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

Report No.: 50-395/92-10

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

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

Facility Name: V. C. Summer Nuclear Station

Inspection Conducted: May 1-31, 1992

Inspectors: For R.W. Wright 6/22/92
R. C. Haag, Senior Resident Inspector Date Signed

For R.W. Wright 6/22/92
L. A. Keller, Resident Inspector Date Signed

Approved by: Floyd S. Cartrell, Jr. 6/22/92
Floyd S. Cartrell, Chief Date Signed
Project Section 1B
Division of Reactor Projects

SUMMARY

Scope:

This routine inspection was conducted by the resident inspectors onsite in the areas of monthly surveillance observations, monthly maintenance observations, and operational safety verification. Selected tours were conducted on backshift or weekends. Backshift or weekend tours were conducted on four occasions.

Results:

Two violations were identified.

The first violation involved equipment control and procedures associated with equipment control, and had two examples. The first example involved the degraded voltage protection for one bus being inadvertently rendered inoperable, due to the failure to fully understand the effects of removing control power to a relay (paragraph 4). The second example dealt with an inadequate procedure for the adjustment of radiation monitor setpoints (paragraph 5.b). The second example indicates that improvements in the control of radiation monitor setpoints are warranted.

The second violation involved the failure of procedural requirements by not referring to the applicable technical manual for additional instructions (paragraph 5.c).

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Temporary repairs for the feedwater isolation valves and pressurizer relief tank level transmitters will require permanent repairs in the future (paragraph 4). The licensee's program for assessment of leaks containing boric acid lacked adequate guidance to ensure all leaks receive the required level of inspection and evaluation (paragraph 5.e). The post trip review effort for the second reactor trip was thorough and exhibited strong management support to identify and resolve the cause of the trip (paragraph 5.d).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

F. Bacon, Associate Manager, Chemistry
*W. Baehr, Manager, Chemistry and Health Physics
K. Beale, Supervisor, Emergency Services
*C. Bowman, Manager, Maintenance Services
*M. Browne, Manager, Design Engineering
*B. Christiansen, Manager, Technical Services
H. Donnelly, Senior Engineer, Nuclear Licensing
M. Fowlkes, Associate Manager, Shift Engineering
*S. Furstenberg, Associate Manager, Operations
*G. Hall, Associate Manager, Health Physics
*W. Higgins, Supervisor, Regulatory Compliance
*S. Hunt, Acting General Manager, Nuclear Safety
*A. Koon, Manager, Nuclear Licensing
*J. Nesbitt, Supervisor, Instrumentation and Control
*K. Nettler, General Manager, Station Support
*H. O'Quinn, Manager, Nuclear Protection Services
*M. Quinton, General Manager, Engineering Services
*J. Skolds, Vice President, Nuclear Operations
*G. Taylor, General Manager, Nuclear Plant Operations
A. Torres, Associate Manager, Quality Control
B. Williams, Manager, Operations

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

*Attended exit interview

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

2. Plant Status

The plant operated at or near 100 percent power until the unit was shutdown and the reactor taken to cold shutdown condition on May 12, 1992. In addition to the main repair activity of correcting a secondary S/G manway leak, a pressurizer PORV and moisture separator reheater tube leaks were repaired. During the plant startup on May 20, 1992, a reactor trip from three percent power occurred due to personnel error while repairing the power range instrumentation. A second reactor trip occurred on May 21, 1992, during plant startup. The trip was caused by low S/G level due to closure of a feedwater isolation valve. On May 22, 1992, the main generator was tied to the grid and 100 percent power established on May 24, 1992.

3. Monthly 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 technical specifications and procedure requirements, any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel, and the systems were properly returned to service. Specifically, the inspectors witnessed/reviewed portions of the following test activities:

- Annual test of electric driven fire service pump XPP134A (STP 228.001). Both the total flow and starting pressure setpoints were verified during the test.

- Leak rate test of the reactor building (RB) personnel hatch airlock seals (STP 215.001A). The leak rate test was required due to the use of the personnel airlock for support of the S/G manway repairs. The inspector reviewed the test logs and verified that leak rate tests were performed within the 72 hour time interval required by TS 4.6.1.5 for the time period that the personnel hatch was used to support work in the RB.

- Valve operability testing while shutting the plant down (Modes 1, 2 and 3) (STP 130.003). While observing the turbine driven EFW pump operate for testing of associated EFW valves, the inspector noted a steam leak at the packing for the turbine governor valve XVM11025. An MWR was initiated to repair the steam leak.

- Calibration of neutron flux source range detector N-31 bistable at shutdown (1CP310.008).

- Safety-related chill water balance check with throttle valves fully open for "A" train (GTP 216). MRF 22114 installed orifices in selected individual legs of the safety-related chill water system. This modification was necessary to prevent having to throttle flow to individual loads via throttle valves, and the need to verify adequate flow through each leg whenever one of the throttle valve's position was changed. This test measured the flow across each coil or component, with all throttle valves wide open. The test was repeated for all feasible combinations of equipment in service (i.e., chill water pumps, charging pumps, component cooling water pumps). The test results were satisfactory and the subject throttle valves were left in the fully open position.

- Operational testing of containment purge supply valves PVB-1A and PVB-1B and purge exhaust valves PVB-2A and PVB-2B
(STP 130.05B). This test verified acceptable valve stroke time and travel. After initial opening of the valves an automatic closure signal closed the valves. This was caused by radiation levels in the purge flow exceeding the trip setpoint of the purge flow radiation monitor (RM-A4). During Modes 1, 2, 3 and 4 the trip setpoint for RM-A4 is two times background levels. However, when stroking the valves as allowed in mode 5, the setpoint is based on the requirements of the Offsite Dose Calculation Manual. The setpoint is provided on a gaseous waste release permit.

The new RM-A4 setpoint, which had been calculated for 188,000 CPM by health physics personnel, was given to operations via the release permit prior to operating the purge valves. However, RM-A4 was not adjusted for the new setpoint. Confusion among operation's personnel on the need to adjust the setpoint and poor communication regarding the status of the adjustment resulted in the setpoint adjustment not being completed prior to testing the purge valves. Also, the Surveillance Test Procedure, STP 130.005B, did not provide instructions for adjustment of RM-A4 setpoint prior to stroke testing the purge valves. This procedure was inadequate in that it did not recognize nor require the accomplishment of a critical prerequisite, i.e., setpoint adjustment prior to testing the containment purge valve. Poor communications among operators contributed to the failure to adjust RM-A4 setpoint and the resulting automatic closure of the containment purge valves. This lack of procedural detail appears to be a weakness that should be reviewed by the licensee.

- Loop calibration of "C" steam generator feedwater flow transmitter (STP 395.009).

One violation involving an inadequate procedure for the testing of containment purge valves was identified. The remaining activities demonstrated acceptable results.

4. Monthly 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, and industry codes or standards and in conformance with TS.

The following items were considered during this review: that 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 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:

- Investigation and repair of the alarm function for electrical switchgear room cooling fan XFN106A (MWR 9200015). Previously operators had discovered that the fan breaker had tripped open, however, the associated alarm had not come in. Inspections in the HVAC control board termination cabinet identified a broken wire on an alarm module chassis. The wire was replaced and the alarm was retested satisfactorily.

- Re-pressurization of electrical penetration XRP0018 (MWR 9203443). The sulfur hexafluoride cover gas pressure had been identified as below the minimum 15 psig pressure during the last 90 day pressure verification. Since this was the second occurrence of low pressure for XRP0018, NCN 4471 was written. The previous low pressure was identified in February 1991. Due to the slow rate of the pressure decrease (from 30 psig to 14 psig in 14 months), the NCN disposition directed re-pressurization of the penetration and a check for air leaks. No leaks were identified. Engineering will provide an additional NCN evaluation, however, based on the leak rate the licensee does not consider this problem as having a current impact on operation of the penetration.

- Preventive maintenance task to calibrate loop "B" service water flow indicator IFI4587 (PMTS P0156278).

- Installation of two diodes (in series) in the pressure switch circuit for "B" FW isolation valve XVG1611B (MWR 92I3084). The licensee had previously experienced a failure of the pressure switch for "A" FW isolation valve when the switch contacts stuck together. Investigations by the licensee identified that the current surge experienced when the contacts in this 120 VDC circuit opened was the cause of pitting and arcing of the contacts and their subsequent sticking together. The diodes were installed to isolate the contacts from the current surge in the circuit. Diodes were also installed in the pressure switch circuit for "A" and "C" FW isolation valves. The licensee believes the current surge is common to DC powered circuits. Therefore, the diode installation is considered a short term corrective action until a plant modification is completed for an AC power supply to the pressure switch circuit.

- Troubleshooting and repair of degraded voltage relay 27B-1DA-1 on safeguards bus 1DA (MWR 9203463). On May 3, 1992, an operator noted that the relay flag was indicating the relay had tripped. Since bus voltage was normal, operators tried to reset the relay but were unsuccessful in clearing the tripped indication. Due to the unknown condition of the relay and the scope of the required repairs not being fully developed, the fuse for the 120 VDC control power to the relay was pulled. The licensee's basis for removing the fuse was their understanding that the relay, without control power, would fail to it's safety position, i.e., contacts open.

  The degraded voltage protection scheme for the 7.2 kV safeguards buses has three relays per bus. One relay monitors each bus phase. The protection logic requires a three out of three actuation of the relays to generate a degraded voltage signal. The output contacts which are normally closed will open on a degraded voltage condition. The degraded voltage relays are Asea Brown Boveri (ABB) solid state, type IFE 27N relays. They were installed during the last refueling outage (1991) due to increased accuracy over the older style ITE 27E relays.

While testing a replacement relay for 27B-1DA-1 on May 4, 1992, the licensee discovered that the output contacts for the ABB type IFE 27N relay failed to the closed position when control power was de-energized. Based on this, the licensee recognized that by removing control power to the relay they had defeated the degraded voltage protection from bus 1DA. With the output contacts for one relay failed close, then the three out of three logic (with the relay output contacts open) could never be satisfied. The licensee also recognized the action statement of TS 3.2.2 should have been entered when the fuse was pulled. The replacement relay was installed and the TS LCO exited. Subsequent testing in the shop revealed that the relay time delay feature was out of calibration, but the cause of the relay not resetting could not be identified. The licensee plans to send the relay to ABB for additional testing and repair.

While reviewing technical information on the degraded voltage protection circuit, the inspectors noted that electrical elementary drawing B-208-037 does not depict the position of the relay output contacts when control power is removed. However the technical manual, IB 7.4.1.7-7, for the degraded voltage relay clearly shows the output contacts in the closed position when no control power is applied. When the licensee made the decision to pull the control power fuse, only the electrical drawings were used to assess the impact of the fuse removal. Instructions for the control of equipment and system status are provided in SAP 205, "Status Control and Removal and Restoration". Paragraph 2.2.2 of SAP 205 requires the implementation of a R&R checksheet anytime a system required to be operable is found inoperable or made inoperable. The licensee's failure to fully understand the effects of removing control power for a degraded voltage relay is the first example of Violation 50-395/92-10-01 failure to maintain adequate control of equipment

The modification which installed the new style relay during the last refueling outage contributed to the lack of understanding. The output contacts for the older style relays failed to the open position on a loss of control DC power. The information that was added to the electrical drawing as a result of the modification does not provide a clear indication that the new relays output contacts fail to the closed position. A review of the relay technical manual is required to fully understand the positioning of the output contacts on a loss of control power.

While reviewing this event, the inspectors noted the additional vulnerability of the degraded voltage protection system which resulted from installation of the new style relay. With the current system configuration, a loss of DC control power to one relay would defeat the entire degraded voltage protection for that train of safeguards electrical power. A loss of control power to only one relay would not be easily recognized since there is no indication of control power to individual components such as the degraded voltage relays. Single failure criteria is applicable at the redundant bus level, therefore, the loss of control power is not an issue with single failure criteria. Yet the modification that installed the new relays did add a new failure method for the loss of a train of degraded voltage protection. The licensee is reviewing the current design of degraded voltage protection to determine if improvements can be made to prevent this type of failure.

- Troubleshooting and repair of pressurizer relief tank (PRT) level indicators ILT470 and 470A (MWR 9203589 and 9203600). Initially the two level indicators, one on the main control board and the other on the remote shutdown panel, would not pass a channel check test. Then the licensee noted that the level indications were erratic when actual PRT level or pressure changed. The problem was corrected when high pressure nitrogen was blown through the common upper PRT sensing line. The licensee believes that water had accumulated in a horizontally mounted diaphragm valve in this sensing line and was partially blocking air flow. This prevented equalization of pressure between the PRT and the level transmitter. The correction of the PRT level indicators was a "temporary accept-as-is" resolution until the diaphragm valve can be mounted in a vertical position. Instructions were issued to monitor and trend both level indicators every time the PRT is pumped down (approximately two times per week) to detect any future accumulation of water in the valve.

- Repair of pressurizer power operated relief valve (PORV) PCV445B (MWR 9102325). The valve was disassembled and new valve internals were installed to correct previous seat leakage. Due to reports from utilities concerning incorrect bolting material for similar Copes Vulcan valves, the bolts used to attach the valve yoke to bonnet were inspected. The licensee discovered that three of the four bolts were non-magnetic. The vendor drawing specified material is ASTM A193 B6 which is magnetic. Testing by Copes Vulcan and other utilities determined that the bolts from other valves manufactured during the same time period were ASTM F837 type XM7. A technical evaluation by Copes Vulcan of XM7 bolts determined the maximum seismic loading for the worst case valve operator restrained by four bolts.

  The bolts in PCV445B were replaced with A193 B6 bolts. An inspection of the bolts in the other two PORVs identified magnetic material. Based on testing of the bolts removed from PCV445B and comparison with other test results, the licensee concluded that the material for the three non-magnetic bolts was XM7. Using this information and the evaluation from Copes Vulcan on loading capacity for XM7 bolts, the licensee temporarily accepted the bolts in the other two PORVs until the next refueling outage when they will be replaced. A similar justification was used to temporarily accept six other Copes Vulcan valves that were manufactured during the same time as the PORVs and have the potential for use of XM7 bolts. These six valves were not inspected due to ALARA considerations and interferences that would have to be removed.

  Testing PCV445B after the repairs verified the leakage had been corrected, however, a concern on the valve open stroke time was noted. During successive stroking of the valve to determine the baseline stroke time, the open times would start to increase after several valve cycles. Engineering evaluated the data and determined that the valve would have consistent opening stroke times for two cycles, then opening times could increase for additional valve cycles. The cause of the increase in open stroke time is believed to be a defective air regulator or a partially clogged air line to the air accumulator. Closing stroke times are not effected because the valve is closed by spring pressure in the operator. Due to the limiting open time (5.0 to 5.5 seconds) assumed in the discharge piping stress analysis, the licensee has not been able to declare

PCV445B operable. Engineering is pursuing two options that will either accept the available two valve stroke cycles or will increase the maximum opening time assumed in the discharge piping stress analysis. The inspectors will followup on the licensee's effort to return PCV445B to operable status. This item will be identified as IFI 395/92-10-03, Failure of PORV (PCV 445B) to open satisfactorily after successive strokings of the valve.

A second example of the violation involving the failure to control equipment was identified when a degraded voltage relay repair effort inadvertently resulted in a bus of degraded voltage protection being inoperable. Continued engineering and management support is needed to return a PORV to operable status. Permanent repairs are needed to resolve the problems with PRT level indication, PORV bolting material, and the FW isolation valve pressure switches which were corrected or accepted on a temporary basis.

5. Operational Safety Verification (71707)

a. Plant Tour and Observations

The inspectors conducted daily inspections in the following areas: control room staffing, access, and operator behavior; operator adherence to approved procedures, TS, and limiting conditions for operations; and review of control room operator logs, operating orders, plant deviation reports, tagout logs, jumper logs, and tags on components to verify compliance with approved procedures.

The inspectors conducted weekly inspections in the following areas: verification of operability 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.

Plant tours included observation of general plant/equipment conditions, fire protection and preventative measures, control of activities in progress, radiation protection controls, physical security controls, 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.

The inspector observed the final check-out testing of the new FTS 2000 telecommunication system at V. C. Summer. The FTS 2000 system was recently connected to the various emergency response facilities at the plant. The purpose of FTS 2000 is to replace the old emergency notification system (ENS) and to provide additional communication functions for emergency response. The final check-out testing was the last portion of the "30 day dual operation test". All locations and functions of FTS 2000 were tested satisfactorily. The licensee was subsequently informed that the FTS 2000 was approved for sole use as the emergency telecommunications system and that the old ENS would be removed.

b. Control of Setpoints for the Radiation Monitoring System (RMS)

To resolve an alarm condition for the SPDS containment critical safety function, the alarm setpoints for the containment high range radiation monitors (RM-G7 and RM-G18) were changed. On May 1, 1991, actions were initiated to change the alarm setpoints from one R/hr to two R/hr. The "RMS Setpoint Change Log" which is maintained in the control room was updated with the new setpoints. The HP technician then attempted to change the alarm setpoints using the adjustments on the front control panel for RM-G7 and RM-G18 monitors. However, this adjustment only affected the "zero scale" on the meters located in the control room. The actual alarm setpoints were not affected by the adjustment. To change the alarm setpoint for the high range radiation monitors, i.e., RM-G7 and RM-G18, an adjustment inside the control drawer is required. This is normally accomplished by I&C technicians per instructions in the surveillance test procedures for the monitors.

On May 3, 1992, while reviewing the status of the setpoint changes, licensee personnel identified that the actual alarm setpoints for RM-G7 and RM-G18 had not changed but only the meter calibration had been altered. Both RM-G7 and RM-G18 were declared inoperable per TS 3.3.3.1 until the meters were recalibrated and the setpoints were properly adjusted.

The requirement for I&C to perform the RM-G7 and RM-G18 setpoint adjustments was known by certain members of the operation's shift. Yet this information was not passed on to the operator that was authorizing work. This allowed the improper adjustment of the alarm setpoint which resulted in the control room meters for RM-G7 and RM-G18 being out of calibration. Health Physics Procedure, HPP 904, provides instructions for use of the RMS. The inspector noted that HPP 904 deals mainly with establishment and control of radiation monitor setpoints. However, no instructions are provided for the actual adjustments of setpoints or which groups are responsible for performing these adjustments. HPP 904 was inadequate for the control and use of the RMS due to the lack of instruction for radiation monitor setpoint adjustments. This inadequate procedure will be documented as the second example of Violation 50-395/92-10-01. This event is similar to another example (paragraph 3, page 3) in that operations personnel were aware of the information that was not provided by the procedure; however, poor communication among operators resulted in the failure to prevent these errors.

c. Reactor Trip Due to Personnel Error

At 11:25 AM on May 20, 1992, during a plant startup the reactor tripped from approximately three percent power. Equipment response was normal. The actual trip signal was high neutron flux on intermediate range channel NI-36. The licensee determined the trip signal was caused by a related maintenance activity on neutron power range channel NI-42. At power less than ten percent, the one out of two logic for the intermediate range high neutron flux trip is unblocked. One 120 VAC power supply is provided to the cabinet which contains NI-36 and NI-42. All the equipment in the cabinet uses the same power supply.

The maintenance activity on NI-42 involved replacement of the high voltage power supply. To isolate the 120 VAC from the high voltage power supply, the fuses at the front of the drawer were removed. To remove the power supply, the I&C technicians believed that the hold down fasteners on the bottom of the drawer had to be removed. Due to the close proximity of a capacitor to these fasteners, the technicians decided to discharge the capacitor. Since the electrical schematic drawings of the power range drawer were not reviewed for replacement of the power supply, the technicians were unaware that the capacitor was not isolated by the fuse removal. When the capacitor was discharged (connected to a ground) a spike occurred on the 120 VAC supply to the cabinet which resulted in a trip of the intermediate range high flux bistable and a subsequent reactor trip.

The technical manual for the power range instrumentation, 1MS-94B-016, Volume 1, provides instructions for replacement of high voltage power supply. The instructions require de-energizing the drawer assembly by removing the instrument power fuses and disconnecting the primary power input connector at the rear of the drawer. In addition, the instructions stated that the power supply can be slid out the back of the drawer, therefore, removal of the hold down fasteners is not required. The I&C technicians admitted that the technical manual was not referred to during the maintenance activity. General Maintenance Procedure, GMP 100.005, provides general guidance and information for I&C tasks which do not require detailed instructions. Prerequisite 3.5 of GMP 100.005 states, "If required for the technicians further information, obtain an approved copy of the particular technical manual." Also, step 7.1.1 for removal of equipment requires electrical input be isolated whenever possible. The failure to [illegible] the procedural requirement of GMP 100.005 by [illegible] reviewing the technical manual prior to the replacement of an NI-42 power supply is a Violation 50-395/92-10-02. The inspector considers the current instructions for the use of technical manuals as a weakness and contributed to the failure to use the technical manual for this event.

d. Reactor Trip During Plant Startup

At 5:53 AM on May 21, 1992, the reactor tripped due to a "Low-Low" level signal on "C" steam generator (S/G). The low level was caused by the closure of "C" feedwater isolation valve (FWIV). Initially, the plant was at 28 percent power when the isolation valve closed and the operators had reduced power to 12 percent prior to the "Low-Low" signal. The licensee's post trip review identified that the FWIV closure was caused by a FW protective feature which isolates FW to a S/G when FW temperature is below 225 degrees Fahrenheit and FW flow is less than 13 percent. This protective feature is designed to prevent water hammer in the S/G preheater section and to prevent excessive stresses in the bolting material for the S/G FW baffles during low flow conditions. The licensee attributed the low FW temperature to the slow turbine "roll up" time (30 minutes versus a normal 20 minutes). The low FW flow condition was attributed to swings in mainsteam (MS) flow which resulted in swings with FW flow since S/G level control was in automatic.

Following the trip, no problems were identified while troubleshooting the turbine speed control system. During the subsequent startup the turbine speed control system operated as designed. The cause of the MS flow swings was the opening and closing of the MS PORVs. Increases in S/G pressure (up to the PORV relief setpoint) was the cause of the PORVs opening, then as pressure decreased the PORVs closed. At 28 percent power, the condenser steam dump valve should be able to maintain control of S/G pressure. The eight dump valves have a total steam dump capacity of approximately 48 percent rated power. When the dump valves were later tested, six of the eight valves only opened approximately ten percent. These six valves are grouped together with a single controlling driver card providing the open and close signal. All eight dump valves were modified in 1991, when the style of the controller for the valve actuator was changed. During their review, the licensee identified that the driver card which provides the electrical signal to operate the valves was not properly sized for six valves with the new style controller. The controllers were replaced under the licensee's "Equal To or Better Than" program. During the review for this replacement, the difference in the impedance between the different controllers was not considered. Since only two controllers were replaced at a time, the subsequent stroke test was satisfactory because the "driver card" was adequate to operate a maximum of three dump valves. This problem was resolved by installing the old style controllers.

The inspector observed the initial power increase and closing the main generator output breaker for the plant startup on May 22, 1992. The operators were briefed on possible problems with the turbine roll up and required action. Also, parameters dealing with FW flow and temperatures were closely monitored. The licensee performed a thorough post trip review and verification that all corrective actions were completed prior to the startup. Management's attention during these efforts was viewed as a strength.

e. Review of Boric Acid Leakage Assessment Program

During the initial walkdown of the RB after the plant shutdown, the licensee identified three components as having, or previously had, leakage which contained boric acid. During the outage two additional leakage locations were identified. With the exception of XVT8146 and 8147 (normal and alternate charging valves) all the leaks were repaired. Due to isolation difficulties and the small amount of leakage, the licensee accepted the body to bonnet leakage from XVT8146 and 8147.

The inspector attempted to review the licensee's assessment of the identified leaks which contained boric acid. The inspector referred to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants", for guidance in determining critical elements that should be included in the licensee's assessment. The inspector noted that all possible leaks with boric acid are not programmatically inspected and evaluated for potential impact on the RCS pressure boundary. Examples were leakage at the cap for vent valve XVT8363A and leakage from vent valve CV-5. Since these valves are not ASME code pressure boundary components, they did not receive specific QC inspection to identify any damage as a result of the leakage. A general housekeeping inspection was performed by QC in these areas at the completion of work activities. The inspector questioned if this type of inspection was adequate to identify and assess any related damage due to the leakage. This was a particular concern for the leakage from XVT8363A since the inspector had noted a large amount of boric acid crystals on the RB basement floor as a result of XVT8363A leakage. During the cleanup approximately 60 to 70 pounds of crystals were removed.

For the leakage from XVT8146 and 8147 that was temporarily accepted until the next refueling outage, no documentation was available for the assessment of any current damage or potential damage. The licensee informed the inspector that they had reviewed both the current impact or any future impact of these leaks on the RCS pressure boundaries, but this review was not documented. Based on the Generic Letter, the inspector considers that these types of reviews/assessments should be performed per programmatic guidance and should include documentation of the results.

After discussion of the inspector's concerns with the licensee, QC performed an inspection of the boric acid leakage locations. To facilitate this inspection, the boric acid crystals on XVT8146 and 8147 and a carbon steel spring can were removed. QC documented the inspection results and an engineering evaluation was provided to accept the conditions based on no visible damage. The inspector was informed that the licensee will review their boric acid leakage assessment program to determine the required changes for consistent implementation of the program.

Another example of the violation involving failure to control equipment occurred when radiation monitor setpoint adjustments were performed incorrectly. A violation was identified for the maintenance activity on a neutron power range channel which resulted in a reactor trip. Management's attention and corrective actions following the second reactor trip were viewed as a strength. The inspector noted that all possible leaks with boric acid are not programmatically inspected and evaluated for potential impact on the RCS pressure boundary.

6. Exit Interview (30703)

The inspection scope and findings were summarized on June 2, 1992, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed the inspection findings.

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 inspectors during the inspection.

Item Number	Description and Reference
395/92-10-01 Example 1	Violation - Failure to adequately control equipment, while investigating the cause of a relay failure, paragraph 4.

395/92-10-01 Example 2	Violation - Inadequate procedure for the control of radiation monitor setpoint adjustments, paragraph 5.b.
395/92-10-02	Violation - Failure to follow procedural requirement for the use of a technical manual when performing a maintenance activity, paragraph 5.c.
395/92-10-03	Inspector Followup Item - Failure of PORV (PCV 445B) to open satisfactorily after successive strokings of the valve, paragraph 4.a.

7. Acronyms and Initialisms

ABB	Asea Brown Boveri
AC	Alternating Current
ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
DC	Direct Current
EFW	Emergency Feedwater
ENS	Emergency Notification System
ESF	Engineered Safety Feature
FW	Feedwater
FWIV	Feedwater Isolation Valve
GMP	General Maintenance Procedure
GTP	General Test Procedure
HPP	Health Physics Procedure
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Control
KV	Kilovolt
LER	Licensee Event Reports
MRF	Modification Request Form
MS	Mainstream
MWR	Maintenance Work Request
NCN	Nonconformance Notice
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PMTS	Preventive Maintenance Task Sheet
PORV	Power Operated Relief Valve
PRT	Pressurizer Relief Tank
PSIG	Pounds Per Square Inch Gauge
QC	Quality Control
RB	Reactor Building

RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RWP	Radiation Work Permits
SAP	Station Administrative Procedure
S/G	Steam Generator
SPDS	Safety Parameter Display System
SPR	Special Reports
STP	Surveillance Test Procedures
TS	Technical Specifications
VAC	Voltage Alternating Current
VDC	Volts Direct Current