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

Report No.: 50-395/95-15

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: August 1-31, 1995

Inspectors: via phone 9/15/95
T. R. Farnholtz, Resident Inspector Date Signed

Other Inspectors: D. C. Payne, August 14-18, 1995
T. M. ~~Ross, August~~ 28 through September 1, 1995

Approved by: 9/15/95
H. O. Christensen, Chief Date Signed
Reactor Projects Branch 1B
Division of Reactor Projects

SUMMARY

Scope:

This routine inspection was conducted by the resident inspectors onsite in the areas of operational safety verification; maintenance observations; surveillance observations; onsite engineering; plant support activities; and followup on previous operation findings. Selected tours were conducted on backshift or weekends. These tours were conducted on August 21, 25, 28, 30, and 31, 1995.

Results:

Operations

A weakness in the operator's decision making process was identified concerning the performance of the monthly control valve testing. The operators continued control valve testing without a full understanding of the plant's response. A high number of main control board discrepancies were noted. A large number of main control board indicators were reading above or below the "green band". These issues indicate a lack of a questioning attitude by the operations staff.

Maintenance and Surveillance

Inadequate pre-job planning resulted in the inability of maintenance technicians to perform scheduled maintenance on a charging/safety injection

pump. A faulty test switch was identified in the solid state protection system. A non-cited violation was identified for the failure to perform a technical specification required surveillance test on selected component cooling water system valves (paragraph 4.c). A non-cited violation was also identified for the failure to document and control the repositioning of two jumper wires on a reactor vessel water level indication system circuit card (paragraph 4.d).

Engineering and Technical Support

An initiative by the systems and component engineering group to perform a detailed walkdown of plant systems was considered to be a good effort to improve the material condition of the plant. A non-cited violation was identified concerning a wiring error in a reactor building radiation monitor (paragraph 5.b).

Plant Support

A violation was identified concerning a smoke detector which was inadvertently rendered inoperable with no compensatory actions in place (paragraph 6.b). A meeting of the Plant Safety Review Committee was beneficial.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*F. Bacon, Manager, Chemistry Services
*L. Blue, Manager, Health Physics
*M. Browne, Manager, Design Engineering
*S. Byrne, General Manager, Nuclear Plant Operations
*M. Fowlkes, Manager, Nuclear Licensing & Operating Experience
*S. Furstenberg, Manager, Maintenance Services
*S. Hunt, Manager, Quality Systems
*D. Lavigne, General Manager, Nuclear Safety
*J. Nesbitt, Manager, Technical Services
K. Nettles, General Manager, Station Support
*H. O'Quinn, Manager, Nuclear Protection Services
M. Quinton, General Manager, Engineering Services
G. Taylor, Vice President, Nuclear Operations
*R. Waselus, Manager, Systems and Component Engineering
R. White, Nuclear Coordinator, SC Public Service Authority
*B. Williams, Manager, Operations
G. Williams, Associate Manager, Operations

Other licensee employees contacted included engineers, technicians, operators, mechanics, security force members, and office personnel.

NRC Personnel

*T. Farnholtz, Resident Inspector

*Attended exit interview conducted September 8, 1995

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. PLANT STATUS AND ACTIVITIES

a. The plant operated at or near 100 percent power during the entire inspection period.

b. Other NRC inspections or meetings:

- Mr. Charlie Payne, a regional inspector, was onsite August 14-18, 1995, to provide site coverage during the Resident Inspectors' meeting.

- Mr. Thierry Ross, Senior Resident Inspector from Farley, was onsite August 28 through September 1, 1995, to tour the plant and meet with the resident inspector.

- Mr. Chris Christensen, Branch Chief, DRP, was onsite September 1, 1995, to review resident inspector's activities, tour the plant, and meet with licensee management.

3. OPERATIONS

a. Plant Operations (71707)

The inspectors conducted frequent inspections in the following areas: control room staffing, access, and operator responsiveness; operator adherence to approved procedures, TS requirements, and limiting conditions for operations; status of control room annunciators and instrumentation; and review of control room operator logs, operating orders, plant deviation reports, tagout logs, equipment out-of-service log, and tags on components to verify compliance with approved procedures. Routinely, the inspectors attended the operations shift turnover meetings.

The inspectors conducted weekly inspections for the operability verification of selected ESF systems by valve alignment, breaker positions, condition of equipment or component(s), and operability of instrumentation and support items essential to system actuation or performance. The containment spray and reactor makeup water systems were included in these inspections.

Plant tours included observation of general plant/equipment conditions, control of activities in progress, plant housekeeping conditions/cleanliness, and missile hazards. Reactor coolant system leak rates were reviewed to ensure that detected or suspected leakage from the system was recorded, investigated, and evaluated; and that appropriate actions were taken if required. Selected tours were conducted on backshifts or weekends.

b. Control Valve Testing (71707)

On August 26, 1995, the licensee conducted monthly turbine control valve testing. The procedure for this test directs the operators to position the load limiter on the main control board to 10. This allows the number 4 control valve, which is normally throttled, to fully open as control valves 1, 2, or 3 are closed during the test. The purpose for this is to maintain steam flow to the high pressure turbine. After positioning the load limiter, the operators proceeded to test the first control valve. The plant responded in an unexpected manner by automatically stepping in the control rods further than normal and actuating the steam dump system. The steam dump system normally does not actuate during this test. Following the completion of the first valve, the plant returned to normal. After some discussion, the operators proceeded to test the second valve and the plant reacted the same way. The third valve was also tested with the same results. After the test, while returning the plant to normal, the operators noted that the load limiter had failed to go to position 10 due to a malfunction. Instead, it had been at approximately position 9 during the test which would explain why the plant did not respond as expected due to the number 4 control valve not being able to fully open.

The inspector reviewed this event and concluded that, although the safety significance was minimal, the decision to proceed with the second and third control valve test without having a full understanding of why the plant reacted as it did was uncharacteristic. Typically, when an unexpected result occurs during a test, the licensee stops the test and searches for the reason. Only after gaining a full understanding does the testing resume following any necessary repairs or adjustments. This event reflects a lack of a questioning attitude and a weakness in the operators decision making process.

c. Main Control Board Discrepancies (71707)

During routine tours of the control room the inspector noticed a large number of MWRs on the main control board (MCB). The inspector discussed this with the licensed operators onshift and reviewed the current week's log of "MCB Discrepancies" dated August 25, 1995. From this review, the inspector determined that there were 95 outstanding MCB discrepancies (which included ancillary panels in the control room), approximately half of which were identified since June 1, 1995. Only about 20 of the discrepancies appeared to be older than 18 months, with just a few being five years or more. The current backlog of MCB discrepancies has remained relatively constant over the past several months. A large number of MCB discrepancies can present work-around difficulties for the operators. The inspector expressed his concerns regarding the excessive number of MCB discrepancies to site management. The licensee indicated they would review this issue.

d. Off Normal Indicators (71707)

The inspector noticed that many of the principal MCB indicators were reading significantly outside their designated green bands (e.g., pressurizer surge line temperature, RCS flow, MS and MFW flow, MFW temperature, RCS T-hot, letdown and regenerative heat exchanger temperatures, refuel cavity temperature). However, none of these measured parameters were in alarm nor were the indicators reading in excess of the identified red band. Subsequent discussions with shift operators determined that most of the indicators had been reading this way for years. Furthermore, no apparent actions were in progress to change them. Indicators that consistently read outside the green band could be indicative of instrumentation problems, errors made in the original establishment of the normally expected ranges, or a fundamental change has occurred in the measured process; all of which warrant a specific evaluation. At a minimum, consistent operation of plant parameters outside the normal green band fosters a level of complacency among operators to tolerate off normal conditions. These concerns were discussed with plant management for resolution. The licensee indicated they would review this issue.

e. Control Room Evacuation Panel Review (71707)

The inspector walked down the control room evacuation panels (CREP) located in the Intermediate Building 436 foot elevation. All indicators appeared to be functioning properly. Those expected to be onscale, were exhibiting indications consistent with their MCB counterparts. All transfer switches were properly selected to the "remote" position. Principal panels were in good physical condition; the surrounding area was bright and clear of all non-essential material. The inspector also examined the contents of a normally locked tool locker located in the immediate area. This locker contained essential tools, equipment and procedures needed to shutdown and operate the plant from outside the control room during an emergency. All the tools and equipment were in good working order; required procedures were in place and up-to-date. Plant operators expressed overall familiarity with the CREP, but indicated that there has been little if any recent training on the use of this facility.

4. MAINTENANCE

a. Maintenance Observation (62703)

Station maintenance activities for the safety-related systems and components listed below were observed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, industry codes or standards, and in conformance with TS.

The following items were considered during this review: limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; activities were accomplished by qualified personnel; parts and materials used were properly certified; and radiological and fire prevention controls were implemented. Work requests were also reviewed to determine the status of outstanding jobs and to ensure that priority was assigned to safety-related equipment maintenance that may affect system performance.

The following maintenance activities were observed:

(1) Internally inspect, clean, and lube fuel handling building charcoal filter plenum "A" (PMTS P0190748). No discrepancies were noted.

(2) Component cooling water/service water heat exchanger performance test (PMTS P0189278). The inspector observed maintenance technicians installing test equipment and gathering required data. The collected data was used by the heat

exchanger component engineer to determine the heat exchanger fouling factor. The required parameters included inlet and outlet temperature for both the service water and component cooling water sides, and the flow rates for each side. The work performed to gather this data was done in accordance with the approved procedure.

(3) Charging/safety injection pump "B" suction pressure indicator calibration (PMTS P0186052). The observed work was performed satisfactorily, however, the inspector noted that the lighting in the area of the indicator was poor due to the lack of a lighting fixture.

(4) Clean boron buildup and perform a torque check on the "B" charging/safety injection pump (MWR 95T3370). Boron crystals in the area of the pump seals were removed using distilled water and brushes using proper radiological controls. A small leak was observed near the bottom of the pump at the junction between the casing and the head. To stop this leak, the licensee planned to perform a torque check of the nuts, which were used to secure the head to the casing. The technical manual specifies a gap between these two components of .040-.045 inches. The as-found gap was determined to be .043-.044 inches, which provided sufficient clearance for some additional tightening of the nuts. The licensee did not torque any nuts because of some interference in the area of the nuts. The inspector concluded that better job planning would have identified the interference problems. This maintenance work request was kept open and the work was rescheduled for a later date. The leak was considered minor and did not affect operability of the pump.

All observed maintenance activities were conducted using good work practices and approved procedures. Poor lighting was identified in the area of the "B" charging/safety injection pump room. Poor job planning resulted in the inability of the maintenance technicians to perform scheduled maintenance.

b. Surveillance Observation (61726)

The inspectors observed surveillance activities of safety-related systems and components listed below to ascertain that these activities were conducted in accordance with license requirements. The inspectors verified that required administrative approvals were obtained prior to initiating the test, testing was accomplished by qualified personnel in accordance with an approved test procedure, test instrumentation was calibrated, and limiting conditions for operation were met. Upon completion of the test, the inspectors verified that test results conformed with TS and procedure requirements, any deficiencies identified during the testing were properly reviewed and resolved, and the systems were properly returned to service.

Specifically, the inspectors witnessed/reviewed portions of the following test activities:

(1) Fuel handling building charcoal filter plenum "A" fire damper inspection (STP 428.064). No discrepancies were noted.

(2) Solid state protection system (SSPS) train "A" actuation test (STP 345.037). During the test, position six on the logic "D" switch tested as bad. The technicians attempted to test the position a second time with the same results. The licensee declared the "A" train SSPS inoperable and entered TS 3.3.1 and 3.3.2 six hour action statement. The I&C technicians determined that the problem was in a circuit card. After determining which card was most likely at fault, a new card was obtained from the warehouse, verified as the correct replacement, and was installed. The retest of position six on the logic "D" switch resulted in the same bad indication as before. Further investigation revealed that the logic "D" test switch itself was at fault. Dirty switch contacts appeared to be the problem. The original circuit card was returned to service. The logic "D" switch is used only during logic testing and is not in the circuit during normal operation of the SSPS. The remainder of the test was completed with no further incident. At the completion of the test, the "A" train SSPS was declared operable. The logic "D" switch was not replaced. The inspector considered the initial suspicion of a faulty circuit card, although incorrect, was reasonable. It was not practical to attempt to replace the test switch while the SSPS was energized. Since the switch plays no role during normal operation, the inspector concluded that not replacing the switch was acceptable.

(3) Chilled water pump "B" surveillance test (STP 229.001). The inspector observed Test Unit personnel, with an operator, perform a quarterly inservice test of the train B chilled water pump in accordance with STP 229.001, "HVAC Chilled Water Pump Test." The test was conducted in a deliberate step-by-step manner per the STP. All test and measurement equipment was within calibration, or within the established grace period. Discharge and inlet pressure readings of the "B" pump were extremely close to the expected reference values and well within the required range. Vibration readings were also within acceptable limits. Pump differential pressure and measured flow conformed very well with the reference pump curve. Test unit personnel maintained close radio communications with the control room throughout the test. Pump performance data was reviewed with the control room supervisor immediately after the test.

All observed surveillance activities were performed in an acceptable manner. The licensee's actions regarding a defective test switch in the solid state protection system was appropriate.

c. Failure to Perform TS Required Surveillance (61726)

During refueling outage eight, the licensee completed a modification to change the cooling medium for the charging/safety injection pump oil coolers and the CCW pump motors from chilled water to CCW. As a part of this modification, the newly installed CCW valves should have been added to the surveillance intended to meet the requirements of TS 4.7.3, which states that each valve will be verified to be in its correct position at least every 31 days. The new CCW valves associated with the "C" charging pump were not included in this surveillance test. A valve lineup including these valves was performed and documented on December 20, 1994, and the "C" charging pump was operated on the "A" train during the period from January 23 to February 13, 1995, with no further valve verification. This gave a period of approximately 55 days during which time no verification of valve position was performed. During operation of the pump, oil cooler outlet temperature was logged every 12 hours. If the CCW valves had not been in the required positions, the oil outlet temperatures would likely have been higher than normal. No such elevated temperatures were noted. When another valve lineup was performed on February 13, 1995, no out of position CCW valves were noted. The failure to perform the TS required surveillance on the CCW valves associated with the "C" charging/safety injection pump was identified as a violation. However, based on the low safety significance, it is being treated as a non-cited violation, consistent with Section IV of the NRC Enforcement Policy (NCV 395/95-15-01).

d. Calibration of Reactor Vessel Water Level Indication System (RVLIS) (61726)

During a November 1994, calibration of the RVLIS circuitry, the I&C technicians repositioned two jumper wires on a temperature compensating card in accordance with the applicable technical manual. This caused the output of this card to remain at zero regardless of input. No record was made of this repositioning which resulted in the failure to return the two jumpers to their correct positions following calibration activities. This condition was identified on August 21, 1995 during card replacement activities. During this approximately 10 month period, no temperature compensation was available for this portion of the RVLIS system. The licensee determined, and the inspector concurred, that the RVLIS system was operable during this time because the error was in a conservative direction (water level indication would have been lower than actual water level). The analysis performed for a worst case loss of coolant accident indicated that a temperature of 268°F would be experienced in the area of the reactor vessel. This would result in approximately a 3.5 percent error in the narrow range instrument and a 1 percent error in the wide range instrument without temperature compensation.

Station Administrative Procedure (SAP-300), Conduct of Maintenance, gives instructions to maintenance technicians on how and when to document lifted leads and jumpers. The failure to document and control the repositioning of the two jumpers on the RVLIS temperature compensation card was identified as a violation. Because of the minimal safety significance, the violation will be non-cited, consistent with Section IV of the NRC Enforcement Policy (NCV 395/95-15-02).

5. ENGINEERING

a. System and Component Engineering Walkdown (37551)

During the inspection period, the inspector observed system and component engineering personnel performing detailed walkdowns and inspections of the plant systems. The purpose of these walkdowns was to identify any and all problems which existed with any component in the systems inspected. The systems and component engineering personnel were divided into four groups and each group assigned an area of the plant to inspect in detail. The assignments were made such that the engineers were inspecting systems other than those normally assigned to them. This provided a "fresh look" at the systems inspected. The result of the walkdowns was an extensive list of discrepancies including location and description. The licensee stated that it was their intention to correct all of these discrepancies and track each one to completion. In addition, more such walkdowns and inspections were planned for the future. The inspector considered this to be an excellent initiative which has resulted in the identification of many "small" items which, if corrected, would improve the material condition of the plant.

b. Radiation Monitoring System Walkdown (37551)

On August 24, 1995, the licensee documented a wiring error noted in the cabinet containing the circuits for RMG-7. This radiation monitor is the reactor building high range instrument which provides indication of the magnitude of a loss of coolant accident. The system engineer, while performing a verification of the as-built drawing of this circuit, noted that a fuse in the power supply was by-passed and not able to perform its function. The purpose of this fuse was to isolate the 1E and non-1E portions of the radiation monitor if a fault were to develop and was one of a series of fuses in the circuit. The radiation monitor circuitry continued to be protected by additional fuses in series with the jumpered fuse. In this case, the as-built drawing did not match the as-built configuration. A modification performed to this system in the mid 1980's should have removed this jumper but failed to do so. The failure to control the design of a safety-related radiation monitor was identified as a violation. However, because of the minor safety significance, it is being treated as a non-cited violation, consistent with Section IV of the NRC Enforcement Policy (NCV 395/95-15-03).

The inspector considered the identification of this wiring error to be a positive outcome of an engineering initiative to verify plant drawings. The complexity of the wiring cabinets makes the task of tracing individual wires difficult and tedious.

c. Followup - Engineering (92903)

(Closed) Unresolved Item 95-02-02, Substitution of Operator Action for a Loss of ESF Function.

On February 7, 1995, the licensee determined that the plant was subject to a previously unreviewed accident scenario that could result in the loss of automatic ESF actuations. The scenario involved a steam line break in the turbine building which could damage SSPS circuits resulting in a loss of electrical power for both trains of SSPS. The loss of SSPS power would cause the reactor trip breakers to open, but would prevent SSPS from initiating any ESF actuation signals. In response to this potential accident scenario, the licensee did not declare SSPS inoperable, but used the guidance in Generic Letter (GL) 91-18, to determine that SSPS remained operable by use of manual action in place of automatic action.

The NRC has completed its review of this issue and concluded that GL 91-18 was not appropriately used in this case. For GL 91-18 to have been correctly applied to this scenario, the licensee should have demonstrated that the design basis analysis remained bounding when manual operator action was used in place of automatic initiation. In its analysis, the licensee concluded that the design basis was met because the departure from nucleate boiling ratio (DNBR) was acceptable even with no automatic ESF actuations. The licensee's argument was based on manual ESF actuations that would be performed 10 minutes after the event's initiation. However, UFSAR Section 15.4.2, "Major Secondary Steam Pipe Rupture", specifies that automatic ESF actuation will occur after a major steam line rupture to successfully mitigate this type of event. Therefore, the UFSAR analysis is based on the mitigating ESF actuations occurring immediately. For example, the UFSAR and TS Table 3.3-5, "Engineered Safety Features Response Times", states that the main steam lines will be isolated within 10 seconds of a large break in the steam line. Since the UFSAR specifies that the action will be done within 10 seconds, any time greater than 10 seconds is no longer bounded by the UFSAR analysis and thus outside the design basis. Therefore, the licensee was incorrect when they concluded they were still within their design basis by using manual actions in place of automatic actions.

Once the licensee found that the Section 15.4.2 analysis was no longer a bounding analysis, the licensee could have performed a 10 CFR 50.59 analysis to determine if the UFSAR could be changed. Assuming there was no unreviewed safety question, the licensee's 10 CFR 50.59 analysis could be used to temporarily change the UFSAR to reflect the steam line rupture without automatic ESF actuation as the new design basis Chapter 15 analysis. In addition to the temporary design basis change, in order to use the GL 91-18 guidance the licensee should have shown that DNBR was acceptable with the (1) steam line rupture initiating event, (2) consequential total loss of automatic ESF actuation, (3) a worst-case single failure (e.g., loss of offsite power, stuck control rod, failed MSIV), and (4) any postulated consequential failures.

Based on the above discussion, the licensee's analysis was not thorough which resulted in an incorrect conclusions. However, this accident scenario was considered to be a low-probability event and thus the safety significance was low. Additionally, the licensee took prompt corrective action and modified the SSPS circuity to correct the design deficiency. This Unresolved Item is closed.

6. PLANT SUPPORT

a. Plant Support Activities (71750)

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, fire brigade readiness, and fire watch patrols. Effective implementation of the physical security program continued to be demonstrated during inspector 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.

b. Inoperable Smoke Detector (71750)

On August 14, 1995, while testing fire service smoke detector IXA04950L, the technician at the Simplex graphic monitor in the control room inadvertently by-passed this instrument by touching the by-pass block on the screen. This had the effect of rendering the smoke detector inoperable. The smoke detector is located in room 12-18 of the auxiliary building. After completing the task, the technician did not notice a priority 1 message on the screen which would indicate that the system was not in a normal configuration. This condition continued for approximately 18 hours until discovery when the system was restored to normal and the smoke detector returned to operable status.

The Limiting Condition for Operation in Station Administrative Procedure (SAP-131A), Attachment I, Fire Detection Instrumentation, states that, "As a minimum, the fire detection instrumentation for

each fire detection zone shown in Table 1 (One) shall be operable". Table One in SAP-131A indicates that the total number of instruments (smoke detectors) in room 12-18 of the auxiliary building is one, and that the minimum number of operable instruments is one. This requirement was not met when smoke detector IXA04950L was by-passed and rendered inoperable. In addition, no hourly fire watch patrol was established to compensate for the inoperable instrument. The failure to have the minimum number of fire detection instruments operable or compensated for was identified as violation, 395/95-15-04, for failure to follow work instructions.

c. Meeting of the Plant Safety Review Committee (PSRC) (40500)

The inspector attended a meeting of the PSRC on August 23, 1995. Items discussed during the meeting included several off-normal occurrence closures, a final safety analysis report (FSAR) revision, and revisions to SAP-500 (Health Physics Manual) and SAP-131 (Fire Protection Program). The discussions were open and all members were given the opportunity to voice any concerns or comments they had. Several items were sent back to the originating organizations for additional work before being approved by the PSRC. The inspector concluded that the meeting was beneficial.

7. EXIT INTERVIEW

The inspectors met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on September 8, 1995. During this meeting, the inspectors summarized the scope and findings of the inspection as they are detailed in this report. The licensee representatives acknowledged the inspector's comments and did not identify as proprietary any of the materials provided to or reviewed by the inspectors' during this inspection.

In response to the closure of Unresolved Item 92-02-02 (paragraph 5.c), the licensee disagreed with the NRC assessment outlined in this report. The following statement was made by the licensee at the exit interview: "The postulated scenario for the V. C. Summer Nuclear Station was determined to be bounded by the Main Steam Line Break (MSLB) analysis performed as part of the Design Basis Analysis of Chapter 15 of the Final Safety Analysis Report (FSAR). The acceptance criteria (i.e. No Departure from Nucleate Boiling) stated in the FSAR was met for the postulated event without credit for the automatic actuation of engineered safety features. The worst case accident analyzed for the FSAR remains bounding for the accident postulated in IE 95-10. Therefore, the condition was determined to be within the design basis as described in the FSAR."

Item Number	Status	Description and Reference
95-02-02	Closed	URI - Substitution of operator action for a loss of ESF function.

95-15-01	Open/Closed	NCV - Failure to perform the TS required surveillance on the CCW valves associated with the "C" charging/safety injection pump.
95-15-02	Open/Closed	NCV - Failure to document and control jumper wires on a RVLIS temperature compensation card.
95-15-03	Open/Closed	NCV - Failure to control the design of a safety-related radiation monitor.
95-15-04	Open	NOV - Failure to have minimum number of fire detection instruments operable and no compensatory patrols were established.

8. ACRONYMS AND INITIALISMS

CCW	Component Cooling Water
DNBR	Departure From Nucleate Boiling Ratio
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
GL	Generic Letter
I&C	Instrumentation and Control
LER	Licensee Event Report
MWR	Maintenance Work Request
NCV	Non Cited Violation
NOV	Notice of Violation
NRR	Nuclear Reactor Regulation
PMTS	Preventive Maintenance Task Sheet
PSRC	Plant Safety Review Committee
RCS	Reactor Coolant System
RWP	Radiation Work Permit
SAP	Station Administrative Procedure
SPR	Special Report
SSPS	Solid State Protection System
STP	Surveillance Test Procedure
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item