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

Report Nos.: 50-424/89-33 and 50-425/89-38

Licensee: Georgia Power Company
P.O. Box 1295
Birmingham, AL 35201

Docket Nos.: 50-424 and 50-425 License Nos.: NPF-68 and NPF-81

Facility Name: Vogtle Units and 2

Inspection Conducted: October 28 - December 1, 1989

Inspectors: Leg L Tescine for 12-28-89
J. F. Rogge, Senior Resident Inspector Date Signed

Leg L Tescine for 12-28-89
R. F. Aiello, Resident Inspector Date Signed

Accompanied by: R. D. Starkey

Approved By: ARHerdt 12/29/89
K. E. Brockman, Chief Date Signed
Reactor Projects Section 3B
Division of Reactor Projects

SUMMARY

Scope: This routine inspection entailed resident inspection in the following areas: plant operations, radiological controls, maintenance, surveillance, security, technical support, and quality programs and administrative controls affecting quality.

Results: One violation was identified in the area of administrative controls affecting quality. This violation involved a failure to ensure legibility of control room drawings (paragraph 2.b(1)).

The report notes responsiveness of the radiological controls department in lowering the alarm setpoints of the personal dosimetry devices to afford better control (paragraph 2.b(4)).

DETAILS

1. Persons Contacted

Licensee Employees

*J. Aufdenkampe, Plant Engineering Supervisor
*G. Bockhold, Jr., General Manager - Nuclear Plant
C. Coursey, Maintenance Superintendent
*G. Frederick, Safety Audit and Engineering Group Supervisor
H. Handfinger, Manager Maintenance
*W. Kitchens, Assistant General Manager - Plant Operations
*R. Legrand, Manager - Chemistry and Health Physics
*G. McCarley, Independent Safety Engineering Group Supervisor
*A. Mosbaugh, Plant Support Manager
W. Mundy, Quality Assurance Audit Supervisor
*R. Odom, Nuclear Safety and Compliance Manager
*J. Swartzwelder, Manager - Operations

Other licensee employees contacted included technicians, supervisors, engineers, operators, maintenance personnel, quality control inspectors, and office personnel.

*Attended Exit Interview

An alphabetical list of acronyms and initialisms is located in the last paragraph of the inspection report.

2. Operational Safety Verification - (71707)(93702)

The facility began this inspection period with both units at full power.

Unit 1:

The unit remained at full power with the exception of minor power reductions for maintenance through the end of this inspection period.

Unit 2:

On November 5, 1989, an operator manually tripped the reactor due to decreasing levels in all four SGs. The loss of level in the SGs was due to the tripping of "B" MFP on loss of suction pressure. Later that same day, the unit entered Mode 2 (Start Up). On November 6, the reactor went critical, entered Mode 1 (Power Operation), and synchronized to the grid. The unit remained at full power with the exception of minor power reductions for maintenance through the end of this inspection period.

On November 26, 1989, during the performance of a surveillance on containment radiation monitor 2RE-003, a CVI occurred.

a. Control Room Activities

Control Room tours and observations were performed to verify that facility operations were being safely conducted within regulatory requirements. These inspections consisted of one or more of the following attributes, as appropriate at the time of the inspection:

- proper control room staffing;
- control room access and operator behavior;
- adherence to approved procedures for activities in progress;
- adherence to technical specification LCOs;
- observance of instruments and recorder traces of safety-related and important-to-safety systems for abnormalities;
- review of annunciators alarmed and action in progress for correction;
- control board walkdowns;
- safety parameter display and the plant safety monitoring system operability status;
- discussions and interviews with the On-Shift Operations Supervisor, Shift Supervisor, Reactor Operators, and the Shift Technical Advisor (when stationed) to determine the plant status, plans, and to assess operator knowledge; and
- review of the operator logs, unit logs, and shift turnover sheets.

No violations or deviations were identified.

b. Facility Activities

Facility tours and observations were performed to assess the effectiveness of the administrative controls established by direct observation of plant activities, interviews and discussions with licensee personnel, independent verification of safety systems status and LCOs, licensee meetings, and facility records. During these inspections, the following objectives were achieved:

(1) Safety System Status - Confirmation of system operability was obtained by verification that flowpath valve alignment, control and power supply alignments, component conditions, and support systems for the accessible portions of the ESF trains were proper. The inaccessible portions are confirmed as availability permits.

On November 22, 1989, the control room drawings were inspected for legibility. The inspection included a review of a major portion of Unit 1 and 2 control room drawings. The following drawings and as-built-notices were determined to have legibility problems severe enough to restrict their use by control room personnel:

ABN 87-V1E0325A002 T	1X4DB164-2	1X6AA02-234-7
ABN 87-01000A351 T	1X4DB167-1	2X3D-AA-A005
1X4DB162-1	1X4DB167-2	2X3D-AA-B05A
1X4DB162-2	1X4DB167-4	2X3D-AA-F16A
1X4DB163-1	1X4DB171-4	2X3D-AA-F24A
1X4DB163-2	2X4DB179-2	2X3D-AA-G060
1X4DB163-5	1X6AA02-228-7	2X3D-AA-H01B
1X4DB163-6	1X6AA02-239-7	
1X4DB163-7	1X6AA02-236-7	

The inspector's initial findings were elevated to the General Manager and NRC Regional Management. The General Manager directed a review to commence immediately. On November 27, 1989, the licensee discussed its findings with NRC regional management. These findings indicated that approximately 5 to 7 percent of control room and clearance tagging drawings were illegible. The cause of illegibility is attributed to poor reproduction and poor drafting. With the exception of two drawings for which aperture cards had to be remade, all drawings in these two areas were corrected. The licensee also established a long-term corrective action to review all satellite document stations.

Administrative procedure 00101-C, "Drawing Control," Rev. 7, Step 3.4.4, requires that drawing legibility be ensured prior to distribution, and engineering procedure 50009-C, "As-Built Notices," Rev. 7, Step 4.6.3, requires ABNs to be legible and reproducible.

Failure to ensure the legibility of control room drawings constitutes a violation of administrative procedure 00101-C and engineering procedure 50009-C.

This violation is identified as 50-424/89-33-01 and 50-425/89-38-01, "Failure To Implement Procedures 00101-C And 50009-C Concerning Legibility Of Control Room Drawings."

(2) Plant Housekeeping Conditions - Storage of material and components and cleanliness conditions of various areas throughout the facility were observed to determine whether safety and/or fire hazards existed.

(3) Fire Protection - Fire protection activities, staffing, and equipment were observed to verify that fire brigade staffing was appropriate and that fire alarms, extinguishing equipment, actuating controls, fire fighting equipment, emergency equipment, and fire barriers were operable.

(4) Radiation Protection - Radiation protection activities, staffing, and equipment were observed to verify proper program implementation. The inspection included a review of the plant's program effectiveness. Radiation work permits and personnel compliance were reviewed during the daily plant tours. Radiation Control Areas were observed to verify proper identification and implementation.

On November 8, 1989, the manager of radiological protection informed the inspector of new personnel dosimetry alarm setpoints which would be utilized in controlling dose. When a worker receives a dosimetry alarm, he is to leave the radiological area and report to health physics. In the past, the alarm was set at 200 mrem. During a recent maintenance activity, a worker received 135 mrem dose when a 40 mrem dose had been expected. The radiation work permit allowed for a 200 mrem dose. The inspector noted to the licensee that a fixed alarm at 200 mrem did not function to limit dosage. For example, a visitor would be expected to get no more than 1 mrem. In response to the issue, the licensee pursued a change to the computer software and implemented the following:

RWP No.	Dose Alarm
20, 21, 23, 24, 26, 27	25 mrem
65 (Visitor)	10 mrem
Others	1/20 of remaining quarter (50 mrem)

The inspector considers the responsiveness of the radiological department to be noteworthy.

(5) Security - Security controls were observed to verify that security barriers were intact, guard forces were on duty, and access to the Protected Area was controlled in accordance with the facility security plan. Personnel were observed to verify proper display of badges and that personnel requiring escort were properly escorted. Personnel within Vital Areas were observed to ensure proper authorization for the area. Equipment operability or proper compensatory activities were verified on a periodic basis.

(6) Surveillance (61726)(61700) - Surveillance tests were observed to verify that approved procedures were being used, qualified personnel were conducting the tests, tests were adequate to verify equipment operability, calibrated equipment was utilized, and TS requirements were followed. The inspectors observed portions of the following surveillances and/or reviewed completed data against acceptance criteria:

Surveillance No.	Title
14806-1, Rev. 5	Containment Spray Pump and Check Valves Inservice Test
14807-2(1), Rev. 2(6)	MDAFW Pump Inservice Test
14804-2, Rev. 2	Safety Injection Pump Inservice Test
14825-1, Rev. 11	Quarterly Safety Injection System Valve Inservice Test
14420-1, Rev. 8	SSPS Train A(B) Operability Test
14980-2, Rev. 2	Diesel Generator Operability Test

PORV surveillance testing was examined in support of a regional inspection conducted during this reporting period. A potential violation of TS was identified in that a portion of the electrical circuitry for the automatic function of the PORV was not tested as required by TS 4.4.4. The circuitry for manual operations had been properly tested. The inspector interpreted the surveillance requirement to include the "automatic" function. However, the licensee's interpretation was that the manual mode was adequate for operability because of the following reasons:

1. action statement "a" of this LCO (if there was excessive seat leakage) would allow indefinite plant operation with both block valves closed (i.e., without the PORV automatic function);

2. the language of the Vogtle FSAR, Chapter 15, does not require (nor use) the PORV automatic actuation; and

3. under severe accident conditions, the PORVs will be used in the manual mode.

The licensee agreed that the surveillance of the automatic circuitry was inadequate. They did not, however, intend to immediately rectify this shortcoming.

The reasons the licensee wanted to delay this test were to avoid an additional PORV block valve stroking and because the part of the circuit which had not been tested was a circuit of very high reliability.

The inspector's review of the accident analysis noted that the licensee took credit for the automatic function of the PORVs for the following reasons:

1. TS 3/4.4.4 bases states that the PORVs and steam bubble function to relieve RCS pressure during all design transients, up to and including the design step load decrease with steam dump.

2. Two cases, for both the minimum and maximum reactivity feedback, are analyzed in the FSAR, Chapter 15, paragraph 15.2.3.2.1. One case takes full credit for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure with safety valves available.

3. The FSAR, Chapter 15, Table 15.2.3-1, indicates that with and without offsite power available for a feedwater system pipe break, pressurizer PORVs are expected to actuate in 19.5 seconds.

4. One of the major assumptions in the FSAR, Chapter 15, paragraph 15.2.8.2.1, used for a double ended rupture of the largest feedwater pipe at full power is that credit is taken for the pressurizer PORVs and the safety relief valves. No credit is taken for pressurizer spray.

5. FSAR-Q, Question 420.19, states, "Using detailed schematics, describe the design of pressurizer power-operated relief valve control and the block valve control, and verify that no single failure will preclude the automatic actuation logic for all modes of operation."

On November 17, 1989, at 10:00 a.m., a conference call took place between the licensee and NRC management regarding this issue. The purpose was to review the NRC staff position, which concluded that the automatic function was required and should be immediately tested. In response to this, the licensee tested both remaining PORVs on November 17, 1989. Details of this issue are available in NRC Inspection Report Nos. 50-424/89-31 and 50-425/89-36.

(7) Maintenance Activities (62703) - The inspector observed maintenance activities to verify that correct equipment clearances were in effect; work requests and fire prevention work permits, as required, were issued and being followed; quality control personnel were available for inspection activities as required; retesting and return of systems to service was prompt and correct; and TS requirements were being followed. The Maintenance Work Order backlog was reviewed. Maintenance was observed and/or work packages were reviewed for the following maintenance activities:

MWO No.	Work Description
28905765	Repair Packing Leak On Main Steam Valve 2LV-6191
28905799	Clean And Repair EHC Fuller's Earth Filter Due To Leaking Filter Canister Caps.
18905983/ 28906212	Increase The "Low Hydraulic Pressure" Pressure Switch Set Point To The SG ARV Actuators Due To Insufficient Stored Energy To Operate The Valves During Worst Case Conditions.
18903554	Repair Auxiliary Steam Leak On Auxiliary Steam Supply Valve To The TDAFW Pump (1-1301-U4-264)

One violation was identified in paragraph 2.b(1) above.

3. Review of Licensee Reports (90712)(90713)(92700)

a. In-Office Review of Periodic and Special Reports

This inspection consisted of reviewing the below listed reports to determine whether the information reported by the licensee was technically adequate and consistent with the inspector knowledge of the material contained within the report. Selected material within the report was questioned randomly to verify accuracy and to provide a reasonable assurance that other NRC personnel have an appropriate document for their activities.

Monthly Operating Report - The report dated November 10, 1989, was reviewed. The inspector had no comments.

(Closed) Special Report dated October 16, 1989, "Loose Part Detection System."

On October 16, 1989, plant personnel discovered that the channel calibration for the Loose Part Detection System was not performed as specified in the FSAR, Section 16.3, Requirement 3. A Maintenance Work Order generated under the Preventive Maintenance Program was initiated on July 27, 1988, to perform the 18-month channel calibration during the first refueling outage for Unit 1. The MWO was erroneously voided due to the belief it was not required to be included in the scope of the refueling outage. The FSAR commitment to demonstrate the operability of the Loose Part Detection System was not properly identified in the PM program which resulted in the MWO not being worked during the outage. For corrective action, the channel calibration for the Loose Part Detection System has been entered into the Technical Specification Surveillance Tracking Program. Since this surveillance can only be performed in Mode 5, the operability of the Loose Part Detection System will be demonstrated during the next refueling outage for Unit 1 (1R2). Additionally, it should be noted a channel operational test has been performed at least once per 31 days in accordance with FSAR Section 16.3. The inspector has no further questions.

b. Deficiency Cards and Licensee Event Reports

Deficiency Cards and Licensee Event Reports were reviewed for potential generic impact, to detect trends, and to determine whether corrective actions appeared appropriate. Events which were reported pursuant to 10 CFR 50.72, were reviewed following occurrence to determine if the technical specifications and other regulatory requirements were satisfied. Each LER was reviewed for enforcement action in accordance with 10 CFR Part 2, Appendix C, and where the violation was not cited, the criteria specified in Section V.G of the Enforcement Policy were satisfied. Review of DCs was performed to maintain a realtime status of deficiencies, determine regulatory compliance, follow the licensee corrective actions, and assist as a basis for closure of the LER when reviewed. Due to the numerous DCs processed, only those DCs which result in enforcement action or further inspector followup with the licensee at the end of the inspection are listed below. The DCs and LERs denoted with an asterisk indicate that reactive inspection occurred following the event and prior to receipt of the written report.

(1) The following Deficiency Cards were reviewed:

(a) DC 1-89-1544, "TS Violation Regarding Excessive Tendon Grease Void".

During performance of the third year tendon surveillance a tendon was found with a grease void exceeding 5 percent. This is a violation of TS 3.6.1.6.b and 4.6.1.6.1.d.2. This item will be further followed up when submitted as a special report.

(b) DC 1-89-1562, "Degradation Of Containment Structural Integrity."

During performance of third year tendon surveillance, grease was emitted into level C of the auxiliary building while regreasing horizontal tendon #6. This resulted in a violation of TS section 4.6.1.6.1.d.4. This item will be further followed up when submitted as a special report.

(c) DC 1-89-1579, "Rosemont Transmitters Supplied To VEGP Were Of Manufacturing Groups That Are Susceptible To Failure Due To Oil Loss."

It was identified that a group of Rosemont Transmitters supplied to VEGP contain a defect which, given a failure due to oil loss, would possibly lead to a substantial safety hazard. This item is under review for applicability and reportability as a Part 21 report.

(d) *DC 2-89-1490, "Manual Reactor Trip Due To SG Levels Approaching The Lo-Lo Level Setpoint."

On November 5, 1989, an operator manually tripped the reactor due to the steam generators approaching the LoLo level setpoint. Operators were placing a heater drain tank level control valve in service when the valve opened, causing a low suction pressure to the steam generator feed pumps. The standby condensate pump failed to start and the feed pump tripped on low suction pressure. This resulted in a partial loss of feedwater to the steam generators. All control rods fully inserted, main feedwater isolated and AFW actuated on the trip. All systems functioned as required. This item will be further followed up when submitted as a LER.

(e) *DC 2-89-1508, "Containment Ventilation Isolation Due To Containment Radiation Monitor 2-RE-003."

During the performance of a Surveillance on Containment Radiation Monitor 2RE-003, a Containment Ventilation Isolation occurred. Operators verified proper Isolation. Radiation levels were checked to be at normal values. Computer printouts indicated that the monitor returned to "Normal" during the surveillance. All radiation levels were checked to be normal and the system was returned to normal operating status. This event will be further followed up when submitted as a LER.

(2) The following LERs were reviewed and closed.

(a) 50-424/88-37, Rev. 0, "O-Ring Found Missing In Post Accident Monitoring RTD's Junction Boxes."

On November 16, 1988, while performing Maintenance Work Order 18808056, O-rings were discovered missing from four CONAX T-8 Head junction boxes. Three of the boxes service resistance temperature detectors that provide reactor coolant T-hot wide range temperature indication for post accident monitoring. The detectors were in an untested configuration. Technical Specification 3.3.3.6, "Accident Monitoring Instrumentation," requires that these detectors be operable during plant operation. On November 4, 1988, while reviewing environmental qualification documentation, it was noted that installation of O-rings was required in the tested configuration to seal the CONAX T-8 Head junction boxes. A check of material inventory revealed that no O-rings had been ordered as replacement spares. An MWO was written to inspect the subject boxes. During the inspection, four O-rings were discovered missing. This event occurred because the O-rings were not installed during initial installation. All the CONAX T-8 Head junction boxes were inspected under MWO 18808056. The missing O-rings were replaced and the boxes sealed. Environmental qualification documentation has been updated to clarify the requirements for O-rings and maintenance procedures have been revised to address their replacement. Inclusive in this review was a Westinghouse review of the issue. The inspector's questions regarding the corrective actions' completeness were resolved. Enforcement action was discussed in NRC Inspection Report No. 50-424/89-25.

(b) 50-424/88-47, Rev. 0, "Error In Procedure Leads To Technical Specification 3.0.3. Entry."

On June 14, 1988, while the unit was at 100 percent power, handswitches for manual actuation of Containment Isolation - Phase A and Containment Ventilation Isolation were tested. Each handswitch was taken out of service, tested, and returned to service. On October 13, 1989, while preparing to perform the test, a system engineer identified an error in the procedure which resulted in simultaneously disabling both handswitches. This condition is in conflict with the requirements of Technical Specification Table 3.3-2 which requires both handswitches to be operable in Modes 1, 2, 3, and 4. Although, during the previous test, LCO entries had been made for each handswitch being out of service, it was not recognized that both handswitches were out of service at the same time. This condition resulted in entry into TS 3.0.3. Automatic actuation capability was not affected by the testing and was available, if required. Procedures 54708-1 and 54708-2 have been revised to ensure that the manual actuation handswitches will not be simultaneously disabled.

(c) *50-425/89-20, Rev. 0, "Loss Of Power To NI Channel Causes Reactor Trip During Surveillance Test."

On May 12, 1989, while personnel were performing surveillance of nuclear instrument channel N44, a two out of four Hi Flux rate trip coincidence signal was received, causing an automatic reactor trip. Power range channel N43 experienced a momentary loss of power, which tripped the Rate Trip bistable on N43. The control room operator acknowledged the alarm for the tripped bistable, but failed to notice that the wrong bistable had tripped for the work being performed. A step of the surveillance procedure, which was being performed for N44, requires the fuses to be pulled. This action tripped the Rate Trip bistable for N44. The N43 and N44 tripped bistables satisfied the two out of four logic for a power range reactor trip. All automatic systems functioned as designed. The causes of this event were the loss of power to channel N43 and the failure of control room operators to notice that the wrong bistable had tripped. Extensive troubleshooting of N43 was performed. The cause for the power loss could not be determined. The operations requalification training program includes increased emphasis on recognizing the cause of the alarm being acknowledged. Nuclear instrument calibration procedures were revised to require reactor operator signoff (in addition to instrument technician signoff presently required) prior to manually tripping bistables or removing instrument power. The inspector has no further questions.

(d) *50-425/89-27, Rev. 0, "Reactor Trip On High Flux Rate Due To Rod Drop."

On October 12, 1989, an automatic reactor trip occurred with the reactor in stable operation at 58 percent of rated thermal power. Following a review of computer printouts of data associated with the trip, the first out annunciator was identified as a high flux rate trip annunciator. Operability testing of the control rods then indicated that a problem existed with rod K-2 in control bank B. Investigation of the control rod circuitry identified a failed diode which had apparently resulted in a loss of current to the stationary gripper coil. This allowed the rod to drop into the core and initiate a negative flux rate trip. Corrective action included replacing the diode for rod K-2. The inspector reviewed the post trip data and monitored the repair and restart activities. The inspector has no further questions.

(e) 50-425/89-28, Rev. 0, "Arcing Power Cable Leads to Containment Ventilation Isolation."

On October 16, 1989, a technician was preparing to replace a faulty circuit board in a containment vent effluent radiation monitor panel. While performing this work, he contacted a power cable and arcing occurred at the thermal connection. The arcing resulted in power fluctuations at the Input/Output circuit board which subsequently failed. This led to a Containment Ventilation Isolation actuation. The cause of this event was an inadequate design. The screw on the radiation monitor terminal block was too short to adequately engage the threaded opening and provide a tight, permanent connection with the attached power cable. When the technician's hand contacted the cable, the connection was loosened and arcing occurred. This screw and a similar screw in Unit 1 have been replaced. The inspector has no further questions.

No violations or deviations were identified.

4. Actions on Previous Inspection Findings - (92701)(92702)

a. (Closed) IFI 50-424/88-21-02 and 50-425/88-31-02, "Update Posted TSC Facility Layout With Existing Operations Area Confirmation And Designated Work Stations."

The inspector verified that the postings had been updated. The setup of the Technical Support Center was verified during the July exercise to be in accordance with the new posting. This verification was made as a followup to that documented in NRC Inspection Report Nos. 50-424/89-01 and 50-425/89-01.

b. (Closed) VIO 50-424/89-14-01 and 50-425/89-15-01, "Six Examples Of Failure To Establish Or Implement Procedures."

The inspector reviewed the licensee's response dated July 12, 1989, and correction letter dated September 18, 1989, against the closure documentation assembled by the licensee. The inspector determined that the actions are adequately complete.

c. (Closed) VIO 50-424/89-19-01, "Failure To Implement Operations Procedure 10001-C As Required By TS 6.7.1.a To Verify Proper Operation Of Control Room Chart Recorders."

The licensee committed to full compliance on August 31, 1989, in the licensee's response dated August 30, 1989. All corrective actions were found to be satisfactory. This verification was made as a followup to that documented in NRC Inspection Report Nos. 50-424/89-01 and 50-425/89-01.

5. Management Meetings - (30702)

This activity involves inspector participation and preparation in support of the following meetings which presented site readiness.

On October 26, 1989, the inspectors attended a licensing issues status meeting conducted on site.

6. Exit Interviews - (30703)

The inspection scope and findings were summarized on December 1, 1989, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection results. No dissenting comments were received from the licensee. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspector during this inspection. Region based NRC exit interviews were attended during the inspection period by a resident inspector. This inspection closed two violations (paragraphs 4.b and 4.c), one inspector followup item (paragraph 4.a), and five Licensee Event Reports (paragraph 3.b(2)). One new item was identified during this inspection:

Violation 50-424/89-33-01 and 50-425/89-38-01, "Failure To Implement Procedures 00101-C And 50009-C Concerning Legibility On Control Room Drawings" - Paragraph 2.b(1).

7. Acronyms And Initialisms

ABN	As Built Notice
AFW	Auxiliary Feedwater System
ARV	Atmospheric Relief Valve
CFR	Code of Federal Regulations
CONAX	(trade name)
CVI	Containment Ventilation Isolation
DC	Deficiency Cards
EHC	Electrohydraulic Control
ESF	Engineered Safety Features

FSAR	Final Safety Analysis Report
IFI	Inspector Followup Item
LCO	Limiting Conditions for Operations
LER	Licensee Event Reports
MDAFW	Motor Driven AFW Pump
MFP	Main Feed Pump
mrem	Millirem
MWO	Maintenance Work Order
NI	Nuclear Instrument
NPF	Nuclear Power Facility
NRC	Nuclear Regulatory Commission
PM	Planned Maintenance
PORV	Power Operated Relief Valve
RCS	Reactor Coolant System
Rev	Revision
RTD	Resistance Temperature Detector
RWP	Radiological Work Permit
SG	Steam Generator
SSPS	Solid State Protection System
TDAFW	Turbine Driven AFW Pump
TS	Technical Specification
TSC	Technical Support Center
VEGP	Vogtle Electric Generating Plant
VIO	Violation