



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

Report No.: 50-425/89-16

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

Docket Nos.: 50-425

License No.: NPF-79

Facility Name: Vogtle 2

Inspection Conducted: March 27-31 and April 10-13, 1989

Inspector: *P. T. Burnett* 5-4-89  
P. T. Burnett Date Signed

Inspector: *P. A. Taylor* 5-4-89  
P. A. Taylor Date Signed

Inspector: *J. Zeiler* 5-4-89  
J. Zeiler Date Signed

Approved by: *G. A. Belisle* 5-4-89  
G. A. Belisle, Chief Date Signed  
Test Programs Section  
Engineering Branch  
Division of Reactor Safety

SUMMARY

Scope: This routine, unannounced inspection addressed the areas of reviewing completed, precritical tests; witnessing initial criticality; witnessing and evaluating postcritical, low-power physics tests; and witnessing the shutdown from outside the control room test and the loss of offsite power test.

Results: Initial criticality of Vogtle 2 was attained in a well-controlled manner. However, two minor discrepancies involving a lack of attention to procedural details were observed, paragraph 3. Agreement between predicted and actual critical configurations was acceptable.

The zero power physics tests, conducted after initial criticality, were performed with care and adequately verified the basic design characteristics of the reactor core. Test results appeared valid and met appropriate test acceptance criteria.

The loss of offsite power and shutdown from outside the control room tests were performed in accordance with approved procedures and met the requirements of the procedures, FSAR, and technical specifications. The plant responded as designed and good licensee performance was observed.

No violations or deviations were identified.

## REPORT DETAILS

### 1. Persons Contacted

#### Licensee Employees

- #M. J. Ajluni, Operations Superintendent
- #\*G. B. Dockhold, Jr., General Manager, Vogtle Nuclear Operations
- W. L. Burmeister, Operations Superintendent
- C. L. Christiansen, Shift Supervisor
- \*G. R. Frederick, Quality Assurance Site Manager - Operations
- \*W. C. Gabbard, Senior Regulatory Specialist
- H. M. Handfinger, Maintenance Manager
- T. S. Hargis, On-shift Operations Supervisor
- #\*W. F. Kitchens, Assistant General Manager, Operations and Maintenance
- \*C. F. Meyer, Operations Superintendent
- #\*A. L. Mosbaugh, Plant Support Manager
- #W. T. Nicklin, Regulatory Compliance Supervisor
- #\*R. M. Odom, Plant Engineering Supervisor
- #\*J. E. Swartzwelder, Operations Manager

Other licensee employees contacted included engineers, technicians, operators, and office personnel.

#### Other Organizations

- C. B. Holland, Westinghouse
- W. C. Phoenix, Consul Tec
- O. D. Hall, Consul Tec

#### NRC Resident Inspectors

- #\*R. F. Aiello, Resident Inspector
- #J. F. Rogge, Senior Resident Inspector, Operations
- C. W. Burger, Resident Inspector

- \*Attended exit interview March 31, 1989
- #Attended exit interview April 13, 1989

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

### 2. Precritical Test Procedure Review (72596)

The completed precritical test procedures listed below were reviewed by the inspectors following acceptance by plant management:

- a. 2-5SF-04 (Revision 1), Rod Drop Time, was completed on March 25, 1989, at plant conditions of 556°F, 2235 psig, and 109% flow. All 53 control rods dropped in less than the 2.2 seconds to dash pot entry specified in TS 3.1.3.4. One rod was found to have a drop time longer than the mean of the 53 plus two standard deviations. It was dropped an additional six times with acceptable results, good consistency. The inspector independently confirmed that none of the rods were within three standard deviations of the mean of the limiting value.
- b. 2-5EB-01 (Revision 0), RCS Final Leak Rate, was completed on March 22, 1989, by application of the surveillance procedure 14905-2 (Revision 1), RCS Leakage Calculation (Inventory Balance). Over a two-hour test, the gross leakage was 0.160 gpm, the identified leakage 0.124 gpm, and the unidentified leakage 0.026 gpm, which was well below the TS 3.4.6.2 limit of 1.0 gpm.
- c. 2-5BB-06 (Revision 0), Reactor Coolant Flow Coastdown, was completed on March 26, 1989. The low flow trip delay time of 0.963 second, which included a conservative value of gripper delay time of 0.150 seconds, was acceptably less than the 1.0 second used in the safety analyses. The flow coastdown rate was within the safety limit; since the time constant of 13.5 seconds was greater than the limiting value of 11.8 seconds. The test met the description and satisfied the acceptance criteria of FSAR 14.2.8.2.3.
- d. 2-5BB-02 (Revision 1), Pressurizer Heater and Spray Capability and Continuous Spray Flow Verification, was completed on March 27, 1989. The pressurizer pressure response to opening of both pressurizer spray valves was within the allowable range, as was the pressure response to activation of all pressurizer heaters. The pressurizer power operated relief valves opened in two seconds or less. The test met the description and satisfied the acceptance criteria of FSAR 14.2.8.2.2.

No violations or deviations were identified.

### 3. Initial Criticality Witnessing (72592)

Vogtle Unit 2, Cycle 1 achieved initial criticality at 1342 (EST) on March 29, 1989. The approach to initial criticality was performed using test procedure 2-600-02, Initial Criticality, and under the direction of test procedure 2-600-04, Initial Criticality and Low Power Physics Test Sequence. Procedure 2-600-04 defined the sequence of tests and operations, beginning with initial criticality, which constituted Unit 2 low power testing program. Initial criticality activities began with the plant at normal operating temperatures and pressures with RCS boron concentration greater than 2000 ppm. Criticality was initiated by withdrawing shutdown banks A through E in increments of 114 steps. Control banks A through D were withdrawn in 50 step increments until bank D was at 160 steps. Criticality was then achieved by diluting the

reactor to criticality. At each rod withdrawal increment and during the dilution process, the licensee performed ICRR calculations to monitor reactivity changes. Also, SRNI count rate data was taken to determine proper statistical functioning of the detectors.

During the dilution process the inspectors made two observations of the licensee's lack of attention to detail. The first observation involved a test procedure requirement to plot measured boron concentration versus ICRR, once the boron dilution process began. Although ICRR plots against times and dilution were maintained, this plot is the best prediction of criticality, particularly if dilution is interrupted or the rate changed. Soon after the dilution process started, the inspectors observed that the plot had not been initiated. This was brought to the attention of the test supervisor who then proceeded to plot the data. The second observation involved a procedural requirement to ensure that at least 1000 SRNI sampling counts are obtained to verify the statistical accuracy of the SRNIs. Below this minimum sampling value, less confidence is given to the accuracy of the statistical reliability tests for the SRNIs. At one point early in the dilution process, the inspectors observed that only approximately 700 counts were obtained in the ten second counting interval being used. This was brought to the attention of the test supervisor who immediately increased the sampling interval to obtain greater than the minimum number of counts required. Since these events occurred early in the dilution process and since timely corrective action was taken by the test supervisor, the inspectors determined that no significant safety problems occurred as a result of these events.

Criticality was approached in an orderly, controlled manner. The measured ARO boron concentration was 1340 ppm which was in acceptable agreement with the predicted value of 1322 ppm. The inspectors observed activities from the control room beginning with the withdrawal of shutdown bank C. Periodically, the inspectors independently verified the licensee's calculation of ICRR, the plotting of the points, and the extrapolation to criticality. Proper SRNI performance was independently verified by the inspectors throughout the approach to criticality by performing chi-squared statistical reliability tests. The inspectors also witnessed the collection and analysis of boron samples from the pressurizer and RCS. The technicians involved in the sampling appeared to be familiar with the requirements of the procedures in use.

The inspectors also monitored control room staffing levels and the conduct of the operations shift personnel during the initial criticality sequence. Control room manning was more than adequate to support required testing. The shift supervisor was effective in maintaining the proper control room atmosphere and controlling access to the area.

No violations or deviations were identified.

#### 4. Zero Power Physics Tests (61708, 61710)

Portions of the following tests were witnessed and the results were reviewed in part to ensure that test acceptance criteria were satisfied.

The inspectors will perform a final review of the test package after appropriate licensee review and approval of the results.

a. Boron Endpoint Measurement

Boron endpoint measurements were conducted on March 29-31, 1989, in accordance with Attachment 10.11, Boron Endpoint Determination, of procedure 2-600-04. Measurements were performed for the following control rod configurations: ARO; control bank D fully inserted; control banks D and C fully inserted; control banks D, C, and B fully inserted; and control banks D, C, B, and A fully inserted. The boron endpoints were measured by moving the rods as necessary to the desired positions. The resulting reactivity change was measured and converted to an equivalent boron concentration. This change in boron concentration was then algebraically added to the existing RCS boron concentration to obtain the boron endpoint. The measured ARO boron endpoint was 1344.5 ppm, which differed from the predicted value of 1329 by 15.5 ppm. This was well within the test acceptance criteria of  $\pm 50$  ppm. For each of the other rod configurations, agreement between the measured and predicted values was within the acceptance criterion of  $\pm 10\%$ .

b. Isothermal Temperature Coefficient Measurement

The ITC measurements were conducted on March 29-30, 1989, in accordance with Attachment 10.13, Isothermal Temperature Coefficient Measurement, to procedure 2-600-04. The measurements were performed by lowering and raising RCS temperature at a constant rate less than  $10$  °F/hr and using a reactivity computer which recorded reactivity and average RCS temperature versus time. ITC was determined from the change in reactivity resulting from a uniform change in moderator temperature. ITC measurements were performed at three different control rod configurations: ARO, control bank D fully inserted, and control banks D and C fully inserted. ITC measurements were well within the  $\pm 3$  pcm/°F acceptance criterion for the tests.

ITC is the sum of the moderator temperature coefficient and the Doppler coefficient; MTC was calculated by subtracting the Doppler coefficient from the measured ITC. At ARO, the average MTC was calculated as  $0.09$  pcm/°F. The test acceptance criterion and plant technical specifications require a negative MTC. The licensee initiated the appropriate action to establish control rod withdrawal limits to assure that the operating MTC would be negative.

c. Control Rod Worth Measurements

Control rod worth measurements were conducted on March 28-31, 1989, in accordance with Attachment 10.13, RCCA Bank Worth Measurement At Zero Power, to procedure 2-600-04. Control rod worth of each control bank was measured using a boron exchange technique and using a reactivity computer to measure reactivity. Each control bank was diluted in the

core starting from ARO and finishing with all rods fully inserted. The control banks were then borated out in overlap to 130 steps on control bank D. The reactivity worth for the remaining 98 steps to the ARO position was added to the overlap worth. The total overlap worth was then compared to the total control bank worth determined by individual control bank measurement. Differential and integral rod worths were calculated and plotted to compare with predicted data.

All individual control bank worths agreed with their predicted values to within the  $\pm 10\%$  acceptance criterion. The total measured integral rod worth of 3638.95 pcm compared favorably with the predicted value of 3520 pcm. The inspectors independently analyzed the reactivity traces obtained during the bank D and C measurements and compared the results with that calculated by the licensee. The comparison for bank C is shown graphically in the Attachment. The measured value for integral worth for bank C of 1208.9 pcm differed from the predicted by less than 3%, which was well within the acceptance criterion of 10%.

No violations or deviations were identified.

#### 5. Shutdown From Outside the Control Room Test (72583)

This test was performed on April 11, 1989, under the guidance of startup test procedure 2-600-08, Remote Shutdown Test, Revision 1, and AOP 18038-2, Operation From Remote Shutdown Panels, Revision 3. Prior to the start of the test, the inspectors conducted a walkdown of the three remote shutdown panels A, B, and C. The inspectors observed that appropriate access control to the panels was maintained; that appropriate revisions of AOP 18038-2 and other pertinent procedures were available at the panels; and housekeeping controls were being maintained at these locations.

The inspectors attended the pre-test briefing where the test sequence was discussed and test personnel were given final instructions. The duties and responsibilities of the shutdown and backup crews were discussed including a review of the permissible actions allowed by the backup crew. The importance for plant safety was emphasized and reviewed by the shift supervisor. The test termination criteria, under which plant control would be transferred back to the control room if unsafe conditions develop, were also reviewed.

Test personnel included three shutdown crew members located at remote shutdown panels A, B, and C. In addition, a shift supervisor controlled shutdown activities and plant operations from shutdown panel B. The backup crew members, located in the control room, were responsible for equipment not directly involved in the test and were available to reestablish immediate control to the control room should adverse conditions have occurred.

The reactor was tripped simultaneously from remote shutdown panel A and B at 1647 (EST) with reactor power at approximately 20%. All subsequent

actions were controlled and performed from the remote shutdown panels using AOP 18038-2 to stabilize the unit in hot standby. The unit was maintained in stable hot standby conditions for a minimum of 30 minutes. Plant control was returned to the control room after this time period.

The inspectors witnessed the test activities at remote shutdown panels A, B, and C and accompanied the outside operator assigned to panel C in performing various required actions in the plant. Plant systems responded as designed and no abnormal or adverse conditions were reported. A smooth and orderly transfer of control of the unit to the shutdown panels was accomplished. The licensee verified that reliable communication among all shutdown crew members was available prior to commencing any control transfer evolutions, and good communication between all shutdown panels and the control room was observed throughout the test. All test activities were well coordinated and all test criteria appeared to have been met. The inspectors will review the approved test package during a future inspection after the licensee has approved the test results.

Plant startup after the remote shutdown test was delayed due to an incorrect ECP calculation made by the licensee. Although the inspectors were not present during this startup, a meeting was held afterwards with plant operations management to discuss the events. The incorrect ECP was calculated as 84 steps on control bank D and 199 steps on control bank C. Control rods were withdrawn to 95 steps on Control bank C; at this point control room personnel noticed a substantial increase in SRNI count rates. Control rods were then withdrawn another 5 steps at which point count rates increased by over a factor of 10. It became clear to control room personnel that criticality was imminent and control rods were inserted. After reanalysis of the ECP calculation, it was discovered that an error in the sign notation for the power defect term had been made. Specifically, a negative value was used instead of a positive. The licensee attributed this sign error to the Unit 2 Plant Technical Data Book graph of power defect versus power level, which uses negative values.

This event was complicated by the fact that the ECP calculation was independently verified by a reactor engineering person and the shift supervisor, and both made the same error. The inspectors also concluded that better use of ICRR plots to predict criticality could have been made. Extrapolation of the ICRR versus rod withdrawal curve clearly indicated that criticality would have occurred before the calculated ECP control rod position.

The licensee discussed the following corrective actions to be taken as a result of the event.

- a. Change the sign notation (negative to positive) in the graph of power defect versus power level, located in the Unit 2 Plant Technical Data Book;



- b. Revise procedure 14940-1, Estimated Critical Condition Calculation, Revision 7, to include the absolute value of the reference reactivity used to calculate ECP;
- c. Revise procedure 12003-C, Reactor Startup, Revision 13, to ensure better use of ICRR plots in predicting criticality; and
- d. Provide more training to operations and reactor engineering personnel in calculating ECPs.

The inspectors determined that good corrective action was planned by the licensee. These actions were discussed with management at the exit meeting.

No violations or deviations were identified.

6. Loss of Offsite Power Test (72582)

The inspectors witnessed licensee performance of test procedure 2-600-09, Loss of Offsite Power At Greater than 10% Power, on April 13, 1989. This included attending the pre-test briefing where personnel involved in the test discussed the test sequence, test administrative restrictions, and were given final instructions by the on-shift operations supervisor and shift supervisor. The test began at approximately 1000 hours (EST) with the plant in stable condition and reactor power at approximately 15%. The loss of offsite power was initiated by tripping all seven lowside breakers supplying power to the unit from the auxiliary transformer. Following offsite power isolation, the unit responded as designed; safety-related buses shed load, the emergency diesel generators started, and safety-related loads sequenced on the diesel generators. Decay heat removal was accomplished by natural circulation which was verified by the inspectors by observing RCS hot and cold leg temperature differences. The unit was stabilized in hot standby conditions for approximately 30 minutes before offsite power was restored.

The inspectors observed test activities in the main control room and the technical support center. There was adequate coordination among personnel involved in the test, and their actions appeared to be correct and timely during the test performance. The plant response during test appeared to be satisfactory. The inspectors will review the approved test package during a future inspection after the licensee has given final approval of the test results.

No violations or deviations were identified.

7. Exit Interview (30703)

The inspection scope and findings were summarized on March 31 and April 13, 1989, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection findings. Dissenting comments were not received from the

licensee. Proprietary information was reviewed in the course of the inspection, but is not contained in this report.

#### 8. Acronyms and Initialisms

AOP - Abnormal Operating Procedure  
ARC - all-rods-out  
ECP - estimated critical position  
EST - Eastern Standard Time  
FSAR - Final Safety Analysis Report  
ICRR - inverse count rate ratio  
ITC - isothermal temperature coefficient  
MTC - moderator temperature coefficient  
pcm - percent millirho  
ppm - parts per million  
RCCA - rod cluster control assembly  
RCS - reactor coolant system  
SRNI - source range nuclear instrument  
TS - technical specifications

ATTACHMENT

ATTACHMENT

# VOGTLE 2, CYCLE 1, CONTROL BANK C

*Differential Worth*

—□— Inspector

—x— Licensee

