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

January 22, 2001

South Carolina Electric & Gas Company
ATTN: Mr. Stephen A. Byrne
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. 50-395/00-06

Dear Mr. Byrne:

On December 23, 2000, the NRC completed an inspection at your Virgil C. Summer reactor facility. The enclosed report documents the inspection findings which were discussed on January 3, 2001, with you and other members of your staff.

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

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green). These issues were determined to involve violations of NRC requirements. However, because of the very low safety significance and because these issues have been entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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 facility.

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

(ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.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: Integrated Inspection Report No. 50-395/00-06

Attachments: (1) Supplemental Information
(2) NRC's Revised Reactor Oversight Process

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-395
License No.:	NPF-12
Report No.:	50-395/00-06
Licensee:	South Carolina Electric & Gas (SCE&G) Company
Facility:	Virgil C. Summer Nuclear Station
Location:	P. O. Box 88 Jenkinsville, SC 29065
Dates:	September 24 through December 23, 2000
Inspectors:	M. Widmann, Senior Resident Inspector M. King, Resident Inspector W. Crowley, Senior Reactor Inspector, RII (Section 1R08) G. Kuzo, Senior Health Physicist, RII (Sections 2OS1, 2OS2, and 2OS3)
Approved by:	K. D. Landis, Chief, Reactor Projects Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 50-395/00-06, on 09/24 -12/23/2000, South Carolina Electric & Gas Co., Virgil C. Summer Nuclear Station. Refuel and Outage Activities, Access Control to Radiologically Significant Areas, Event Follow-Up and Licensee Identified Violations.

The inspection was conducted by resident inspectors, a regional senior reactor inspector, and a regional senior health physicist. The inspection identified two Green findings, both of which were non-cited violations. The significance of the findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process.”

A. Inspector Identified Violations

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation for failure to establish adequate procedures, as required by Technical Specificiation (TS) 6.8.1, to ensure that the pressurizer temperature heatup and cooldown limits were maintained within the requirements of TS 3.4.9.2. As a result during the shutdown for refueling outage 12, the licensee failed to recognize that the TS pressurizer temperature heatup and cooldown limits were exceeded for short period of times, i.e., less than the allowed TS action statement time.

  The finding was of very low safety significance because a licensee’s engineering evaluation, which included fracture toughness considerations, determined that the pressurizer remained acceptable for continued operation. (Section 1R20)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a non-cited violation for failure to adhere to a radiation protection procedure as required by TS 6.11, “Radiation Protection Program.” On October 26, 2000, electronic dosimeters (ED) were used as radiological controls for scaffold construction activities in a residual heat removal heat exchanger room. Contrary to a health physics procedure, ED dose rate alarm setpoints were established at 300 millirem per hour (mrem/hr) rather than greater than the 400 mrem/hr general work area dose rates adjacent to the residual heat removal heat exchangers. As a result workers were not properly responding to dose rate alarms.

  The finding was of very low safety significance because an overexposure did not result, a substantial potential for such an exposure did not exist and the licensee’s ability to assess worker’s dose was not compromised. (Section 2OS1)

B. Licensee Identified Violations

Cornerstone: Initiating Events

- To Be Determined. An apparent violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” was identified for failure to assure that the seismic design basis was properly translated into specifications, drawings, procedures, and instructions. The licensee identified that a seismic tie-back support on the steam generator B vent line was not designed or installed. A licensee's evaluation determined that the vent line could fail during a seismic event due to the missing support. The NRC's significant determination evaluation has not been completed. (Section 4OA3)

- Violations of very low significance which were identified by the licensee have been reviewed by the inspectors. The licensee has captured these issues in their corrective action program. These violations are listed in Section 4OA7 of this report.

Report Details

The unit began the inspection period at 94 percent power in power coast down to the refueling outage (RF-12). The unit commenced a planned shutdown from 84 percent power on October 6 and entered Mode 5 (Cold Shutdown) on October 8. Refueling began on October 11 and the licensee completed defueling on October 17.

Refueling of the core was delayed due to the discovery of an axial crack in the reactor coolant system (RCS) A loop hot leg. At the end of the inspection period, the unit was still defueled and repair activities for the A hot leg were partially completed. Inspection activities related to the hot leg crack are being performed by an NRC Special Inspection Team and will be documented in NRC Inspection Report No. 50-395/00-08.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed Operations Administrative Procedure OAP-109.1, “Guidelines for Severe Weather,” Revision 1C, and Electrical Maintenance Procedure EMP-120.002, “Freeze Protection Heat Tracing Inspection,” Revision 3. The review assessed the adequacy of the procedures to provide guidance for preparation and response to adverse weather conditions, including the adequacy of cold weather protection of the refueling water storage tank (RWST) and condensate storage tank level sensing lines. The inspectors conducted system walkdown inspections to assess the overall readiness of various heat tracing systems and to review the licensee’s preparation prior to the onset of cold (sub-freezing) weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

To verify that systems / components were correctly aligned, the inspectors reviewed various documents including plant procedures, drawings and the Final Safety Analysis Report (FSAR). The inspectors also reviewed outstanding maintenance work requests (MWR) and related Problem Identification Program reports (PIPs) to verify that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact mitigating system availability. In addition, the inspectors verified through plant walkdowns that with a train of equipment removed from service that the opposite train of equipment was correctly aligned, available and operable. The following systems / components were verified:

- A emergency diesel generator (EDG) (while the B EDG was out of service for preventative maintenance);

- A EDG (while the B EDG was undergoing fuel oil transfer cross-connect piping and day tank modifications);

- B EDG (while the A EDG was out of service for jacket water expansion joint modifications); and

- Spent Fuel Cooling (during full core off-load with the A EDG out of service).

The applicable portions of the following station operating procedures (SOPs), FSAR, Technical Specifications (TSs), drawings and NRC guidance were reviewed:

- SOP-306, “Emergency Diesel Generator,” Revision 14B;

- SOP-307, “Diesel Generator Fuel Oil System,” Revision 9B;

- SOP-123, “Spent Fuel Cooling,” Revision 11H;

- D-302-085, “Diesel Generator - Miscellaneous Services,” Revision 9;

- D-302-351, “Diesel Generator - Fuel Oil,” Revision 8;

- D-302-651, “Spent Fuel Cooling,” Revision 38;

- FSAR Sections 8.3.1, 9.1;

- TS Sections 3/4.8.1, 3/4.9.10; and

- NUREG-1275, Volume 12, “Operating Experience Feedback Report: Assessment of Spent Fuel Cooling,” February 1997.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed current PIPs, Work Orders (WO), and impairments associated with the fire suppression system. The inspectors reviewed the status of ongoing surveillance activities to determine whether they were current to support 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 verified proper control of transient combustibles and ignition sources.

The inspectors conducted routine inspection of the following areas:

- Emergency Safeguard Feature Transformer Area;

- Relay Room (fire zone CB-6);
- A and B Diesel Generator Rooms (fire zones DG 1.1, 1.2, 2.1 and 2.2);
- 1DA/1DB Switchgear Room (fire zone IB-20);
- Component Cooling Water Pump Areas (fire zones IB-25.1.1 and 25.1.2); and
- Turbine Building (fire zone TB-1).

The majority of these areas are important to safety based on the licensee's fire risk analysis (Individual Plant Examination for External Events (IPEEE) External Fires Request for Additional Information (RAI), dated January 1999).

The inspectors also observed Preventive Test Procedure (PTP)-114.074, "Transformer Deluge Operational Test," Revision 4, Section 6.4. This surveillance performs a spray flow test of the fire deluge system for engineered safeguard transformers and associated 7.2 KV voltage regulators.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI)

a. Inspection Scope (71111.08)

The inspectors observed in-process ISI work activities and reviewed selected ISI records. The observations and records were compared to the TS and the applicable Code (American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no Addenda). This was the last outage of the second period of the second interval. Since most of the inspection items were completed during the first two outages of the period, the current ISI was limited in scope. The ISI scope included only liquid penetrant (PT) of 27 small diameter pipe welds and eddy current (ET) inspection of all steam generator (SG) tubes.

PT inspection was observed for Chemical Volume Control System welds 1-4506A-42,1-4506A-43,1-4506A-44, and 1-4506A-45. In addition, PT records for Chemical Volume Control System Welds 1-4506A-19,1-4506A-20,1-4506A-21, 1-4506A-37, 1-4506A-38, and 1-4506A-39 and Safety Injection System Welds 1-4311-1, 1-4311-3, 1-4311-5, 1-4311-12, 1-4311-20, 1-4311-21, 1-4311-22, 1-4311-41, 1-4311-44, 1-4311-46, 1-4311-47, and 1-4311-50 were reviewed.

The scope of ET of SG tubes included bobbin coil inspection of 100% of the tubes in all three SGs. In addition, Plus Point probe inspections were performed on a small sample of tubes to address known generic problems such as row 1 and 2 u-bend problems, and for evaluation of bobbin coil indications where needed. The inspectors observed the following ET activities:

- Independent qualified data analyst (QDA) analysis of calibration group SG1CCCAL0035 tubes;

- Resolution of analyst calls for calibration groups SG1BCCAL00023, SG1CCCAL00027, SG1CCCAL00030, SG1CCCAL00035, and SG1BCCAL00024; and

- Data acquisition for a sample of tubes in all SGs.

Qualification and certification records for examiners, equipment and consumables for the above PT and ET examination activities were reviewed. In addition, PIP 0-C-00-1547, the only ISI corrective action item identified during the current outage, was reviewed. The PIP was issued to disposition surface indications identified visually during PT inspection of safety injection (SI) system pipe weld CGE-1-4506A-5.

The inspectors also reviewed ASME Section XI repair and replacement packages for the following: (1) MWR 9807713 - replacement of a 90 degree elbow in the service water system (including PIP 0-C-98-0369), (2) MWR 9903935 - replacement of a number of sections of service water piping, and (3) MWR 0014946 - replacement of EDG heat exchanger (HX) outlet expansion joint with hard pipe in accordance with Engineering Change Request (ECR)-50320.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On October 25, the inspectors observed senior reactor operators and reactor operators on the plant’s simulator during licensed operator training. The training included familiarizing operators with Emergency Operating Procedure (EOP)-2.2, “Transfer to Cold Leg Recirculation,” Revision 12 (Draft 3) and validating that the RWST swap-over to reactor building (RB) sump can be successfully completed within assumed time lines. The inspectors assessed overall crew performance and the observed licensee training and the capability of the operators to successfully perform the swap-over in the times assumed in the ECR-50308 which will be used to update the FSAR. The issues surrounding the RWST swap-over time lines are further documented in Sections 4OA3 and 4OA7.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors sampled portions of selected performance-based problems associated with structures, systems or components (SSCs), to assess the effectiveness of maintenance efforts. Reviews focused, as appropriate, on (1) maintenance rule scoping in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) or (a)(2) classifications; and (5) the appropriateness of performance criteria for SSCs classified as (a)(2) or goals and corrective actions for SSCs classified as (a)(1). The selected SSCs were the Emergency Feedwater System and the Electrical System.

For the equipment issues described in the PIPs and LER listed below, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) to determine if maintenance preventable functional failures may have existed that the licensee did not capture in their program:

- 0-C-99-0342, 0-C-00-1107, P-12 permissive failed to energize due to card IBT00422E amplifier failure;

- 0-C-99-1457, RCS Loop C OPDT 7300 Card Failure, 1TY00432B;

- 0-C-00-1235, TDEFW pump discharge isolation valve found locked closed versus locked open; and

- LER 50-395/1999008-00, Manual Reactor Trip Due to Main Turbine High Vibration, May 18, 1999.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

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 maintenance risk assessments and 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 emergent work activities listed below:

- A RCS hot leg crack at nozzle weld repair activities;

- Transformer (XTF-31, 32) work and discovery of main transformer leaks;

- RCS vent path to the primary relief tank (PRT) obstructed by collapsed diaphragm of vent path valve during RCS drain down;

- A and B EDG fuel oil system cross-connect piping modifications;

- A EDG jacket water expansion joint modification; and

- A EDG electronic governor failure and replacement.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

This inspection evaluated the licensee operator response for non-routine plant evolutions to ensure they were appropriate and in accordance with the required procedures and ensure necessary conditions were captured in the licensee's corrective action program. The following events or evolutions were reviewed:

- An inadvertent trip of XSW1EA (7.2 KV safety-related service water pump house bus) during work on XSW-1DA 04 (SW pump house bus 1EA feeder breaker) which resulted in loss of a running service water pump while in cold shutdown (reference PIP 0-C-00-1342); and

- Inadvertent over-pressurization of all three charging pumps suction piping. This condition caused operators to respond, troubleshoot, and vent the piping (reference PIP/Nonconformance Notice (NCN) 0-C-00-1759). The inspectors reviewed the licensee's root cause evaluation.

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 (LCOs) and the risk significance in accordance with the Significance Determination Process (SDP). The evaluations were contained in the following PIPs:

- 0-C-00-1367, RCS to PRT collapsed vent path valve diaphragm;
- 0-C-00-1471 Test equipment post calibration data out of tolerance following STP-230.006A, “ECCS/Charging Pump Operability Testing (Refueling),” Revision 3C;
- 0-C-00-1508, Guard pipe on containment encapsulation vessel support issue (lack of support between penetration 329 and XSM-5A-SI, protection chamber for RB recirculation sump valve XVG8811A-SI); and
- 0-C-00-1759, PIP and associated NCN for charging pump suction piping over-pressurization.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed a Station Order (SO) 00-04 which required operator actions until modification work packages on the EDG fuel oil system cross-connect and day tank level setpoint change were completed. Engineering Information Request (EIR)-80355 provided compensatory actions necessary to assure minimum EDG fuel oil day tank levels were maintained until ECR-50335 was completed. The SO also required designated operators to be capable of responding to the diesels within 15 minutes if a diesel should start for any reason. The SO use was limited to defueled conditions and required prior operations management approval for use in Mode 6 (Refueling).

This review was to determine whether the functional capability of the related system or human reliability in responding to an initiating event was affected by the operator workaround. The inspectors specifically considered whether the workaround affected the operators’ ability to implement abnormal or emergency operating procedures for the modes of operation involved.

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 either witnessed 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:

- EIR-80396B, Retest requirements for the control relays replaced during RF-12 on the B EDG (replace control relays on B EDG per NCN 0-C-97-1334 for XCX5202, Diesel Generator B Control Cubicle, WO 9917653);

- MMP-180.008, Revision 8A, "Emergency Diesel Lube Oil System Maintenance" EDG A, (replacement of a lube oil relief valve, PMT for WO 0014994);

- MMP-180.033, Revision 9, "Emergency Diesel Generator Miscellaneous Maintenance and Sections 7.50 and 7.51 "PMT" (Attachment XV), adjusting maximum kilowatt load of B EDG (B EDG replace jacket water heat exchanger per ECR-50056; various preventive maintenance items performed during refueling outage WOs 9709153 and 9709154. Post maintenance run per SOP-306, "Emergency Diesel Generator," Revision 14);

- MMP-460.024, Revision 5A, "Testing and Balancing of HVAC System and Components" (Change Charcoal for Control Room Filter Plenum B, WO 0013286);

- STP-230.006A, Revision 3C, "ECCS/Charging Pump Operability Testing (Refueling)" (A, B, C Charging pump tests, retests of B and C charging pumps following several test instruments being out of tolerance during post test calibrations, PIP 0-C-00-1471); and

- STP-401.003, Revision 9, "Code Relief Valves ASME XI Test" (ASME Section XI code relief valve (retest after initial failure) for XVR08865-SI, Hot Leg Injection relief valve, WO 9914760).

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

.1 Routine Outage Activities

a. Inspection Scope

The unit began a refueling outage on October 7, and was still in progress at the end of the inspection period on December 23. 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 (Independent Safety Engineering Group of Refuel-12 Pre-Outage Schedule Safety Review Report, dated August 31, 2000) 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 confirmed that when the licensee removed equipment from service 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.

For selected components which were removed from service, the inspectors verified that tags were properly hung and that associated equipment was appropriately configured to support the function of the clearance. In addition, the inspectors reviewed the circumstances surrounding the tagging operation sequencing which resulted in the three charging pumps inadvertently being overpressurized. The inspectors also reviewed the licensee's determination that there was no resultant impact on the system, piping, or the associated valves and flanges and the system remained operable.

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 verified that switchyard activities were controlled commensurate with the safety and 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 verified that FSAR commitments and TS requirements for spent fuel pool cooling were met;

- 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; and,

- The inspectors observed, monitored and reviewed licensee activities associated with the RCS A hot leg loop axial crack identified at the beginning of the outage; This issue is being reviewed by a NRC Special Inspection Team and will be documented in NRC Inspection Report No. 50-395/00-08.

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. A sampling of PIPs that were specifically reviewed by the inspectors are listed below. The PIPs identified below were initiated during the refueling outage and were considered significant by the licensee:

- 0-C-00-1324, boric acid leak and accumulation in A RCS loop hot leg air boot;

- 0-C-00-1392, crack in A RCS hot leg piping at the weld between the piping and the reactor vessel nozzle;

- 0-C-00-1406, breaker tripped open for FCV00602A-RH, RHR pump A miniflow valve, during XVG8706A, charging/safety injection pump suction header residual heat header inlet valve, differential pressure (DP) testing;

- 0-C-00-1723, during the performance of STP-125.017, “A Train Blackout Test,” Revision 2E, Relay 87T1EA1 tripped locking out XSW 1EA1, 7.2 KV switchgear for service water; and

- 0-C-00-1827, spent fuel pool licensing requirement (Safety Evaluation Report for License Amendment 27, dated 9/27/84) is not contained in fuel handling procedures. The procedures do not contain a requirement that initially discharged spent fuel assemblies may only be moved into regions 2 or 3 after the core is reloaded.

b. Findings

A non-cited violation was identified. During the shutdown of the unit, the inspectors observed and reviewed portions of the cooldown process to verify that TS cooldown / heatup restrictions were followed. These limits were controlled through station operating procedure SOP-101, “Reactor Coolant System,” Revision 22C, and verified through performance of STP-103.001, “Reactor Coolant System and Pressurizer Heatup / Cooldown Surveillance,” Revision 7. A review of the completed test surveillance procedure indicated that no limits had been exceeded. However, the inspectors later reviewed plant computer data which indicated that the heatup and cooldown limits on the pressurizer were exceeded for short durations, for several consecutive minutes. TS 3.4.9.2, “Pressurizer,” action statement required the licensee to restore pressurizer temperatures to within specified limits within 30 minutes and perform an engineering evaluation to determine the impact of the out-of-limit conditions on the fracture toughness properties of the pressurizer. The inspectors verified temperatures were restored within limits and that the required engineering evaluation was completed prior to increasing pressurizer pressure above 500 psig. The licensee determined that the pressurizer remained acceptable for continued operation. This analysis was consistent with the results of earlier evaluations performed in 1994 and 1991 for similar thermal cycling events.

This condition of thermal cycling of the pressurizer, if left uncorrected, could become a more significant concern and could cause an increase in the frequency of the an initiating event due to vessel toughness fracture concerns. However, the inspectors and a NRC Region II Senior Reactor Analyst (SRA) determined through the SDP process that with no actual loss of safety function, and only a slight potential for affecting initiating event frequency (LOCA) this finding would have a negligible effect on CDF.

This NRC identified finding was determined to be of very low safety significance (Green).

TS 6.8.1, “Procedures and Programs,” required, in part, that procedures be established, implemented, and maintained covering surveillance and testing activities and activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. The licensee established STP-103.001 and SOP-101 to maintain temperature limits for the pressurizer. Instructions for filling and cooldown of the pressurizer are contained in Section B of SOP-101. A note in this procedure states that a 200°F decrease in pressurizer temperature should not be exceeded in any one hour period. No other specific instructions were provided to prevent exceeding the TS limit on pressurizer cooldown rate. Based on the review of the pressurizer data, the inspectors determined that the surveillance test procedure was inadequate, in that, the licensee failed to recognize that pressurizer heatup and cooldown limits had been exceeded using the existing surveillance procedure STP-103.001. In addition, the licensee failed to properly control the temperature while implementing SOP-101. This NRC identified issue is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 50-395/00006-01. This condition is documented in the licensee’s corrective action program as PIP 0-C-00-1564.

1R22 Surveillance Testing

a. Inspection Scope

For the 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-125.004B, “Diesel Generator B Load Rejection Test,” Revision 0B;
- STP-125.009, “Diesel Generator B Refueling Operability Test,” Revision 6B;
- STP-125.010, “Integrated Safeguards Test Train A,” Revision 8B;
- STP-125.011, “Integrated Safeguards Test Train B,” Revision 8B;
- STP-130.005C, “Component Cooling Valve Operability Test (Mode 5),” Revision 3, (Section 6.8, for XVC09632-CC / XVC09633-CC, Reactor Building Return Header Check Valves);
- STP-225.001A, “Diesel Generator Support System Pump and Valve Test,” Revision 6, for air start valve test and fuel oil transfer pump test for A EDG;
- STP-230.006A, “ECCS/Charging Pump Operability Testing (Refueling),” Revision 3C, for C Charging Pump; and
- STP-454.002, “Control Room Emergency Air Cleanup System Performance Test,” Revision 3.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modifications to assess the impact on risk-significant SSC parameters, such as, availability, reliability and functional capability. The inspectors verified the temporary modifications had not adversely affected safety functions of required systems:

- NCN 0-C-00-1459 disposition, relief valve XVR08865-SI, hot leg injection header relief valve, a temporary blind flange installed to allow the plant to establish a RCS drain path (reference Technical Work Review (TWR) RB14809); and

- ECR-50432, installation of shims on steam generator A and hot leg piping.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

Radiological controls for the following refueling outage activities were reviewed:

- Radiation Work Permit (RWP) 00-00074, All Work Associated with Valves, Pumps, and Motors Not Covered by a Specific RWP;

- RWP 00-00077, Replace Valve XVT08362B-CS RB 438 Elevation (B Loop) Near Reactor Coolant Pump;

- RWP 00-00080, Reactor Vessel Split Pin Work;

- RWP 00-00089, Steam Generator Eddy Current Work;

- RWP 00-00090, Steam Generator Sludge Lancing and Miscellaneous Work; and

- RWP 00-00135, Repack Valve XVG08706B-RH.

For the subject tasks, the inspectors reviewed administrative and engineering controls for high radiation, locked-high radiation, and very high radiation areas. The reviews included, where applicable, verification of administrative and physical controls, and direct observation of pre-job briefings, work-in-progress, and Health Physics job coverage. Radiation surveys results for containment and auxiliary building work areas were verified. Controls implemented for areas having significant dose rate gradients, transient high dose rates, or potential for elevated airborne radioactive material concentrations were reviewed. Reviewed procedures included HPP-401, "Issuance, Termination, and Use of RWPS and SRWPS," Revision 13, and General Employee, Station Orientation Training (SOT)-03, "Training to Qualify for Unescorted Access to the Radiation Controlled Area (Site Specific Radiation Protection)," Revision 15.

Licensee activities were reviewed against TS and 10 CFR Part 20 requirements.

b. Findings

A non-cited violation was identified for failure to establish electronic dosimeter (ED) dose rate alarm setpoints in accordance with health physics procedures. On October 26, the inspectors observed radiation controls, including the use of EDs, during scaffold construction in the B residual heat removal (RHR) heat exchanger room conducted under RWP 00-00074/001, "All Work Associated with Valves, Pumps, and Motors not covered by a specific RWP." The October 22 survey map, used to brief the workers, documented 400 millirem per hour (mrem/hr) general area dose rates for several locations 30 centimeters from the heat exchanger. The licensee set the ED dose rate alarm setpoints at 300 mrem/hr rather than greater than the work area general dose rates, i.e., 400 mrem/hr, as required by HPP-401, paragraph 4.1.5.B. Subsequent to the work crew entering the room and proceeding to the far end of the heat exchanger, the inspector heard several workers' EDs alarm in the dose rate mode. Dose rate alarms continued to sound intermittently as individuals moved about both the work general area and in close proximity to equipment during the 20 minute task. A health physicist technician (HPT) was assigned to provide worker radiation protection coverage but was not dressed-out to enter the work area. The HPT allowed workers to continue to receive sustained exposures at unknown levels above 300 mrem/hr which should have required a more detailed survey and an evaluation of stay time.

The following concerns were identified and discussed with licensee management. At the sound of the ED alarms, no individuals were observed backing out of their immediate work locations to allow the ED dose rate alarms to clear. After the inspector requested the HPT to conduct measurements of specific work location dose rates, extension of the teletector probe into the work areas resulted in a failed detector and required the technician to leave the room for several minutes to obtain new monitoring instrumentation. Although selected individuals' EDs continued to alarm intermittently in dose rate mode at that time, the inspector observed that the workers did not cease work, did not check their ED readouts, and did not move out of their immediate work locations to lower dose rate areas to clear the alarms.

The integrated dose and dose rate data for the workers was discussed and reviewed. Early in the task, the HPT identified an individual whose ED was in dose rate alarm continuously for approximately three minutes and directed the individual to leave the room. Dosimetry data indicated the individual received an integrated dose of approximately 83 mrem and had been in a maximum dose rate field of 736 mrem/hr. The 83 mrem was the highest dose received by any of the workers.

The failure to follow procedure HPP-401 was considered more than minor, in that, if left uncorrected, the same issue could become a more significant safety concern. Improper setting of ED alarm setpoints resulted in frequent alarms which workers did not properly respond to and continued to work while sustaining exposures exceeding the setpoint with no specified stay time. Significant additional radiation dose can result from workers ignoring ED alarms. The issue affected the Occupational Radiation Safety Cornerstone, in that, the examples involved the failure of radiation barriers that could result in a significant or unplanned dose. Consequently, this issue was screened by the Occupational Radiation Safety Significance Determination Process and was determined to be of very low safety significance (Green) because: 1) there was no overexposure of workers; 2) there was no substantial potential for such an exposure; and 3) the licensee's ability to assess dose to the workers did not fail.

The failure of the licensee to establish ED dose rate alarm setpoints based on the 400 mrem/hr general work area dose rate was a violation of HPP-401 and TS 6.11, "Radiation Protection Program." The failure to implement a radiation protection procedure as specified in TS 6.11 is being disposition consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 50-395/00006-02. This issue was entered into the licensee's corrective action program as PIP 0-C-00-1595.

2OS2 "As Low As Is Reasonably Achievable" Program Planning and Controls

a. Inspection Scope

The inspectors reviewed the plant collective exposure history, current exposure trends and ongoing high dose rate and high person-rem exposure activities. Site specific trends in collective exposures and source term data were reviewed and discussed. Licensee dose reduction initiatives and program for estimating and tracking department and job specific dose expenditures were reviewed. Engineering controls were verified. Worker performance and knowledge, health physics technician proficiency, and supervisory oversight in reducing occupational dose during the current refueling outage were evaluated. Licensee's "As Low As Is Reasonably Achievable" program job evaluations, and estimated dose budgets were compared with actual dose expenditures.

Estimated dose budgets and current expenditures as of October 25, for the following RWPs were reviewed and discussed in detail.

- RWP 00-00071, RVLIS, CRDM, Detension/Tension, Nozzle Covers;
- RWP 00-00072, All Work Associated with C Reactor Coolant Pump Seal Replacement;
- RWP 00-00080, Reactor Vessel Split Pin Work;
- RWP 00-00089, Steam Generator Eddy Current Work; and

- RWP 00-00090, Perform Sludge Lancing on A, B, and C Steam Generators.

The inspectors discussed positive whole-body count analyses conducted for potential radionuclide intakes and subsequent internal dose assessments. In addition, the status of skin and "hot particle" contamination events as of October 25, associated with the current refueling outage activities were reviewed and discussed.

Guidance documents and their implementation were reviewed against FSAR, TS, and 10 CFR Part 20 requirements.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring and Protection Equipment

a. Inspection Scope

Availability and operability of personnel radiation survey instruments and equipment were evaluated during the week of October 23. Whole body count analysis equipment, both "fast-scan" and "chair" geometries, calibrations were verified. Calibrations were also verified for five portable radiation monitoring instruments, including RO-2 and teletector detectors.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Safety System Unavailability - Emergency AC Power System

a. Inspection Scope

The inspectors verified the accuracy of the "Emergency AC Power System Unavailability" PI through the third quarter year 2000. The inspectors reviewed selective samples of station logs, removal and restoration logs, LERs, and corrective action program database and discussed system unavailability tracking with the system engineer and PI coordinator for the period of January through September 2000.

b. Findings

No findings of significance were identified.

.2 RCS Leak Rate

a. Inspection Scope

The inspectors verified the accuracy of the PI through the third quarter year 2000 for "RCS Leak Rate." The inspectors reviewed selective samples of station logs, RCS leak rate surveillance test procedures, TS requirements and corrective action program database for the period of January through October 2000. Early in the inspection period the inspectors observed performance of the surveillance activity (STP-114.002, "Operational Leakage Test," Revision 10) that determines RCS identified leakage rate.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) LER 50-395/1999004-01: Fuel assembly top nozzle holddown spring failure. (Note: This is the second LER issued as Supplement 1. The original LER and Supplements 1 and 2 were closed in NRC Integrated Inspection Report No. 50-395/99-09, Section E8.1. Due to a licensee error in labeling supplemental information letters as Supplements to the original LER, the licensee has issued this Supplement as LER 50-395/1999004-01.)

This LER Supplement documented the identification that 29 twice burned fuel assemblies had top nozzle holddown spring screws that were failed or degraded. These assemblies were identified following offload of the Cycle 12 core. Visual inspection of once burned assemblies indicated that there were no failed holddown spring screws. Replacement of 24 top nozzle blocks on the twice burned fuel assemblies planned for reinsertion in the Cycle 13 core was completed successfully. This LER Supplement also formally documented the root caused determined by Westinghouse. The licensee documented the conditions identified in their corrective action program as PIP 0-C-00-1494.

The inspectors observed associated refuel activities in the spent fuel pool and reviewed the LER corrective actions documented. No findings or issues of significance were identified. This event did not constitute a violation of NRC requirements.

.2 (Closed) LER 50-395/1999005-01: ESF components potentially outside the design basis of the plant. The original LER was previously closed in NRC Integrated Inspection Report No. 50-395/99-03, Section E8.1. The subject LER revision describes licensee follow-up investigations and evaluations related to changing the emergency core cooling system (ECCS) pump suctions from the RWST to the RB recirculation sump, i.e., transfer to cold-leg recirculation. The licensee determined that the time to perform EOP-2.2, "Transfer to Cold-Leg Recirculation," Revision 11 and earlier revisions, was significantly longer than the approximately two minutes stated in FSAR 6.3.2.6, "System Design - Coolant Quantity." As a result, adequate suction pressure to the ECCS pumps could be lost and the pumps potentially damaged before the transfer is completed. Although the plant's design basis does not allow RB accident pressure to be credited for preventing a loss of suction pressure, the licensee demonstrated that RB pressure could be available to preclude pump damage. The licensee revised EOP-2.2 and demonstrated on their simulator that operators could successfully perform the transfer without reliance on RB accident pressure.

The reported condition involved an inadequate procedure which had a credible impact on safety, i.e., if RB accident pressure was not available (loss of containment integrity), following EOP-2.2 could result in damage to the ECCS pumps and a loss of their safety function to cool the reactor core after an accident. A significance determination process Phase III evaluation determined that the inadequate procedure was of very low safety significance due to the low initiating event frequency of a large break loss of coolant accident with a loss of containment integrity. This issue is dispositioned in Section 4OA7.

.3 (Closed) LER 50-395/2000005-00: Voluntary Report: Initiation of plant shutdown due to inoperable feedwater isolation valve. The licensee documented the plant being taken offline (i.e., Mode 2) to address and correct a failed actuator assemble for a feedwater isolation valve, XVG01611C. The cause of the failed actuator was attributed to a failure of a vitron poppet seal. The licensee was able to repair the valve without incident. The licensee documented this issue in PIP 0-C-00-0728 and NCN 0-C-00-0745.

This issue was previously reviewed and documented by the inspectors in NRC Integrated Inspection Report No. 50-395/00-04, Sections 1R13 and 1R23. No findings or issues of significance were identified. This event did not constitute a violation of NRC requirements.

.4 (Closed) LER 50-395/2000007-00: Main steam system support found missing. The licensee identified that inside the RB that a B steam generator vent valve line did not have a tie-back seismic support as was the case for the similar vent valve line on steam generators A and C. The licensee determined that the missing support could have caused the vent line to fail during a seismic event resulting in a small line steam break directly off the main steam header. The licensee captured this issue in their corrective action program as NCNs 0-C-00-1019 and 0-C-00-1359. The inspectors verified that a seismic support was installed on the line during refueling outage RF-12.

The inspectors' significance determination process Phase I evaluation indicated that a more detailed Phase III evaluation was necessary. Contrary to 10 CFR 50, Appendix B, Criterion III, "Design Control," the seismic design basis of the plant was not translated specifications, drawings, procedures and instructions, in that, a support was never designed to prevent failure of the B steam generator vent valve line during a seismic event. Pending completion of the Phase III evaluation to determine the safety significance of this violation, this item is identified as an apparent violation (AV) 50-395/00006-03.

.5 (Closed) LER 50-395/2000009-00: TS prohibited operation - fuel handling building (FHB) ventilation outside required range. The licensee discovered on October 16, while in Mode 6, that the FHB differential pressure was less than the TS surveillance requirement contained in 4.9.11.d.3, while fuel handling activities were in progress. The fuel handling activities were immediately suspended until a more effective ventilation configuration was established and tested. This issue was reviewed by the resident inspectors and an SRA who concluded since the FHB ventilation system was still operating (although with a slightly reduced negative pressure) that the violation of TS minimum DP was of very low safety significance. This violation is dispositioned in Section 4OA7.

.6 A Reactor Coolant System (RCS) Hot Leg Crack Event

The licensee reported in accordance with the requirements of 10 CFR 50.72 (reference LER 50-395/2000008-00) the discovery of an axial crack in the A RCS hot leg. Event follow-up and LER closeout, including any findings, will be completed and documented by a NRC Special Inspection Team in NRC Inspection Report No. 50-395/00-08.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. S. Byrne and other members of the licensee's staff on January 3, 2001. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

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

	NCV Tracking Number	Requirement Licensee Failed to Meet
(1)	NCV 50-395/00006-04	TS 6.8.1.a, requires that written procedures shall be established, implemented and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Loss of coolant accidents are an activity covered in Appendix A, under Section 6, "Procedures for Combating Emergencies and Other Significant Events." This requires appropriate procedures to respond to and combat emergencies involving loss of coolant accidents and the associated response involving transfer to cold leg recirculation. The licensee failed to establish, implement and maintain an adequate Emergency Operating Procedure EOP-2.2, "Transfer to Cold-Leg Recirculation," Revisions 0 thru 11, in that, they did not provide the necessary instructions to operators for timely actions. This issue is captured in the licensee's corrective action program as PIPs 0-C-99-1026 and 0-C-00-1101.

(2)	NCV 50-395/00006-05	TS 4.9.11.d.3 surveillance requirement states that the spent fuel ventilation shall maintain the spent fuel area at a negative pressure greater than or equal to 1/8 inches water gauge relative to the outside atmosphere during irradiated fuel movement and during crane operation with loads over the pool. Contrary to that requirement on October 16, 2000, the licensee discovered that fuel movement had occurred without the proper fuel handling building negative pressure. The failure to meet this TS requirement is documented in the licensee's corrective action program as PIP 0-C-00-1455.

Attachment 1

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Archie, Manager, Planning & Scheduling
F. Bacon, Manager, Chemistry Services
S. Bailey, Supervisor, Plant Support Engineering
L. Blue, Manager, Health Physics and Radwaste
M. Browne, Manager, Nuclear Licensing and Operating Experience
R. Cabin, Level III Examiner
R. Clary, Manager, Plant Life Extension
C. Fields, Manager, Quality Systems
M. Fowlkes, Manager, Operations
G. Halnon, General Manager, Engineering Services
L. Hipp, Manager, Nuclear Protection Services
T. McAlister, Supervisor, Quality Control
G. Moffatt, Manager, Design Engineering
K. Nettles, General Manager, Nuclear Support Services
A. Rice, Manager, Plant Support Engineering
J. Weatherford, Eddy Current QDA Level III Examiner
R. White, Nuclear Coordinator, South Carolina Public Service Authority
B. Williams, General Manager, Nuclear Plant Operations
G. Williams, Manager, Maintenance Services

NRC

R. Bernhard, Region II SRA
W. Rogers, Region II SRA

ITEMS OPENED AND CLOSED

Opened and Closed

50-395/00006-01	NCV	inadequate surveillance test and system operating procedures to control pressurizer temperature limits (Section 1R20)
50-395/00006-02	NCV	improper implementation of TS radiation protection program requirements (Section 2OS1)
50-395/00006-03	AV	failure to install a steam generator vent line support (Section 4OA3.4)

50-395/00006-04	NCV	inadequate emergency operating procedure for transfer to cold-leg recirculation (Section 4OA7)
50-395/00006-05	NCV	fuel handling building negative pressure exceeded TS requirement (Section 4OA7)

Closed

50-395/1999004-01	LER	fuel assembly top nozzle holddown spring failure (Section 4OA3.1)
50-395/1999005-01	LER	ESF components potentially outside the design basis of the plant (Section 4OA3.2)
50-395/2000005-00	LER	voluntary report: initiation of plant shutdown due to inoperable feedwater isolation valve (Section 4OA3.3)
50-395/2000007-00	LER	main steam system support found missing (Section 4OA3.4)
50-395/2000009-00	LER	TS prohibited operation - fuel handling building ventilation outside required range (Section 4OA3.5)

DOCUMENTS REVIEWED (Section 1R08 only)

ISI-3, “ASME Section XI Inservice Examination Manual For 2nd Inspection Interval,” Revision 1;

SCE&G-PT-89-1, “Liquid Penetrant examination - Solvent Removable, Visible Dye Technique,” Revision 1;

SAP-643, “ASME Code, Section XI Repair / Replacement Program,” Revision 4;

Framatome Technologies Document 51-5007713-01, “VC Summer Steam Generator Degradation Assessment;”

SAP-1141, “Nonconformance Control Program,” Revision 8;

ES-509, “Disposition of Site Nonconformances,” Revision 6;

Independent QDA Guideline OMT-00-900-01, dated October 20, 2000

ES-804.901, “Steam Generator Tube Inspection,” Revision 5;

ES-560.215, "Steam Generator Tube Inspection Eddy Current Data Analyst Guidelines," Revision 2.

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
AV	Apparent Violation
CB	Control Building
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Control Room
CRDM	Control Rod Drive Mechanism
DG	Diesel Generator
DP	Differential Pressure
ECCS	Emergency Core Cooling System
ECR	Engineering Change Request
ED	Electronic Dosimeter
EDG	Emergency Diesel Generator
EIR	Engineering Information Request
EOP	Emergency Operating Procedure
ESF	Emergency Safeguard Feature
ET	Eddy Current Testing
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report
HPP	Health Physics Procedure
HPT	Health Physics Technician
HR	Hour
HX	Heat Exchanger
IB	Intermediate Building
IR	Inspection Report
ISI	Inservice Inspection
LBLOCA	Large Break Loss of Coolant Accident
LCO	Limiting Conditions for Operations
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MMP	Mechanical Maintenance Procedure
MREM	Millirem
MWR	Maintenance Work Request
NCN	Non-Conformance Notice
NCV	Non-Cited Violation
NPF	Nuclear Power Facility [Type of license]
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NUREG	NRC Technical Report Designation
PI	Performance Indicator
PIP	Problem Identification Program
PMT	Post Maintenance Testing
PRT	Primary Relief Tank
PT	Penetrant

QDA	Qualified Data Analyst
RB	Reactor Building
RBCU	Reactor Building Cooling Unit
RCS	Reactor Coolant System
RF	Refueling Outage
RHR	Residual Heat Removal
RII	Region II [NRC]
RVLIS	Reactor Vessel Level Indication System
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
SI	Safety Injection
SCE&G	South Carolina Electric and Gas
SDP	Significance Determination Process
SG	Steam Generator
SO	Station Order
SOP	Station Operating Procedure
SOT	Site Orientation Training
SRA	Senior Reactor Analyst
SRO	Senior Reactor Operator
SSCs	Structures, Systems or Components
STP	Surveillance Test Procedure
SW	Service Water
TB	Turbine Building
TDEFW	Turbine Driven Emergency Feedwater
TS	Technical Specification
TWR	Technical Work Record
VIO	Violation
WO	Work Order

Attachment 2

NRCs REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.