UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report No.: 50-395/84-38

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

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

Facility Name: V. C. Summer

Inspection Conducted: December 7-11 and 17-21, 1984

Inspector: Frank Jape 1/30/85
for G. M. Nejfelt Date Signed

Approved by: Frank Jape 1/30/85
F. Jape, Section Chief Date Signed
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope: This routine, unannounced inspection involved 53 inspector-hours on site in the areas of reviewing precritical and zero power testing (ZPT) and witnessing ZPTs.

Results: No violations or deviations were identified.

REPORT DETAILS

1. Licensee Employees Contacted

   *R. Rouknight, Technical Specialist, Regulatory Compliance
   *O. S. Bradham, Director, Nuclear Plant Operations
   *M. N. Browne, Manager, Technical Support
   *R. M. Campbell, Engineer, Independent Safety Group
   *J. G. Connelly, Jr., Deputy Director, Operations and Maintenance
   *B. G. Croley, Group Manager, Technical and Support Services
   *B. T. Devendorf, Shift Technical Advisor
   W. O. Dixon, Vice President, Nuclear Operations
   J. W. Halthwanger, Reactor Engineer
   *M. D. Irwin, Nuclear Licensing Specialist
   *G. A. Loignon, Jr., Associate Manager Plant Performance and Results
   P. A. Pickens, Maintenance Supervisor, Instrumentation and Control
   *G. G. Putt, Manager, Scheduling and Material Management

   Other licensee employees contacted included three shift supervisors, a number of operators and office personnel.

   Other Organization

   F. Langford, Westinghouse, Startup Representative
   L. Woolridge, Westinghouse, Site Representative

   NRC Resident Inspector

   C. Hehl, Senior Resident Inspector

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on December 21, 1984, with those persons indicated in paragraph 1 above. The licensee acknowledged the inspection findings without significant comment.

3. Licensee Action on Previous Enforcement Matters

   (Closed) Violation, 395/83-27-08, Use of Non-controlled Drawings For Plant Conditions. Controlled drawings were available for plant evolutions in the control room and no instance of non-controlled drawing usage was noted by the inspector. The inspector verified that the licensee's actions as stated in their response, dated November 4, 1985 were completed.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

5. Independent Inspection Effort (92706)

a. P-9 Interlock Modification

The P-9 interlock function, which is to prevent a direct reactor trip as a result of a turbine trip with the plant below 50 percent power, was tested satisfactorily on December 18, 1984. During the post-modification testing, witnessed and reviewed by the inspector, the circuit used to provide indication of the P-9 interlock was found by the licensee to be incorrectly wired. Also, the control room steam dump actuate indication and armed indication labels were discovered by the licensee to be reversed. This discrepancy caused some confusion, when performing the P-9 interlock post-modification test.

No violation was identified, because the post-maintenance testing detected the potential problems and prompt corrective actions were taken by the licensee.

b. Westinghouse Vantage-5 Radial Peaking Factor

The Westinghouse Vantage-5 fuel assemblies for cycle 2 were found to be acceptable for use in peripheral locations of the core per NRR's letter of December 11, 1984. Westinghouse had limited the heat flux hot channel factor, $F_Q(z)$ to 2.28, rather than 2.32 times the normalized axial peaking factor allowed by Technical Specification (TS) 3.3.2 at full rated power, because of the lower power generation in the peripheral regions where the Vantage-5 fuel assemblies (FAs) were installed. The reduction of $F_Q(z)$ from 2.32 to 2.28 occurred, because coolant flow was diverted from the Vantage-5 FAs containing additional flow baffles to adjacent FAs.

Further comments, concerning the Vantage-5 FAs, were reported in Inspection Report 50-395/84-34.

c. Alternate Dilute Mode

(Open) Inspector Followup Item (395/82-52-03), concerning the normal method of operating in the alternate dilute mode resulted in diluting the volume control tank (VCT) more than the reactor coolant system (RCS) was reviewed during the dilution to achieve criticality on December 18, 1984. The boron concentration in the VCT, pressurizer, and RCS at the time of initial criticality were respectively 1306 ppm, 1319 ppm and 1333 ppm; and were within the 50 ppm TS range limit throughout the dilution. Because of the gradual dilution to achieve criticality (i.e., change of 100 ppm boron concentration in approximately four hours), as discussed in paragraph 6.b, this Inspector Followup Item could not adequately be assessed and it remains open.

d. Boron Concentration Measurement

The method used to determine boron concentration was reviewed and witnessed, using Mannitol to complex the boric acid secondary and tertiary hydronium ions to permit automatic potentiometeric titration with sodium hydroxide. No procedural discrepancy was noted during the sampling or analysis. The actual time required to perform the sampling and analysis was within the 20 minutes specified by reactor startup procedures REP-107-003, to obtain a boron concentration measurement.

In the areas inspected, no violation or deviation was noted.

6. Startup Tests (72700)

The general guidance for the conduct of plant refueling startup test program, REP-107.001, Revision 0, was reviewed and found to be satisfactory. Specific comments of sections reviewed and witnessed are given below:

a. Precritical Test Items

(1) Results of reactor engineering procedure (REP) 107.011, Incore Thermocouple Normalization, Revision 0, to calculate isothermal corrections for the incore thermocouples with the reactor coolant system at hot, no-load conditions were reviewed. An acceptable 3.5°F difference, between test indication and expected indication, was obtained. Reactor thermocouple and resistance temperature detector cross calibrations were performed during the initial refueling and were not required prior to the refueling outage startup. Incore thermocouple wires, which were replaced during the refueling outage, were satisfactorily requalified.

(2) Documentation of REP-107.010, Shutdown and Control Rod Drop Test, Revision 0, for cold and hot tests was reviewed using the visicorder charts obtained.

During both the cold and hot rod drop tests, all control banks and shutdown bank rods were within the required time limit of Technical Specification 3.1.3.4.

The summing circuit used to graphically present the A and B coils combined output for each rod, as the fields collapsed when the reactor trip breaker was opened, provided abnormal curves for the following rods tested cold: E11, F10, and F12 (twice). However, during the subsequent hot rod drop test, no summing circuit abnormalities were observed. Also, two administrative discrepancies were noted with REP-107.010, Revision 0, for licensee's review. First, this procedure referenced Final Safety Evaluation Report, Table 14.1-55, which was applicable only in preparation for initial criticality to test the slowest and fastest rod ten times each at no flow and cold conditions. Cold and hot rod testing for the slowest and fastest rods were performed only once. Secondly, Section 7.32 omitted the following information stamped on visicorder charts: bank and group; procedure number and step; and signature. The licensee will review these items and revise them appropriately.

b. Reactor Startup

On December 18, 1984, the reactor was restarted following a refueling outage. The inspector witnessed the startup and noted following difficulties:

(1) The initial attempt to restart was delayed when attempts to withdraw the first shutdown bank were unsuccessful, because the reactor trip breakers had tripped opened during the performance of STP-302.038, Revision 5, NIS Power Range (N41) Operational Test. With the RCS borated to provide the required shutdown margin and all rods fully inserted during the performance of STP-302.038, no safety significance was associated with this oversight. The administrative controls used to perform the surveillance were reviewed and determined to be adequate in accordance with Station Administrative Procedure (SAP)-134, Revision 2.

(2) The estimated critical conditions resulted in a +62 ppm boron concentration error from the actual conditions needed to obtain criticality. Westinghouse was contacted and provided revised estimated critical conditions using a "2DXY" model analyzing individual rods rather than the "3 Nodel D" model previously used, which analyzed the core by areas. A +3 ppm boron concentration error from the actual boron concentration needed to obtain criticality resulted using the revised calculations. No reactor anomaly report was required, since the boron concentration error was negligible. A report is required only when the error is greater than 100 ppm.

c. Zero Power Testing

The following zero power tests were witnessed and reviewed:

(1) Reactor Engineering Procedure (REP)-107.008, Revision 0, Boron Endpoint Measurement. This test is used to determine the critical RCS boron concentration at hot zero power. A value of 1310 ppm was obtained and since an error of greater than 50 ppm existed between the actual and estimated conditions, the licensee's procedural level II acceptance criteria were exceeded and resolved, as described in paragraph 6.b.

(2) STP-210.002, Revision 0, Isothermal Temperature Coefficient. The moderate temperature coefficient (MTC) from isothermal data provided a -2.58 pcm/°F MTC. The predicted MTC value of -2.69 pcm/°F was in good agreement with measured value.

(3) REP-107.007, Revision 0, Reactivity Computer Checkout and Operation. This test is used to calibrate the parameter and reactivity on-line monitoring system. The test was performed satisfactorily.

(4) REP-107.009, Revision 0, Low Power Flux Mapping. A full-core flux map at low power was obtained utilizing this test. Sufficient information to map the core was provided with detectors A and B inoperable. Detector A was found to be inoperable because of a broken lead to the flux mapping cabinet. Detector B became inoperable, because of moisture leakage into the detector wiring with the plant at operating temperatures and pressures. The maximum radial peaking factors obtained from the maps were 1.60 at <8.2' and 1.56 at ≥ 8.2' were within the TS limits of respectively 1.63 and 1.74. The distances along the FAs were measured from the bottom of each FA.

(5) REP-103.001, Revision 1, Control Rod Worth Measurements. The reactivity worths of the control and shutdown banks were determined to be consistent with nuclear design predictions. The most reactive predicted control bank, bank B, was measured within 40 pcm of its designed integral worth given in the "Cycle 2, Nuclear Design Report," WCAP-10663, and was used as the referenced bank. The remaining control and shutdown banks reactivity worths were within approximately four steps of the predicted values.

Within the areas inspected, no violation or deviation was noted.