualification. The records in support of the maintenance of those qualifications were incomplete. The fossil welders were limited to welding on the non safety-related closed cooling water system.

Direct observation and review of records support the conclusion that welding was accomplished by qualified welders using certified filler materials in accordance with approved procedures and appropriate Codes.

3.4 Steam Generator Eddy Current Testing

During the refueling outage, the licensee performed eddy current testing on the A and B SGs. The licensee evaluated 1393 tubes on the A SG and 1038 on the B SG. TS surveillance requirement 4.4.5.2, SG Tube Sample Selection and Inspection, requires that at least three percent of the total number of tubes in all SGs are selected for each ISI. The licensee inspected approximately 12.8 percent of the total tubes.

The inspectors reviewed and discussed the results of the RF9 SG tube inspection with engineering personnel. The results of the ET identified one non-quantifiable imperfection in SG A and two less than 20 percent of nominal tube wall thickness imperfections in SG B. The licensee indicated that no tubes were plugged as a result of this inspection.

The licensee submitted the inspection results to the NRC in a letter dated May 13, 1996. This letter did not include one of the two SG B imperfections. The licensee planned to supplement the letter.

3.5 Control Rod Drag Testing

In response to NRC Bulletin 96-01, Control Rod Insertion Problems (reference NRC IR 50-395/96-05), the licensee submitted rod drop and drag test data in a letter to the NRC dated May 24, 1996. The licensee drag tested all 48 of the rodded fuel assemblies with acceptable results. Most of the drag test force data was below the drag acceptance criteria of 100 pounds. However, one high burnup assembly (45123 MWD/MTU) rod drag force approached the limit of the drag acceptance criteria. The licensee tested that assembly twice. The data for the two drag tests on that assembly was 96 pounds and 100 pounds respectively. The inspectors observed portions of the rod drag testing and did not identify any concerns.

3.6 Main Steam Isolation Valve Closure Due to Personnel Error

On May 17, 1996, the licensee was restarting the plant following the completion of RF9. The MSIVs and their bypass valves were open to heat the steam lines. An I&C technician was performing surveillance tests on the steam flow transmitters. Procedure STP-395.004, SG A Steam/Feedwater Flow Instrument IFT00474/IFT00477 Calibration, states "If Tavg is </= 552°F, P12 status light is ON and the LO-LO Tavg status light is ON, DO NOT place FS/484A, C3-746 or FS/494A, C3-748 comparator trip switches listed on Attachment 1, to the test position." The I&C technician inadvertently placed both FS/484A and FS/494A transmitters into the test mode simultaneously. This completed the two-out-of-three actuation logic for MSIV isolation. The MSIVs and their bypass valves closed. No other actuations occurred. The bistables were returned to normal and the valves reopened. A four hour report was made as required by 10 CFR 50.72. This licensee identified violation for failure to follow procedure is being treated as an NCV consistent with Section VII.B.1 of the NRC Enforcement Policy. NCV 50-395/96-07-06, Inadvertent Engineered Safety Features (ESF) Actuation Resulting in MSIV Closure.

3.7 High Flux At Shutdown Alarm Setpoint Too High

On May 4, 1996, contrary to Refueling Procedure (RP)-109.003, Source Range Baseline and Statistical Reliability Calculations, revision 0, an I&C technician incorrectly set the Hi Flux at shutdown alarm setpoint too high. This was discovered during the performance of STP 102.001, Operations Weekly Channel Test. At that time, 45 of 157 fuel assemblies had been reloaded into the RV. The setpoint was off during the remainder of fuel reloading. The I&C technician used average counts (taken over a 10 second interval) instead of counts per second as required by RP-109.003. Subsequently, the high flux at shutdown alarm was set too high by an order of magnitude in the nonconservative direction.

With respect to safety, the basis for the alarm function was to detect an unwanted approach to criticality due to an inadvertent dilution of soluble boron. This basis is described in the FSAR and in the plant DBD. However, the alarm was not specified in TS. The prevention of boron dilution events is administratively controlled by locked valves in the closed position. The IPCS redundantly monitors the source range high flux at shutdown function. If a dilution event occurred, an IPCS alarm would input into a group of alarms called "OPCRIT" which not only displays a visual line item on the IPCS monitor in the CR, but also sounds an audible bell. Although the MCB annunciator function was set non conservatively, the IPCS alarm was never disabled or set inappropriately during this time. This licensee identified violation for failure to follow procedure is being treated as an NCV consistent with Section VII.B.1 of the NRC Enforcement Policy. This is identified as NCV 50-395/96-07-07, Source Range High Flux at Shutdown Set too High Due to Failure to Follow Procedures.

3.8 Close Out Issue

(Closed) URI 50-395/96-02-04, Review Difference Between FSAR Statements and TS Requirements For Minimum Monthly Operating Requirements of ESF Filter Trains. This item involved FSAR Table 6.5-1 statements that ESF filter trains would be operated at least 10 hours monthly, whereas, TS required the certain ESF filter trains to be operated at least 15 minutes per month. The licensee currently complies with the TS requirement. However, the inspectors verified by review of operating and surveillance procedures and interviews with licensed operators that the ESF filter trains for the control room are not routinely operated nor run a minimum of 10 hours per month. In addition, the ESF filter trains for the auxiliary building are normally operated with two of the three filter banks in service. However, the third bank is normally only placed in service when one of the other two banks is required to be removed from service for maintenance. The significance of not flowing air through a filter bank for at least 10 hours a month was discussed with cognizant personnel in NRR. Since the filter trains do not contain heaters, there is little or no difference between operating the ESF filter trains for 15 minutes verses 10 hours every month. Thus, the inspectors concluded that the practice of complying with just the TS requirement was technically justifiable. The failure to correctly describe the normal minimum monthly ESF filter train operating times in the FSAR is another example of URI 50-395/96-05-03, Variances Between The FSAR And Plant Operating Practices For The SFP.

In the area of maintenance, two NCVs were identified.

4.0 ENGINEERING (35750, 37551, 92903)

General engineering activities were reviewed to determine their effectiveness in preventing, identifying, and resolving safety issues, events, and problems. During the outage, personnel from Licensing, Engineering, and ISEG took a proactive approach to maintain communications with the NRC resident inspectors on routine and predecisional matters on a weekly basis. The inspectors identified this effort as a positive finding.

4.1 10 CFR 50.59 Screening and Safety Evaluations

The inspectors reviewed SAP-107, 10 CFR 50.59 Unreviewed Safety Review Process, revision 0, October 1, 1990.

The inspectors reviewed Engineering's log of recent design changes and recent changes to chemistry and surveillance test procedures. Twenty-four modifications and procedure changes were selected for further evaluation to determine if an appropriate screening and safety assessment had been performed.

The inspectors found that satisfactory 10 CFR 50.59 screening and required safety assessments had been performed for the plant facility and procedure changes. These reviews referenced the appropriate FSAR and TS requirements affected by the changes to assure that these requirements were evaluated prior to making the changes. The procedure revision screenings prepared by the test group were comprehensive, very detailed in scope, and thorough.

The inspectors reviewed the log of temporary (non-permanent) modifications. These modifications were controlled by design engineering. At the time of this inspection, the log listed a total of five non-permanent modifications. Closure was in process on two of these modifications and two were minor non-safety related items.

The inspectors reviewed the modification package for one of the non-permanent modifications and verified that the modification had received an appropriate 10 CFR 50.59 safety evaluation. A walkdown inspection was made and the installation of the temporary equipment was found to conform to the requirements of the non-permanent modification package.

4.2 EFW Steam Line Supports

On May 5, 1996, the inspectors noted that five variable supports (spring cans) on the steam line to the turbine driven EFW pump were substantially off their indicated cold settings. The springs on two of the supports were fully compressed. Engineering personnel reviewed the as-found settings and concluded that they were correct. An MRF had altered the original settings but had failed to change field markings to reflect the new settings. The MRF indicated that the supports with the fully compressed springs acted as dead weight supports.

The inspectors learned through discussions with plant personnel that the supports on this steam line were not in the ASME Section XI inspection program. The FSAR described this line as part of the Main Steam System and as Class 3 piping. Discussions with NRR personnel concluded that this steam line was not required to be included in the ASME Section XI inspection program.

4.3 Close Out Issue

(Open) IFI 50-395/96-01-01, Partially Submerged Reactor Building Tendon Inspection

This item was issued to review the licensee's verification of tendon integrity during RF9 tendon testing. During a previous inspection, this tendon was identified as partially submerged in water in a containment tendon sump area. The licensee removed the end cover and inspected for water. Procedure data sheets indicated that no water was detected during loosening of the grease can, in the grease can, or around the tendon anchorage. They also removed a sample of grease for analysis. This IFI will remain open until the inspectors review the results of the analysis.

In the area of engineering, no violations or deviations were identified.

5.0 PLANT SUPPORT (71750, 82701, 83750, 84750, 86750, 92702, 92904)

5.1 General

During inspection activities and tours of the plant, the inspectors routinely observed aspects of plant support in the areas of radiological controls, physical security, and fire protection. The level of radiological protection controls applied to work activities observed was commensurate with the difficulty and risk associated with the task. Aspects of the fire protection program that were examined included transient fire loads, and fire watch patrols. Effective implementation of the physical security program continued to be demonstrated during inspectors' observations of: security badge control; search and inspection of packages, personnel, and vehicles; tours and compensatory posting of security officers; and control of protected and vital area barriers.

5.2 Fire Protection

On April 24, 1996, a discharged fire extinguisher was returned to the FPO. The FPO questioned why it had been discharged. The FPO was informed that the extinguisher had been used to extinguish a small fire on a foam protector pad for the high pressure condenser anchor bolts at the northwest corner of the main condenser in the Turbine Building. The fire occurred at 10:00 am on April 19, 1996, and was immediately extinguished by the firewatch. The firewatch failed to notify the CR as required by FPP-020, Program Administration, revision 0 Change B. FPP-020, paragraph 3.5.1.D states, in part, that the firewatch shall promptly notify the CR if a fire should occur and state: the location of the fire, the approximate size of the fire, the type of fire (e.g., specific plant equipment involved, electrical, flammable liquids involved, etc.) and the firewatch's name. This licensee identified violation for failure to follow procedure is being treated as an NCV consistent with Section VII.B.1 of the NRC Enforcement Policy. This is identified as NCV 50-395/96-07-08, Firewatch Failed To Notify Control Room Of A Fire In The Turbine Building.

5.3 NRC Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment

This Bulletin requested the licensee to ensure that the handling of heavy loads is performed safely and within the conditions and requirements specified under 10 CFR. On January 25, 1985, the licensee submitted a report to NRR which responded to the NRC's request to review their controls for the handling of heavy loads to determine the extent to which their facility satisfied the recommendations as reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The inspectors reviewed the licensee's compliance with NUREG-0612 to the extent practicable. This review included a review of Gilbert Associates, Inc. Report 2364, "Control of Heavy Loads at VCSNS" and observations of heavy lifts over the reactor vessel during RF9. The inspectors identified no concerns.

5.4 Health Physics, Chemistry and Radiological Controls

This inspection was conducted in the areas of radioactive effluent monitoring instrumentation, meteorological monitoring, primary and secondary water chemistry, radiation contamination control, solid radioactive waste management and transportation of radioactive materials.

5.4.1 Radiological Effluent Release Report and Environmental Monitoring Program

The Annual Effluent and Waste Disposal Report for 1995 dated April 24, 1996, contains data for the total body dose due to liquid effluents. There were no major changes to radioactive liquid, gaseous, and solid waste treatment systems during the reporting period (January 1 - December 31, 1995). For 1995, V. C. Summer liquid, gaseous, and particulate releases were maintained well below TS, 10 CFR 20, and 10 CFR 50 effluent requirements. The inspectors reviewed the data from the report and determined that the dose consequences to the maximum exposed member of the public in an unrestricted area were negligible and consistent with values reported in previous reports.

The Annual Radiological Environmental Monitoring Report (1995) was submitted on April 24, 1996. The maximum exposed individual was calculated to receive 1.85 mrem/year based on Operating License Environmental Report Source Term using 1995 meteorological data. No detectable fission or activation product activity attributed to the plant was observed in environmental media except for fish and sediment samples from Monticello and Parr Reservoir. A conservative calculation of radiation dose received by an exposed individual from these pathways was 1.6E-1 mrem/year. This dose represents a small fraction of the observed variation in natural background and a small fraction of the licensee's effluent dose limits. The radiological environmental data indicated that plant operations had no significant impact on the environment or public health and safety. The inspectors determined that the calculated dose was similar to values reported in previous annual reports and a small percentage of natural background doses.

5.4.2 Solid Radioactive Waste Management and Transportation of Radioactive Materials

The programs for the classification, packaging and shipment of radioactive materials were reviewed to determine the licensee's compliance with the applicable requirements of regulations in Titles 10 and 49 CFRs and facility procedures. Selected procedures for the classification, packaging and transportation of radioactive material were reviewed. The records of recent radioactive material were reviewed and radioactive waste shipments were also examined. No concerns with the licensee's procedures or records for preparation and shipment of radioactive material were identified during the reviews.

During the fuel movements associated with RF9, a grid strap on one of the fuel elements broke and was dislodged. One section of the grid strap had been retrieved and was being prepared for shipment to a laboratory for analysis. The inspectors reviewed the licensee's preliminary analysis for shipment in shipping container License Number SNM-1460 Amendment Number 05 dated August 31, 1992 (Docket 070-01503). The inspectors had no concerns.

5.4.3 Radioactive Effluent Monitoring Instrumentation

The inspectors walked down and observed the operability, material condition and local panel readouts and compared the results with the CR readouts for 23 process and effluent radiological monitors. The turbine room sump monitor had been taken out of service for plant modifications occurring in the vicinity of the monitor. Grab samples were being taken and analyzed at six-hour intervals. A review of the results of a grab sample for the Turbine Building sump taken May 1, 1996 at 2:00 pm, showed no identifiable gamma activity. Observation of the local and remote (CR) outputs from the monitors were essentially the same. The inspectors concluded that the licensee had implemented an effective program for maintaining radioactive effluent monitoring instrumentation in an operable condition and for performing the required surveillance to demonstrate operability.

5.4.4 Meteorological Monitoring Program

The licensee's 1995 Annual Radioactive Effluent Release Report provided a summary of the meteorological data collected during the year. The combined annual data recovery rate for the meteorological monitoring instruments was 94.5 percent. On May 1, 1996, the inspectors briefly witnessed a meteorological calibration while in the CR. The inspectors concluded that the licensee had implemented an effective program for collecting the required meteorological data and maintaining the meteorological instrumentation in an operable condition.

5.4.5 Primary and Secondary Water Chemistry

The inspectors reviewed six procedures for controlling the chemical environment of the primary and secondary plant systems. These procedures contain provisions for sampling and analyzing reactor coolant for the TS required parameters at the specified frequencies and for implementing (with a few minor exceptions) the EPRI guidelines for PWR primary and secondary water chemistry. The exceptions were made in accordance with guidelines established by the fuel supplier for the plant specific chemistry regimes.

The inspectors reviewed records and trend plots of the analytical results for dissolved oxygen, chloride, fluoride, and DEI in the reactor coolant. Plots of analytical results for selected parameters designated in the EPRI guidelines as control parameters for reactor coolant, feedwater, blowdown, and condensate during power operations were also reviewed. The documents reviewed included data generated during the period from November 1, 1995 through April 19, 1996. During steady state operations the dissolved oxygen concentrations, fluoride concentrations, and DEI were well within TS limits. The other parameters selected for review were maintained well within the EPRI guidelines.

The inspectors reviewed the results of early boration performed during RF9 shutdown activities. The licensee typically used early boration (acid-reducing chemistry) combined with hydrogen peroxide injection (acid-oxidizing chemistry) during unit shutdown and cooldown for refueling to reduce the source term.

The inspectors reviewed reports/evaluations of the early boration results of RF9. The process solubilized and removed approximately 1390 curies of Co-58. The Co-58 was removed via the demineralizers. The inspectors concluded that the licensee was proactive in trying to reduce dose rates by removing significant quantities of activity via the early boration/hydrogen peroxide shutdown program. The inspectors also concluded that the licensee had implemented chemistry control programs in accordance with the TS requirements and the EPRI guidelines for PWR primary and secondary water chemistry.

5.4.6 External Radiation Exposure Controls

The inspector reviewed this area to determine whether individual personnel exposures were controlled, monitored, and less than the 10 CFR Part 20 regulatory limits. The inspection included reviews of licensee procedures and records, interviews with licensee personnel, and observations made during tours of the licensee's RCAs.

5.4.6.1 Radiological Work Controls

The licensee used RWPs for incorporating job planning and radiological exposure controls into work activities performed in site RCAs. The inspectors reviewed four HPPs which outline RWP usage. The inspectors reviewed selected current and recent RWPs for their work activity and determined that they prescribed adequate radiation protection requirements for the assigned task. The inspectors observed personnel reviewing RWPs and logging onto the RWPs with the licensee's access control computer. The access computer was used to track individual personnel radiation exposures with RWP entries.

The inspectors observed plant radiation workers interacting with HP personnel at the main RCA control point. HP personnel were adequately evaluating the job scope that prescribed proper radiological protection measures and controls. A random sampling of workers were questioned about their RWPs, their work activities, and the required radiation protection practices. The inspectors concluded that the workers were knowledgeable about the requirements in the RWPs.

5.4.6.2 Radiological Postings

This area was reviewed to evaluate the licensee's use of radiological postings and to verify that postings met regulatory requirements.

The inspectors observed the licensee's radiological postings and found them conspicuous, clear, and consistent. All postings met 10 CFR Part 20 requirements. Area boundaries were clearly established.

All locked high radiation areas checked by the inspectors were properly secured. Many high radiation areas having radiation levels less than 1,000 mrem/hr were also locked to maintain personnel radiation exposures ALARA.

5.4.6.3 Radiological Surveys

The inspectors reviewed historical and current surveys to verify the licensee was performing adequate radiation and contamination surveys. During tours of the plant, the inspectors observed HP Technicians in the plant perform radiation and contamination surveys. The inspectors independently verified radiation and contamination levels in selected areas of the Auxiliary, Intermediate, Radioactive Waste, and Fuel Handling Buildings. On April 30, 1996, the inspectors (accompanied by the licensee) toured the hot machine shop on 436' elevation. The inspectors noticed labeled trash barrels outside the posted rope boundary in the vicinity of a work bench. The inspector took smears. One of the smears was taken in the vicinity of the stepoff pad between the rope and a container labeled used clothing. The smear had a measured contamination level of approximately 15,000 dpm/100cm^2 (Survey # 96-04-123). This exceeded the procedural posting limit of 1000 dpm/100cm^2, as stated in SAP-500, Health Physics Manual, revision 8, and HPP-158, Contamination Control for Areas, Equipment and Materials, revision 7. The result of the survey was identified as Violation 50-395/96-07-09, Failure to Follow SAP-500 and HPP-158 to Post and Control Contaminated Areas.

The licensee extended the rope and placed the two waste receptacles within the enlarged roped area. The licensee decontaminated the area, without obtaining any additional smears, and performed a survey of the area the same day. A post decontamination survey (# 96-04-117) showed acceptable levels.

The inspectors noted that portable radiation detectors, friskers, and contamination monitors in the plant had up-to-date calibration stickers and had been source-checked as required.

The inspectors reviewed selected current and recent records of radiation and contamination surveys and discussed the survey results with licensee representatives. The inspectors observed that work activities were performed in accordance with the surveys and RWPs.

5.5 Industrial Safety Initiatives

A number of industrial safety problems were experienced during the inspection period. However, the licensee acted effectively to maintain a safe working environment for on site personnel. These initiatives included stand downs to conduct safety briefings and publishing industrial safety notes in a refueling outage newsletter to address specific safety issues. The inspectors considered the initiatives taken by the licensee in the area of industrial safety to be a positive finding.

5.6 RF9 Exposure Control

The licensee's exposure for RF9 was approximately 89 person-Rem. This represents one third of the exposure received in their lowest exposure outage to date. The licensee cited several reasons for the low exposure which included reduced work scope for a short outage and a substantial reduction in the number of major modifications; experienced work crews, technological advances, and up-front ALARA planning for higher exposure jobs; the fine mesh filter program which has kept V.C. Summer source terms substantially below most comparable plants; the HP emphasis to perform online RHR maintenance which resulted in a substantial dose reduction; and good primary chemistry. Furthermore, the licensee's three year rolling average has decreased from 392 person-Rem in 1989 to 212 person-Rem in 1995. The inspectors identified this person-Rem reduction as a positive finding.

5.7 Close Out Issues

5.7.1 (Closed) VIO 50-395/94-03-02, Failure to Declare Portions of Fire Service System Inoperable and Implement Compensatory Actions for the Design Basis Flow Requirements.

(Closed) VIO 50-395/94-03-03, Failure to Confirm Unacceptable Fire Service System Flow Condition Existed and to Take Corrective Action in a Timely Manner.

The inspectors verified that the corrective actions described in the licensee's letter dated March 11, 1994, for these two violations were reasonable and had been completed. The corrective actions included issuance of root cause evaluation report No. 1054 which identified the root cause as personnel errors by the fire protection and design engineering staff; completion of a plant modification to remove several cable tray sprinkler nozzles on the 463 foot elevation of the Auxiliary Building to reduce the sprinkler flow demand; cleaning portions of the interior of the sprinkler system piping; training of the fire protection and engineering staff in the use of Procedures SAP-134, Control of Station Surveillance Test Activities, and SAP-1141, Non-conformance Control Program; and performance of independent assessments of the fire protection program in the areas of quality related programs, surveillance tests and tracking systems, audits, and past events. No similar fire protection related problems have been identified recently.

5.7.2 (Closed) IFI 50-395/94-09-01: Followup action subsequent to the excess free standing water event (Violation 94-09-02). The inspectors reviewed the licensee's reply to the Notice of Violation, procedural revisions, and selected data and found satisfactory implementation of the procedural requirements. There were no additional items of concern.

5.7.3 (Closed) VIO 50-395/94-09-02: Shipment in violation of 10 CFR 61.56 (A)(3), High water content with state and disposal site requirements for free standing liquid in resins shipped to the disposal site. The inspectors reviewed the licensee's reply to the Notice of Violation dated May 31, 1994, procedural revisions, and selected data and found satisfactory implementation of the procedural requirements. There were no additional items of concern.

In the area of plant support, one violation and one NCV were identified.

6.0 Other NRC Personnel On Site

George Belisle, Chief, Reactor Projects Branch 5, was on site May 22 and 23, to review resident inspectors' activities, tour the plant, and meet with licensee management.

Eugene Imbro, Director, Project Directorate II-1, NRR, and Allen Johnson, Licensing Project Manager, NRR, were on site May 7 and 8 to meet with the resident inspectors and licensee management.

7.0 Review Of FSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the FSAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the FSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the FSAR that related to the areas inspected.

NRC Inspection Report No. 50-395/96-05 identified variances between operating practices and the FSAR as URI 50-395/96-05-03. The staff notes that the URI pertains to refueling outages prior to 1996. The licensee submitted a request for license amendment on August 18, 1995, to support an increase in authorized thermal power. In the supporting analysis, the licensee stated clearly that it had chosen to perform full core offloads each refueling outage. The licensee presented an analysis of the impact of a loss of a single fuel pool cooling loop concurrent with the heat load from a full core offload. The licensee presented a justification, based on pool structural considerations, of the acceptability of the resulting bulk pool temperature of 186.1°F. The NRC issued license amendment 133 on April 12, 1996. In the accompanying safety evaluation, the staff observed that the licensee had specifically evaluated the impact of the elevated spent fuel pool temperature on (1) structural integrity of the spent fuel pool and liner, (2) spent fuel pool cooling systems piping and components, (3) spent fuel pool ventilation system, (4) margin to local boiling and (5) net positive suction head for the spent fuel pool cooling pumps. The staff found the licensee's evaluations acceptable. By issuance of the license amendment, the plant licensing basis is revised to reflect the acceptability of performing full core offloads on a routine or normal basis.

The licensee routinely offloaded the full core during each refueling outage in potential conflict with descriptions provided in their FSAR. The licensing basis for the fuel pool cooling system as described in the FSAR lists the partial core offload as the normal design basis cooling situation and a full core offload as the off-normal situation. The licensee has proposed in its power uprate amendment dated August 18, 1995, that single train operation, during full core offload, with a maximum spent fuel pool temperature of 186°F be accepted by the NRC for routine operation. The NRC issued the proposed amendment on April 12, 1996. This item will be addressed as part of URI 50-395/96-05-03.

The licensee is taking a proactive approach by forming a task team to verify that the FSAR reflects the current as-built configuration of the plant for open modification packages that have some or all of the construction complete. This effort is focusing exclusively on the FSAR and the sections that describe the configuration or contain commitments that might be affected by the modifications being reviewed. This proactive approach of reviewing the applicable portions of the FSAR that compares plant practices, procedures and/or parameters to the FSAR is considered a positive finding.

The following project objectives are a backup to design engineering, are ongoing, and were planned for completion by June 1, 1996:

Review open MRFs for incorporation of FSAR commitments with emphasis on uprate and outage MRFs.

Identify and record potential open items/ONOs detected during review.

In the area of review of FSAR commitments, no violations or deviations were identified.

8.0 Review of INPO Evaluation Report

The inspectors reviewed the INPO debrief that was prepared following the April 1996 evaluation. The debrief documented several strengths and areas for improvement. Many of the perceptions of licensee performance presented in the INPO debrief were consistent with NRC assessments documented in inspection reports.

9.0 EXIT

The inspection scope and findings were summarized on June 12 and July 1, 1996, by R. F. Aiello and T. Farnholtz with those persons indicated by an asterisk in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results. The HP, Chemistry, and Radiological controls inspection scope and results were summarized on May 3, 1996. The licensee stated that they did not agree with the proposed violation. They did not believe that the violation occurred in the context presented and requested a teleconference call with regional management. The call was held on May 13, 1996, and the licensee presented their belief that the violation was not safety significant and did not accurately portray the site contamination control program. They requested an opportunity to continue the discussion after the Region evaluated the information and made their decision. On May 16, 1996, in a teleconference call, the licensee was informed that after consideration of the material presented at the exit and the May 13, 1996 teleconference, the Notice of Violation would be issued. The licensee stated that they would deny the Violation in their response. Proprietary information is not contained in this report. A listing of inspection findings is provided.

Type	Item Number	Status	Description and Reference
NCV	50-395/96-07-01	Open/Closed	CCW Speed Switch Event (Paragraph 2.3).
NCV	50-395/96-07-02	Open/Closed	RWST Water In RHR and CS Emergency Sump (Paragraph 2.4).

Type	Item Number	Status	Description and Reference
URI	50-395/96-07-03	Open	Paint Coat in Containment Does Not Meet Application Specification (Paragraph 2.6).
URI	50-395/96-07-04	Open	Control Room Meters Reading Outside Green Indication Zone (Paragraph 2.7).
URI	50-395/96-07-05	Open	Conducting Core Alteration Without Containment Integrity (Paragraph 2.8).
NCV	50-395/96-07-06	Open/Closed	Inadvertent ESF Actuation Resulting in MSIV Closure (Paragraph 3.6).
NCV	50-395/96-07-07	Open/Closed	Source Range High Flux at Shutdown Set too High Due to Failure to Follow Procedures (Paragraph 3.7).
NCV	50-395/96-07-08	Open/Closed	Firewatch Failed To Notify Control Room Of Fire In The Turbine Building (Paragraph 5.2).
VIO	50-395/96-07-09	Open	Failure to Follow SAP-500 And HPP-158 To Post And Control Contaminated Areas (Paragraph 5.4.6.3).
VIO	50-395/94-03-02	Closed	Failure to Declare Portions Of Fire Service System Inoperable And Implement Compensatory Actions For The Design Basis Flow Requirements (Paragraph 5.7.1).
VIO	50-395/94-03-03	Closed	Failure To Confirm Unacceptable Fire Service System Flow Condition Existed And To Take Corrective Action In A Timely Manner (Paragraph 5.7.1).
IFI	50-395/94-09-01	Closed	Followup action subsequent to the excess free standing water event (Paragraph 5.7.2).
VIO	50-395/94-09-02	Closed	Shipment in violation of 10 CFR 61.56(A)(3) high water content with state and disposal site requirements for free standing liquid in resins shipped to the disposal site (Paragraph 5.7.3).

Type	Item Number	Status	Description and Reference
VIO	50-395/94-19-01	Closed	Failure to comply with a System Operating Procedure (Paragraph 2.9.1).
IFI	50-395/95-20-01	Open	Air in SW piping to suction of TDEFW pump (Paragraph 2.9.2).
IFI	50-395/96-01-01	Open	Partially Submerged Reactor Building Tendon Inspection (Paragraph 4.3).
URI	50-395/96-02-04	Closed	Review Difference Between FSAR Statements and TS Requirements For Minimum Monthly Operating Requirements of ESF Filter Trains (Paragraph 3.8).
URI	50-395/96-05-03	Open	Variances Between The FSAR And Plant Operating Practices For The SFP (Paragraphs 3.8 and 7.0).

10.0 ACRONYMS

89NA	1989 Edition No Addenda
ALARA	As Low As Reasonably Achievable
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BTU	British Thermal Unit
CCW	Component Cooling Water
CFR	Code of Federal Regulations
Co	Cobalt
CR	Control Room
CRHF	Control Room Human Factor
CS	Containment Spray
DBD	Design Basis Document
DEI	Dose Equivalent Iodine
DRP	Division of Reactor Projects
EFPH	Effective Full Power Hour
EFW	Emergency Feedwater
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ET	Eddy Current
FCV	Flow Control Valve
FPO	Fire Protection Officer
FPP	Fire Protection Procedure
FSAR	Final Safety Analysis Report
HP	Health Physics
HPP	Health Physics Procedure
I&C	Instrumentation and Control
IFI	Inspection Followup Item

INPO	Institute of Nuclear Power Operations
IPCS	Integrated Plant Computer System
IR	Inspection Report
ISEG	Independent Safety Engineering Group
ISI	Inservice Inspection
MCB	Main Control Board
mrem	Millirem
MRF	Modification Request Form
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NPF	Nuclear Production Facility [Type of license]
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ONO	Off Normal Occurrence
PDR	Public Document Room
PE	Professional Engineer
PT	Liquid Penetrant
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCA	Radiation Control Area
RC&E	Raytheon Engineers and Constructors
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REM	Roentgen Equivalent Man
RF	Refueling Outage
RHR	Residual Heat Removal
RP	Refueling Procedure
RV	Reactor Vessel
RWP	Radiological Work Permit
RWST	Refueling Water Storage Tank
SAP	Station Administrative Procedure
SCE&G	South Carolina Electric and Gas
SCDHEC	South Carolina Department of Health Environmental and Control
SFP	Spent Fuel Pool
SG	Steam Generator
SOP	System Operating Procedure
STP	Surveillance Test Procedure
SW	Service Water
T	Temperature
TDEFW	Turbine Driven Emergency Feedwater
TMI	Three Mile Island
TS	Technical Specification
URI	Unresolved Item
UT	Ultrasonic
VIO	Violation
VT	Visual
WGDT	Waste Gas Decay Tank