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

Report No.: 50-395/94-07

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

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

Facility Name: Virgil C. Summer Nuclear Station

Inspection Conducted: February 1-28, 1994

Inspectors: Charles A Hough 3/11/94
for R. C. Haag, Senior Resident Inspector Date Signed

Charles A Hough 3/11/94
for T. R. Farnholtz, Resident Inspector Date Signed

Approved by: Floyd S Cantrell 3/11/94
Floyd S. Cantrell, Chief Date Signed
Reactor Projects 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, operational safety verification, fire protection, design changes and modifications, review of licensee self-assessment capabilities, review of maintenance training, onsite follow-up of written reports of nonroutine events at power reactor facilities, and action on previous inspection findings. Selected tours were conducted on backshift or weekends. These tours were conducted on seven occasions.

Results: (Summarized by SALP functional area)

Operations

An inspector followup item was identified for further review of main steam isolation valve testing adequacy and incorporation of additional guidance for valve adjustments into the maintenance procedures. A power reduction for fuel conservation was conducted in a controlled manner and received appropriate management oversight.

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Maintenance and Surveillance

Poor procedural documentation was noted for some preventive maintenance activities. The initial proposed post maintenance testing requirements for a diesel generator tank level switch were not complete; however, as conducted, the tests were satisfactory. The maintenance qualification program was well structured with additional improvements currently being implemented.

Engineering and Technical Support

A violation was identified for inadequate design control of a plant operating parameter that exceeded the design basis specification (paragraph 7). Some Coordinated Manual Control switches were not included in the knob replacement program. The basis for this omission was not well documented.

Plant Support

A non-cited violation was identified for propping open a fire door without the required controls being implemented. The hourly roving fire watch patrols in the cable spreading rooms were limited in scope. Walkdown inspections by the licensee of the fire suppression systems have identified several discrepancies and indicate that additional reviews are needed in this area. An unresolved item was identified involving roving fire water duties (paragraph 6.a.) and an inspector followup item involving the number of sprinklers and adequacy of flow to sprinklers (paragraph 6.b.) were identified.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

W. Baehr, Manager, Health Physics
C. Bowman, Manager, Maintenance Services
M. Browne, Manager, Design Engineering
L. Faltus, Acting Manager, Chemistry
M. Fowlkes, Manager, Nuclear Licensing & Operating Experience
S. Furstenberg, Associate Manager, Operations
*S. Hunt, Manager, Quality Systems
*D. Lavigne, General Manager, Nuclear Safety
*J. Nesbitt, Acting Manager, Technical Services
*K. Nettles, General Manager, Station Support
*H. O'Quinn, Manager, Nuclear Protection Services
*J. Proper, Supervisor, Nuclear Licensing & Operating Experience
*M. Quinton, General Manager, Engineering Services
J. Skolds, Vice President, Nuclear Operations
*G. Taylor, General Manager, Nuclear Plant Operations
*R. White, Nuclear Coordinator, South Carolina Public Service Authority
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 approximately 100 percent power until February 9, 1994, when power was reduced to 65 percent to conserve fuel and allow full power operation through the anticipated peak load period prior to the September 1994 refueling outage.

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, and the systems were properly returned to service. Specifically, the inspectors witnessed/reviewed portions of the following test activities:

a. Quarterly leak test of train "B" component cooling water/service water cross connect check value XVC9680B (STP 122.003). This test was performed to satisfy ASME Section XI testing requirements. The measured leakage was within the acceptance criteria.

b. Quarterly test of train "B" service water pump XPP39B and service water booster pump XPP45B (STP 223.002A). When testing XPP39B the STP required that a flow rate of 10,000 to 11,000 GPM be obtained. This was accomplished by throttling the flow to individual components. The procedure did not require recording of the "as found" flow rates nor did the procedure mention the re-establishing of initial flows at the completion of the test. During the actual testing the inspector noted that flows were properly re-established; however, the lack of guidance in this area appeared to be a procedural weakness. In response to the inspector's questions, the licensee stated that the procedure would be revised to restablish the "as found conditions".

c. Power range heat balance (STP 102.002). This surveillance test is performed daily. No adjustments to the nuclear instrumentation were required.

d. Ventilation fire damper inspection (STP 428.060). Several dampers in the control room ventilation system were visually inspected. No discrepancies were noted.

The observed surveillance test were performed in a satisfactorily manner and met TS requirements. A procedure weakness was noted for the service water pump STP as indicated in paragraph b. above.

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.

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:

a. Investigation and repair of the low cylinder exhaust temperature on "A" emergency diesel generator (EDG) (MWR 9403118). During a surveillance test of the EDG on February 1, 1994, the operator recorded a maximum differential temperature (ΔT) of 153°F between the cylinder exhaust temperature readings. A note on the operator's log following a previous EDG run stated to seriously consider immediate shutdown of the diesel if the maximum ΔT limit of 150°F is exceeded. The log readings that recorded the 153°F ΔT were taken at the end of the scheduled one hour run and the EDG was in the process of being shutdown when the operator determined that the ΔT was greater than 150°F. The earlier log reading had a ΔT reading that was less than 150°F.

   The licensee's troubleshooting revealed that the No. 1 cylinder exhaust temperature thermocouple was providing inaccurate indication. This and an additional thermocouple were replaced. The inspector observed the subsequent EDG maintenance run during which cylinder exhaust temperatures were within their normal ranges.

b. Annual preventive maintenance on "A" relay room cooling unit inlet damper XDP101A (PMTS P0174571) and inlet filter XFL056A (PMTS P0174577). During a conversation with the inspector, the mechanics stated that they had previously completed similar PMs on two other dampers associated with the relay room ventilation system. While the mechanics were completing signoffs on the data sheet for damper XDP101A, the inspector noted that the mechanics also started to fill out the data sheets for the two damper PMs which were already completed. This was a preventive procedure that a qualified person is allowed to use by reference but it still requires signoff as indicated. As used in this case, the inspector questioned the practice of completing signoffs for individual steps in the procedure after all the work had been performed. In later discussions, the licensee's management informed the inspector that their expectation for work documentation was to complete an individual signoff once that portion or step of the procedure is completed. This was a procedure that a qualified person is allowed to use by reference, but it still requires signoff as indicated. As performed in this case, the inspector considered this a poor example of procedural documentation. The licensee re-emphasized to maintenance personnel their expectations on documenting completion of procedural instruction.

c. Visual inspection, testing, and lubrication of the "B" charging/safety injection pump motor (PMTS P0175863).

d. Radiograph inspections of a six inch fire service (FS) line in the auxiliary building, elevation 412 (MWR 94Q3007). This inspection was in response to the degraded FS flow condition that was recently identified (see NRC Inspection Report No. 50-395/94-03). Flow testing had identified a 470 ft. long section of six inch carbon steel piping as the cause of the low flow condition. The radiograph and eddy current inspections confirmed a reduced flow area in this section of pipe. To resolve this problem, the licensee is reviewing options for cleaning the piping and/or adding parallel piping to increase flows.

e. Leakage repairs for FS deluge valve XVM6760 (MWR 93M3222).

f. Preventive maintenance to inspect and lubricate the Limitorque operator for component cooling water cross connect valve XVB9526A (PMTS P0174365). The lubrication instructions in the procedure were separated based on the type of lubricant installed in the operator. The basis for the separate instructions was to prevent mixing of incompatible lubricants (See Inspection Report 50-395/94-04). The electricians and the inspector noted confusing instructions in the procedure dealing with the required actions to be taken when a particular type of lubricant is present. The electricians questioned their supervisor on this item and were provided guidance to complete the work. The inspector was informed that a procedural improvement would be made to provide clarification for this item.

g. Preventive maintenance on the post accident sampling system. Observed tasks included a span check of pH meter (PMTS P0174663) and an operational check of post accident sampling system nitric acid pump (PMTS P0174664).

h. Visual inspection, testing, and lubrication of the "B" control room cooling unit fan motor XFN0032B (PMTS P0174916) and the emergency filtering system fan motor XFN0030B (PMTS P0174700).

i. Replacement of level switch, ILS5411, for the "A" emergency diesel generator (EDG) fuel oil day tank (MWR 9313370). On February 9, 1994, the licensee determined that the level switches (for "A" and "B" fuel oil day tanks) may not be seismically qualified. These switches control the fuel oil transfer pumps to maintain the proper level in the day tanks. This condition was identified in correspondence with the vendor when the licensee attempted to purchase new switches to resolve an unrelated problem.

Once this condition was identified, the licensee evaluated the potential affects on EDG operation. An NCN documented engineering's recommendations and the temporary "Accept-As-Is" until seismically qualified replacement switches could be installed. As a compensatory measure, an operator was assigned to the EDG building with instructions on the required actions to be performed for a seismic event. The inspector reviewed the operation of the EDG fuel oil system, the compensatory measures the licensee established and concluded that adequate steps were taken to address the operability concern for the EDGs.

The inspector noted that the proposed post maintenance test requirements for the installed seismically qualified limit switch only required that the switch be calibrated on a test stand (no functional test). The inspector questioned the lack of a functional test that included a change in day tank level or operation of the D/G prior to declaring the D/G operable. After questioning by the inspector, the licensee included a functional test that included operating the EDG and verifying that the transfer pump started as necessary to maintain the day tank fuel level. The final testing as conducted was satisfactory.

j. Replacement of switch knobs for the control switches associated with fire service deluge valves XVM4108 and XVM4110 (PMTS P0169387 and P0169388). This work was part of the program to switch knobs on certain switches identified as model "CMC" switches. In May 1985, a NCN was written due to numerous CMC switch knob failures on the main control board (MCB). The licensee's investigation revealed that the cause of the knob failures was embrittlement of the plastic material due to prolonged heat exposure. As a result of the continuous illumination of the switch indicating lights. The NCN's corrective action was to install new knobs made of a material that was not susceptible to embrittlement.

While responding to an event in March 1993, the knob broke for a CMC switch that the operator was attempting to manipulate. The operator was able to install a temporary knob and continue with the required action without adverse affects. During the inspector's review of that event, it was noted that a total of 139 knob failures had occurred and 65 of the original CMC switch knobs remained on the MCB. This review was documented in NRC Inspection Report No. 50-395/93-14, which also requested that the licensee respond in writing regarding their plan to complete the switch knob replacement effort. In a May 28, 1993, letter the licensee stated that 452 of the 542 deficient switch knobs had been replaced; and the remaining switches were scheduled to be replaced by the end of the upcoming eighth refueling outage.

During a later tour of the plant, the inspector noted that a CMC switch was installed on the local control panel for "B" and "C" chill water system chillers. In response to the inspector's question regarding replacement of these switch knobs, the licensee stated that these switches were not included in the replacement program. While reviewing the original NCN, the inspector noted that all CMC switches in the plant were not specified for knob replacement. The NCN mentioned the MCB, the control room HVAC panel, and the CREP as switch locations where the knobs were to be replaced. The NCN did not specifically address other CMC switches in the plant nor the basis for not replacing these knobs. During recent conversation with licensee's engineering personnel, the inspector was informed that they believe the other CMC switch knobs did not require replacement. This was based on different installation environments for the switches.

The licensee has included other switches, i.e., CMC switches located on local radiation monitor panels, in the scope for knob replacement. According to maintenance personnel, their intent was to replace all CMC switch knobs. The licensee believes the chiller switches were missed in their effort to identify all knobs to be replaced. The licensee did not know how many other original CMC switch knobs were in the plant that were not scheduled for replacement. The licensee's current plans are to identify these "other" CMC switches and include them in the replacement program.

The inspector concluded that the licensee's efforts to replace all the CMC switch knobs were not thorough, in that some switches were not included in the replacement program. Also, the original NCN did not provide a thorough evaluation of why the scope of knob replacements did not include all CMC switches in the plant.

In summary, an example of poor procedural documentation was noted for PM activities involving relay room dampers and filters (paragraph 4.b). The originally proposed post maintenance testing requirements for the EDG fuel oil day tank level switch replacements were not complete; but as performed were fully acceptable (paragraph 4.i.). Some of the CMC switches were not included in the knob replacement program in paragraph 4.j. The basis for this omission was not well documented. The overall maintenance program is satisfactory; however, the examples discussed above indicate a need for improvement in these areas.

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, and tags on components to verify compliance with approved procedures.

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 control room ventilation and fire main systems were included in these inspections.

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.

b. Unauthorized Opening of a Fire Door

While touring the control building, 482 level, the inspector noted that fire door, DRCB-605, was propped open. The required fire barrier removal permit that would have allowed this condition was not posted. The inspector notified the control room supervisor who immediately notified the fire protection officer (FPO) and the mechanical maintenance supervisor, who was in charge of the work being performed on the turbine building overhead crane. The FPO closed the door.

Further investigation by the licensee revealed that the door had been opened for approximately one-half hour by the mechanics who were working on the turbine building overhead crane. A fire barrier removal permit had not been obtained. Access to the crane was gained through this door which opened onto a platform in the turbine building. The reason given by the mechanics for propping open the door was that the door had no handles on the turbine building side making entry back into the control building difficult.

Station Administrative Procedure, SAP 131A, requires that fire rated assemblies, including fire doors, be operable at all times. The process for temporary disabling a fire rated assembly, i.e. opening a fire door, is specified in SAP 131A. During previous plant tours the inspectors have noted diligent compliance with SAP-131A when disabling fire barriers. The individuals involved in rendering door, DRCB-605, inoperable in an unauthorized manner were counseled on procedure compliance. Also the licensee's review of SAP-131A to determine if additional guidance is needed for situations when a fire barrier removal permit would not be required, i.e., opening a fire door for a short period of time while an individual remains at the location during the entire time.

This item is identified as NCV 395/94-07-01, Failure to comply with a procedural requirement by the unauthorized rendering of a fire door inoperable. This NRC identified violation is not being cited because criteria specified in Section VII.B of the NRC Enforcement Policy were satisfied.

c. Main Steam Isolation Valve Maintenance and Testing

Two recent events have occurred at U.S. power reactor facilities involving Atwood and Morrill main steam isolation valves (MSIV). The first event involved a MSIV that failed to fully close when called upon to do so and resulted in the emptying of the associated steam generator. As a result of this event, a second reactor site performed MSIV testing as the plant was being shutdown for another reason. This testing was done with the reactor coolant system at normal operating temperature. The test resulted in one MSIVs failing to fully close and one failing to close within the required time period. All of these problems have been attributed to binding in the valve's pneumatic operator due to thermal expansion.

Similar main steam isolation valves are used at the V. C. Summer Nuclear Station. These valves consist of four parallel yolk rods which are mounted to the valve body. A spring plate that is attached to the valve stem is guided in it's movement by the yolk rods. There are guides on the spring plate which are used to limit lateral movement of the plate by adjusting the clearance between the guide and the yolk rod. This clearance is typically set while the valve is at a relatively low temperature. In the two previously described events, the licensees stroke tested the valves from full open to full shut while the valves were at this lower temperature. As the valves are heated up, the geometry of the yolk rods change due to thermal expansion of the valve body causing the clearances between the guides and the yolk rods to be reduced. If the valve is called upon to close at normal operating temperature, the guides may drag or bind on the yolk rods causing a slower closing time or failure to fully close. This occurred in the two event discussed.

The inspector reviewed the maintenance procedure for the MSIVs and the Technical Manual. No mention of the yolk guide adjustment was in either of these two documents. The maintenance procedure (MMP 290.001) covers the disassembly and reassembly of the valve and does require post maintenance testing. This includes cycling the valve and verification of proper stroke time. This procedure did not specify at what temperature this testing would be performed. In addition, the licensee had a surveillance test procedure (STP 130.004) which required the MSIVs to be tested in mode 4. Reactor coolant system average temperature in mode 4 is between 200°F and 350°F. During normal mode 1 power operation, the reactor coolant system average temperature is between 557°F and 587°F. The licensee considered the temperature difference between mode 4 and mode 1 to have negligible effects on the operation of the MSIVs.

The licensee informed the inspectors that the vendor, was developing guidelines to be used to adjust the yolk guides. The licensee planned to incorporate this information into their procedures. In addition, NRC regional and headquarters personnel are reviewing the licensee's testing requirements to determine if they were adequate to give reasonable assurance that the MSIVs would perform as required. These two items will be followed by the inspector as IFI 395/94-07-02, Main Steam Isolation Valve Maintenance and Testing.

6. Fire Protection (64704)

a. Review of Roving Fire Watch Patrol Activities

While in the cable spreading room, elevation 448, on a plant tour, the inspector observed the actions of roving fire watch personnel (RFW). An hourly roving fire watch patrol had been previously established due to the separation criteria of 10 CFR 50, Appendix R not being satisfied for cabling routed through the room. Fire Protection Procedure, FPP-004, Duties of a Fire Watch, describes the requirements for the fire watch program. To document completion of the hourly patrol, the RFW records the RFW code that is listed on the fire watch area tag. These tags are posted in the area requiring the hourly patrols and the RFW codes are randomly changed.

As shown on cable spreading room sketch (Figure 1), the fire watch area tag was located near the entrance door. When the RFW performed the tour the inspector was standing at location "X". During the short time the RFW was in the room, the inspector noted that the RFW did not step beyond the "entrance hallway" and look into the main portion of the cable spreading room. The inspector verified the RFW had entered the room at that time by reviewing the security access records for the room. The duration of the RFW's patrol in the room was 24 seconds. The inspector obtained security access records for cable spreading rooms, elevations 448 and 425, for the February 7-9, 1994, time period. Cable spreading room, elevation 425, had an assigned hourly RFW for the same degraded condition. Review of the 142 hourly RFW records revealed the following observations: 79 patrol durations were less than or equal to 20 seconds; 12 patrol durations were less than or equal to 10 seconds (the shortest duration was 6 seconds); and only 2 patrol durations were greater than 40 seconds.

The inspector questioned the ability to accomplish the duties of a hourly roving fire watch for areas as large as the cable spreading rooms (approximate room sizes are 4700 and 3600 square feet) within these short periods of time. FPP-004 specifies that a RFW patrol through the affected room or area once per hour, look for signs of smoke or fire, and look for items that may create a fire hazard. Also, FPP-004 states to include a brief description of the specific area to be watched (i.e., exact location of the degraded equipment or transient combustible) on the fire watch area tag. For the tags in the two cable spreading rooms the description block contained "Appendix R Concerns".

The inspector brought these issues to licensee management's attention. In their response management stated that the watches were satisfying their expectations for a hourly roving fire watch, because the ability to detect a fire in the room could be accomplished by either visual sighting of the fire or smelling the smoke while at the entrance at the door. They did agree that the description on the fire watch area tag could be improved to better describe the fire concern and the areas that required additional attention.

The inspector did not dispute the ability of a RFW to detect a fire, however, the fire detection system (smoke detectors) should also be capable of detecting a fire that was noticeable from the entrance of the room. The basis for a roving fire watch is to compensate for some degraded condition of the fire protection system. This compensation should include early detection of a fire (i.e., smoldering or hot items) and inspections for excessive combustible material or other unexpected conditions. The inspector does not believe the hourly RFW could provide these types of compensatory measure based on the length of time they were in the room and the large size of the rooms.

The licensee provided the RFWs with additional verbal guidance for the hourly patrols. Currently the RFWs are walking through the areas and spending more time during their patrols. The inspector verified this in discussions with RFWs and a review of recent security access records for the cable spreading rooms. Also the licensee informed the inspector that they were reviewing the instructions in FPP-004 for needed changes to ensure fire watches fully understand management's expectations.

As implemented, the hourly RFW did not appear to provide the measures needed to compensate for the failure to meet the cable separation criteria required by 10 CFR 50, Appendix R, or as described in the V. C. Summer Fire Protection Evaluation Report. This is identified as unresolved item 50-395/94-07-04, pending further review by the inspector.

b. Discrepancies in the Fire Service (FS) System

A low flow condition in the FS preaction sprinkler system for the auxiliary building, elevation 463, was discussed in NRC Inspection Report No. 50-395/94-03. To temporarily resolve this problem the licensee planned to changeout the sprinkler heads. During this activity, the licensee identified 19 additional sprinklers which were not shown on the FS drawing. This delayed the temporary resolution to the low flow condition. The additional sprinklers were added during the original installation of the FS system; however, an error in the drawing control process resulted in the FS drawing not being updated.

In response to this problem, the licensee performed a walkdown of the fire suppression system in the control building (CB) elevation 448 and the intermediate building (IB) elevation 412. In the CB, a cable tray drop containing three sprinkler heads was identified as missing. This item was shown on the applicable FS drawing. Engineering evaluated this condition and accepted it on a temporary basis. For the IB, the installed configuration was in agreement with the FS drawing. However, the licensee identified 14 sprinklers that were installed in the plant and shown on the drawing, but had not been included in the hydraulic flow calculation for that portion of the building. This was also accepted by engineering on a temporary basis.

Based on the three system walkdowns, which identified three separate problems, the inspector was concerned that the design configuration for the FS suppression system was not installed or properly maintained. The licensee informed the inspector that they are continuing to evaluate the three identified problems and have plans to perform an expanded review of the FS suppression system. The inspector will continue to review the licensee's actions in this area. This is identified as IFI 50-395/94-07-05, Correction of fire protection problems.

7. Design Changes and Modifications (37700)

On January 28, 1994, the licensee identified a condition that was outside the design basis of the plant. A one-hour notification as required by 10 CFR 50.72 was made at that time. The nonconforming condition involved the discharge pressures of the "A" and "B" centrifugal charging/high head safety injection pumps exceeding the applicable system design pressures. The design data for these systems that is provided on the piping and instrumentation drawings (P&IDs) and in Gilbert Design Specifications stipulates a "normal" pressure of 2500 psig and an "upset" pressure of 2735 psig. Normal conditions are defined as those that exist for the majority of the design life, including commonly experienced transients. While upset conditions include deviations from normal conditions that are anticipated to occur, often enough that the design should include the capability to withstand the conditions without operational impairment.

A review of recent operator logs indicated that the discharge pressure from "B" pump varies from 2775 to 2800 psig while "A" pump has a range from 2740 to 2750 psig. The "C" pump discharge pressure is less than the "upset" design pressure (2735 PSIG). The above pressure ranges were taken with normal pump flow (approximately 170 GPM). The highest pressures (approximately 2820 GPM) occurred during pump testing when the discharge valve was closed and only minimum recirculation flow was available. During the last refueling outage (Spring 1993) the rotating element was changed out on "B" pump to provide more consistent performance between the three pumps. When reviewing the test data (following the work on "B" pump), engineering noted the high discharge pressure; however, a concern was not raised since the DBD specified a design pressure of 3100 psig. The licensee plans to reconcile the difference between the DBD pressure and the Gilbert specification.

The issue of higher than expected discharge pressure had been partially addressed in 1983. Unexpected lifting of the relief valve for the boron injection tank (BIT) caused the licensee to evaluate the charging pump's high discharge pressure. A resulting action was the changeout of the rotating element for the "B" pump to lower it's discharge pressure. The licensee informed the inspector that some evaluations had been performed at that time to justify the high discharge pressures. The inspector reviewed NCNs 1473 and 1557 which provided the engineering resolution for this problem in the 1983/1984 time period. While the effects of the high pressure were addressed, the inspector did not see any indication that the documentation conflict between the design basis and actual pressure was ever addressed at that time. In 1990 the issue of high discharge pressure was again questioned by the licensee. The architecture/engineer, Gilbert provided the evaluation which accepted the operating pressures being higher than the design pressures. In one of the responses Gilbert recommended that the design documentation be revised.

The licensee again reviewed the design documentation for individual components and piping in the applicable portions of the systems and determined that the ASME code allowable stresses had not been exceeded. The differential pressure (DP) calculations for motor operated valves (MOVs) were reviewed in response to this condition since the maximum pressures used in the calculations did not reflect actual worst case conditions. The results of the licensee's review indicated that the most limiting system pressure for these MOVs was 2850 psig which is based on the capability of the MOVs to operate under maximum DP conditions. To prevent exceeding this limitation the maximum operating pressure of the volume control tank (VCT), which is the normal pump suction pressure, was reduced from 65 psig to 45 psig. To allow a margin for testing the pumps, a limiting pressure of 2900 psig was established. This was based on the MOVs not being subjected to discharge pressure during testing and the pumps vendor specifying a maximum allowable pressure of 2950 psig. The licensee is also reviewing the hydrostatic testing requirements to determine if the original hydrostatic test bounded the current operating parameters of these systems.

The failure to ensure that the design basis of the plant encompassed the actual operating parameters of the plant is identified as a Violation 395/94-07-03. The failure to recognize that the plant was operating outside the design basis resulted in nonconservative assumptions being used in MOV DP calculations and operating limitations (i.e. VCT pressure) not being implemented. In addition, the original hydrostatic test of these systems may not have satisfied ASME code requirements for the operating modes of the plant.

8. Review of Licensee Self-Assessment Capability (40500)

The inspector attended a meeting in NRC's Atlanta office with licensee management personnel and regional NRC personnel on February 17, 1994. The meeting, held at the licensee's request, to discuss current plant issues. Discussion topics included degraded fire suppression system flows, inadequate separation to meet Appendix R requirements, and charging system operating pressures greater than design specifications. New information conveyed by the licensee included their plans to conduct an independent evaluation of several areas highlighted from their reviews and their plans to implement a new employees concerns program. The meeting was viewed as beneficial based on the additional explanation of these plant issues and a better understanding of the licensee's resulting actions.

On February 24, 1994, the inspector attended a scheduled meeting of the Plant Safety Review Committee (PSRC). The meeting was held to discuss routine items which require PSRC review, (i.e., procedure changes, proposed TS changes), close out reviews of nonconformance notices, etc. Appropriate items were discussed in detail and several items were either deferred for additional action or approved based upon completion of a specific action item.

Both meetings demonstrated a strong management commitment to fully understand and properly resolve issues/problems at the plant.

9. Review of Maintenance Training (41500)

The inspector performed an initial review of the maintenance/craft training program. Both the initial qualification program and continuing training program were reviewed. The licensee was in the process of changing the qualification program to a new format that utilized the "Taskmaster" concept. In this process, individual tasks that a discipline would be responsible for completing are identified. These tasks are then broken down into required skills and knowledge needed to perform the task. The licensee then compared the different skills and knowledge to the previous training objectives that were taught to ensure personnel were properly trained for the task. Portions of "Taskmaster" have already been implemented; however, the actual change to "Taskmaster" for tracking all task/job qualifications is an ongoing process with completion planned for 1995.

The previous method for tracking job qualification used a matrix format. The different skills (both general and job specific) were listed on the matrix along with the individuals from each discipline. Individuals that were qualified for the different skills were annotated on the matrix.

The inspector interviewed supervisors from the different disciplines to measure their understanding of the qualification program and how the program was implemented. All the supervisors were aware of the qualification matrix and all but one supervisor had a copy of the matrix readily available. The matrix appeared to be used on an informal basis. The supervisors relied on their knowledge and understanding of an individual's skills when making work assignments verses checking the matrix to verify that an individual was qualified for a particular skill. The inspector was informed that "journeymen" personnel are qualified for all the basis and routine skills that have been identified for a discipline. Based on the inspector's observations of maintenance activities, there does not appear to be a need to restrict usage of the qualification matrix.

The inspector reviewed the continuing training schedule for 1994. The class topics covered many areas that relate to recent plant problems and upcoming areas requiring additional maintenance involvement.

The licensee has a structured maintenance training program that appears to be effective in meeting the needs of the plant. Efforts are underway to update the program using a more systematic approach. The inspector did not have an opportunity to review the actual qualification process for individuals. Additional inspector reviews will be performed in this area.

10. Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities (92700)

(Closed) LER 93-01, Reactor Trip Due to High Positive Rate.

Due to a relay actuation causing an oil circuit breaker (OCB 8902) in the switchyard to open, the load on the main generator was instantaneously reduced from full load to only balance of plant (BOP) loads. As a result of this load reduction, the frequency on the BOP buses increased 60 Hz to approximately 62.5 Hz. The reactor coolant pumps, which are powered from the BOP buses, increased their speed in response to the frequency increase. The resulting increase in reactor coolant flow caused an increase in core water density (temperature). This, combined with the large negative moderator temperature coefficient at end of life, caused a neutron flux increase that initiated a reactor trip occurred.

The cause of the relay actuation was investigated but could not be determined. The relay, along with others that could have been at fault, were replaced. In addition, the licensee generated a new Abnormal Operating Procedure, AOP 304.3, Loss of All Balance of Plant Buses. The inspector reviewed this procedure and concluded it was adequate.

(Closed) LER 93-02 and LER 93-02, Revision 1, Steam Generator Tube Eddy Current Test Results

Results from the sixth inservice eddy current examination indicated that more than one percent of the inspected tubes were defective. This met the C-3 inspection category of TS for expanding the inspection scope to 100 percent. Tube degradation was localized in the tubesheet area and was the result of primary water stress corrosion cracking. In addition, a number of tubes experienced outer diameter stress corrosion cracking in the tube support and flow distribution baffle plate areas. Three tubes which had the most degradation were pulled to investigate the degradation mechanism in the flow distribution baffle plate area. Additional controls were implemented that included lower tube leakage limits, administrative controls on pressurizer power operated relief valves (PORVs) availability, additional training relating to tube failures, and improvements in S/G leakage monitoring program. current plans are to replace the S/G's during the upcoming refueling outage (Fall, 1994).

11. Action on Previous Inspection Findings (92701, 92702)

(Closed) Inspector Followup Item, 92-23-02, Testing the Turbine Driven Emergency Feedwater Pump (TDEFP) Low Lube Oil (LO) Trip Setpoint

When reviewing scheduled maintenance for the TDEFP, the inspector noted that the planned testing would not fully verify the low lube oil trip setpoint. The low lube oil trip is provided to protect the turbine from extended operation at low speeds which when operated for a period of time could damage the turbine. Maintenance activities were planned for the oil cylinder which initiates the low lube oil trip, however, the specified testing would not operate the turbine at the low speeds required to obtain a trip. The inspector was concerned that the trip setpoint could change to a higher speed that was within the operational range of the TDEFP and post maintenance testing would not identify this condition.

In response, the licensee revised the TDEFP maintenance procedure, MMP 300.015, to verify the low lube oil trip setpoint after maintenance is performed on the tripping mechanism. A five year PM activity to clean, inspect, and repair the trip mechanism has been added to the licensee's PM database. This will result in the low lube oil trip setpoint being verified a minimum of every five years. During the last refueling outage (Spring, 1993) testing of the TDEFP verified that the low LO trip setpoint was correct.

12. Exit Interview (30703)

The results were summarized on March 2, 1994, with those individuals identified by an asterisk in Paragraph 1. The following summary of inspection activity was discussed by the inspectors during this exit:

Type	Item Number	Status	Description
NCV	94-07-01	Closed	Failure to comply with procedural requirements for inoperable fire door.
IFI	94-07-02	Open	Main steam isolation valve maintenance and testing.
NOV	94-07-03	Open	Failure to ensure the design basis encompasses actual plant operating conditions.
URI	94-07-04	Open	Adequacy of Roving Fire Watch Patrols
IFI	94-07-05	Open	Correction of Fire Protection Problems Involving low flow to sprinklers and missing sprinklers. Extra sprinklers not included in the flow calculation.

IFI	92-23-02	Closed	Review of procedure change to test/verify the turbine driven emergency feedwater pump low lube oil trip setpoint.
LER	93-01	Closed	Reactor Trip Due to High Positive Rate
LER	93-02	Closed	S/G Tube Eddy Current Results - Category C-3 Applied
LER	93-02, Rev. 1	Closed	S/G Tube Eddy Current Results - Category C-3 Applied

Proprietary information is not contained in this report. The licensee expressed their disagreement with the inspector's comments on the adequacy of the originally proposed post maintenance testing for the EDG day tank level switch. They stated that they considered the proposed testing was adequate to demonstrate operability.

Subsequent to the exit interview, the licensee was informed by the Senior Resident Inspector that the adequacy of the fire watch patrols was designated as an unresolved item and fire protection problems were identified as an inspector followup item.

13. Acronyms and Initialisms

AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
BIT	Boron Injection Tank
BOP	Balance of Plant
CB	Control Building
CMC	Coordinated Manual Control (Honeywell Switch)
CREP	Control Room Evacuation Panel
DBD	Design Basis Document
DP	Differential Pressure
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
F	Fahrenheit
FPO	Fire Protection Officer
FPP	Fire Protection Procedure
FS	Fire Service
GPM	Gallons Per Minute
HVAC	Heating, Ventilation and Air Conditioning
IB	Intermediate Building
I&C	Instrumentation and Control
ICP	Instrumentation Control Procedure
IFI	Inspection Followup Item
LER	Licensee Event Report
LO	Lube Oil
MCB	Main Control Board
MMP	Mechanical Maintenance Procedure
MOV	Motor Operated Valve

MSIV	Main Steam Isolation Valve
MWR	Maintenance Work Request
NCN	Nonconformance Notice
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OCB	Oil Circuit Breaker
P&ID	Piping and Instrumentation Drawing
PM	Preventive Maintenance
PMTS	Preventive Maintenance Task Sheet
PORV	Power Operated Relief Valve
PSIG	Pounds Per Square Inch Gauge
PSRC	Plant Safety Review Committee
RCS	Reactor Coolant System
RFW	Roving Fire Watch
RWP	Radiation Work Permit
SAP	Station Administrative Procedure
S/G	Steam Generator
SPR	Special Report
STP	Surveillance Test Procedure
ΔT	Differential Temperature
TDEFP	Turbine Driven Feedwater Pump
TS	Technical Specification
VCT	Volume Control Tank

ENCLOSURE 3
FIGURE 1

CABLE SREADING ROOM
ELEVATION 448