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

Report Nos.: 50-424/92-17 and 50-425/92-17

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

Docket Nos.: 50-424 and 50-425 License Nos.: NPF-68 and NPF-81

Facility Name: Vogtle 1 and 2

Inspection Conducted: July 27 - 31, 1992

Inspector: _____ 9-2-92
P. T. Burnett, Reactor Engineer Date Signed

Approved by: _____ 9/2/92
R. V. Crlenjak, Chief Date Signed
Operational Programs Section
Operations Branch
Division of Reactor Safety

SUMMARY

Scope: This routine unannounced inspection addressed the areas of review of completed Unit 2, cycle 3, startup tests and review of completed core surveillance tests for both units.

Results: In general, all procedures reviewed were performed acceptably. Examples of good practice in performing core performance tests and surveillance activities are identified in paragraphs 3, 4, and 7.

One violation for failure to properly implement requirements for reviews of temporary changes to procedures is discussed in paragraph 8.

No additional violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*H. Beacher, Senior Plant Engineer
*S. Bradley, Reactor Engineering Supervisor
W. Burmeister, Manager, Engineering Support
*C. Christiansen, Safety Audit and Engineering Group Supervisor
*G. Frederick, Manager, Maintenance
*D. Huyck, Nuclear Security Manager
*W. Kitchens, Assistant General Manager, Plant Support
*R. LeGrand, Operations Manager
*M. Sheibani, Nuclear Safety and Compliance Supervisor
*W. Shipman, General Manager, Nuclear Plant
*C. Tynan, Nuclear Procedures Supervisor

Other licensee employees contacted included engineers and office personnel.

Oglethorp Power Company Representative

*T. Mozingo

NRC Resident Inspectors

*D. Starkey, Resident Inspector

*Attended the exit interview on July 31, 1992.

Acronyms and initialisms used throughout this report are defined in the final paragraph.

2. Unit 2, Cycle 3 Precritical Activities (72700)

The inspector reviewed the following procedures completed prior to criticality for Unit 2, cycle 3:

a. 88003-C (Revision 2), Shutdown Margin by Minimum Bank Height, was completed on April 8, 1992, using data from the NDR. The requirements of TS 4.1.1.1.1.d (COLR) were satisfied.

b. 88006-C (Revision 3), Rod Drop Time Measurement, with Rod Drop Test Cart, was completed over the period of April 29 - 30, 1992. All four RCPs were running, and RCS was $\geq$ 551°F throughout the measurements. The requirements of TS 3/4.1.3.4.a were satisfied.

No violations or deviations were identified.

3. Unit 2, Cycle 3 Initial Criticality and Low-Power Tests (72700, 61708, 61710)

88002-C (Revision 3), Reload Low Power Physics Testing, was performed during the period February 24, 1992, to May 8, 1992. Activities accomplished under this procedure included initial criticality for the cycle, ARO CBC determination, ITC-MTC measurement, and rod worth measurement. A good feature of the approach to criticality included careful checkout of the SRNIs using a reliability factor test (analogous to the chi-squared test) to assure proper functioning of those instrument channels. The test was performed successfully for both channels prior to withdrawing control banks for ICRR measurements and prior to beginning dilution to criticality. The two SRNIs compared closely in ICRR throughout the approach to criticality.

The procedure specifies using alternate dilute mode with the VCT spray valve closed for the dilution process. These administrative controls preclude the VCT from becoming more dilute than the RCS. This is another good feature of the procedure. Over dilute VCTs have lead to reactivity overshoot during mixing following securing the active dilution process. In one PRA scenario, an over dilute VCT is one of the precursors to a severe reactivity accident.

Step 8.3.2 requires that NR-45 record either both SRNI channels or one SRNI and one IRNI during the approach to criticality. Only one pen of the recorder was operable; so the ERF computer was used to trend one of the instrument channels. Technically, that was probably a sound decision, but documentation within the procedure did not adequately describe the manner in which the computer display was used. Neither the time span displayed nor the sampling rate for the display were recorded. More importantly, no temporary change to the procedure was processed as required by TS 6.7.3 and plant administrative procedure 00052-C, Temporary Changes to Procedures. The compliance aspect of this issue is discussed in more detail in paragraph 8.

The procedure records indicate that initial criticality for cycle 3 was achieved in a well controlled manner at 0315 on May 7, 1992. Subsequently the ARO CBC was determined to be 2091 ppmB. That was 85 ppmB less than the predicted CBC. The acceptance criterion was +50 and -70 ppmB of prediction. The reactivity difference was calculated to be 634 pcm. Westinghouse reviewed the discrepancy and concluded that it did not invalidate the RSE; since all other HZP tests did satisfy the design criteria.

The DRC was checked out using internally generated exponential signals simulating both positive and negative reactivity inputs ranging from -28 pcm to +55 pcm. The dynamic test, using reactor generated periods was performed using only positive reactivities of 15 and 34 pcm. In all cases, measured and predicted reactivities agreed within one percent of the predicted value.

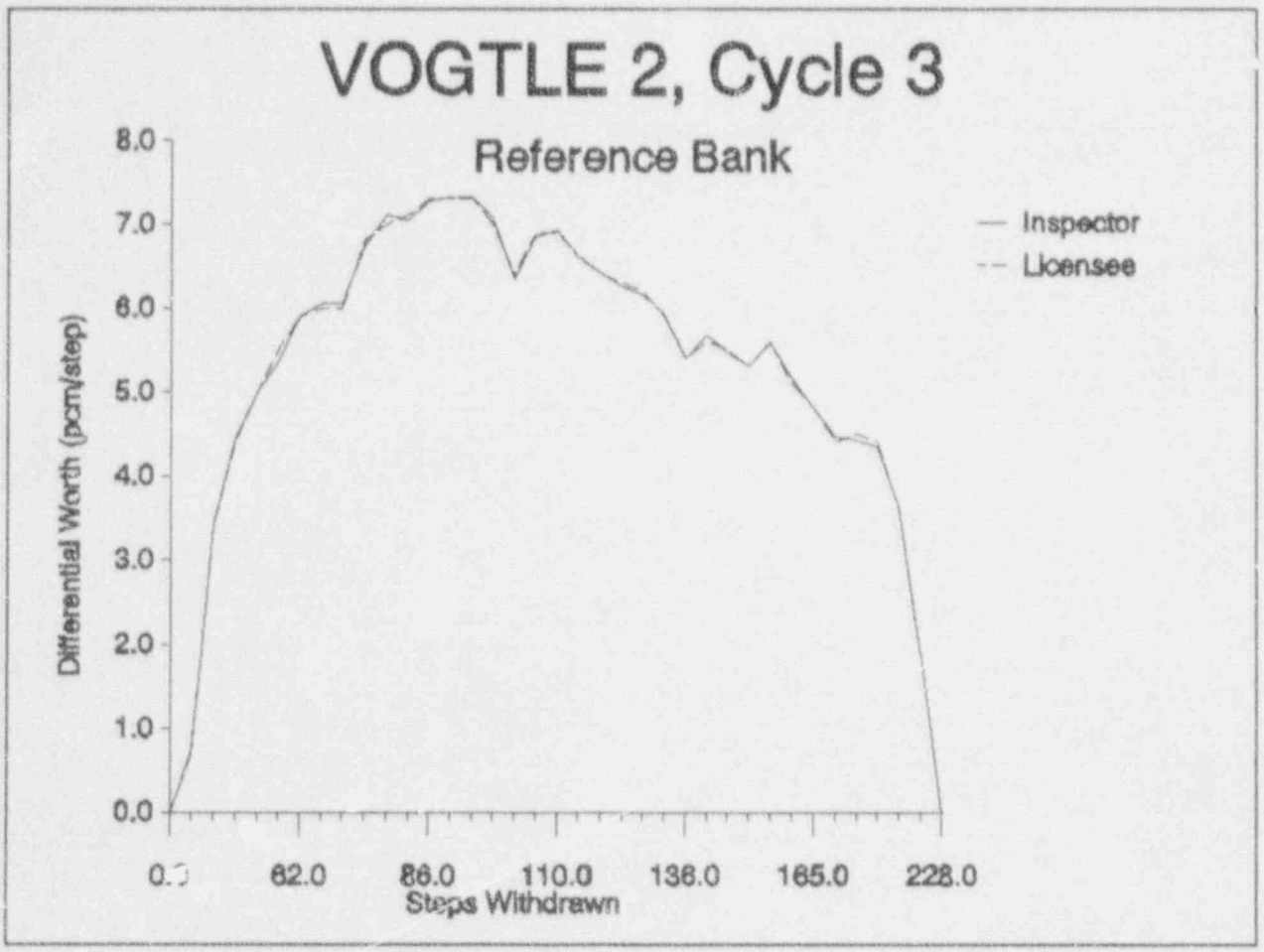
The ITC was measured at ARO for a 3°F heatup followed by a 3°F cooldown. The corresponding ITCs were 3.07 and 2.97 pcm/°F. Agreement within 1 pcm/°F indicates good control of the variables of the measurement, regardless of the relatively small temperature changes used in the measurements. Common practice is to use temperature changes of at least 4°F. The resulting MTC was 4.81 pcm/°F. The predicted values of MTC were 5.3 pcm/°F at BOC and 5.6 pcm/°F at 150 MWD/MTU into the cycle. Consequently, 0.3 pcm/°F was added to the measured MTC before comparing it with the TS 3.1.1.1 (COLR) limit of 7 pcm/°F.

Shutdown bank B, the calculated highest reactivity worth control rod bank, was designated the reference bank, and its reactivity worth was measured during boron dilution. The worth of each of the remaining control rod banks was determined using the rod swap technique with the reference bank. The results for all banks are given below.

	Reactivity worth (pcm)		
Bank	Predicted	Measured	Difference (%)
control A	268	209.2	-21.9
control B	794	820.4	+3.3
control C	778	748.1	-3.8
control D	502	466.8	-7.0
shutdown A	244	234.9	-3.7
shutdown B(REF)	957	996.0	+4.1
shutdown C	456	447.1	-2.0
shutdown D	451	438.8	-2.7
shutdown E	452	427 3	-5.5
TOTAL	4902	4788.6	-2.3

The review criterion for the reference bank worth and total bank worth was ±10% of prediction. For the remaining rod banks the criterion was the larger of ±15% or ±100 pcm of prediction. All criteria were satisfied.

The inspector independently analyzed the reactivity traces for the reference bank worth measurement using a mechanically different means of determining reactivity increments from that used by the licensee. The integral worth obtained from that analysis was 982 pcm, which is acceptable agreement for the method used. The differential reactivity curves are not significantly different, as can be seen from the figure below.



Beyond the violation discussed in detail in paragraph 8, no violations or deviations were identified.

4. Power Escalation Testing (72700, 61702, 61705)

Initial power escalation and testing, for cycle 3, was controlled and scheduled by 88019-C (Revision 3), Power Ascension after Refueling. This procedure also controlled the readjustment of the PRNI high flux trip high setpoint. The trips were increased to 70% prior to increasing to the 50% power plateau and to 95% prior to increasing power to 80%. The normal 109% trips were not established until testing at the 80% power plateau was completed. This conservative management of the high flux trip high setpoint is considered a good practice. The inspector reviewed the following completed tests:

a. 88014-C (Revision 4), Reactor Coolant System Flow Measurement, was performed on May 14, 1992, at 95% RTP. The flow was determined from equating heat balances across the primary and secondary systems, and the results are dependent upon reliable measurement of the tempera-

ture rise through the reactor vessel. Neglecting the uncertainties created by hotleg streaming, the results were acceptable. The greatest hotleg temperature differences observed during this test were about 4°F, which is less than the differences observed at similar facilities.

b. 88075-C (Revision 3), Precision Heat Balance, contains no acceptance criteria, but a step in the procedure requires that a work request for recalibration be issued if the feedwater flows measured on the plant flow meters are not within the span 0 to +2.5% of the precision instruments. That recalibration assures that the heat balance calculated by the plant computer and used for routine power level surveillance and calibration of the PRNIs is conservative with respect to a precision heat balance. Work request 26619 was issued resulting in MWO 29201692, dated May 14, 1992. All affected flow transmitters were recalibrated successfully by May 16, 1992.

c. 88007-2 (Revision 1), Limiting Hot Channel Factor Determination, was performed at nominal power levels of 30, 50, 75, 80, 95, and 100% RTP, during power escalation for cycle 3. The results at the lower power levels were extrapolated to the next power testing plateau before power increase was authorized. The licensee performed this procedure as a routine, 31-EFPD frequency surveillance test on July 27, 1992. The limits on F_Q and $F_{\Delta H}$ were satisfied in all cases for both the LOPAR and VANTAGE5 fuel assemblies.

d. 88023-C (Revision 4), One-Point Incore/Excore Detector Calibration, was performed at nominal power levels of 30, 75, 87, and 100% RTP during cycle 3 power escalation. The correlation fitting constants changed very little with power.

No violations or deviations were identified.

5. Nuclear Instrument Calibrations (61705)

88018-C (Revision 3), NIS Alignment for Refueling, was used to calculate preliminary IRNI and PRNI calibration data for the beginning of cycle for use until measured calibration data became available during power escalation. For each ion chamber, full power currents obtained during the last calibration of the previous cycle were adjusted by the ratio of predicted assembly power, for the current cycle, to measured assembly power, for selected assemblies adjacent to the chambers of interest.

The approach appeared reasonable, but the results were not conservative as demonstrated by the tabular information below.

IRNI Setpoints for Cycle 3

	N35(A)		N36(A)	
Function	Predicted	Measured	Predicted	Measured
rod stop(20%)	57.7	50.3	56.7	46.4
trip (25%)	72.1	62.9	70.9	57.9
max trip (31.1%)	89.1	78.2	88.1	72.1
full power	288.6	251.5	283.5	231.8

The predicted currents were calculated on March 17, 1992. The measured values were determined on May 11, 1992, from a least squares fit, constrained to pass through the origin, of currents measured at 30% and 50% test plateaus.

The predicted currents for the PRNIs were also in error in the non-conservative direction. However, pursuant to procedure 88019-C (discussed above), the high flux trip setpoint for the PRNIs had been adjusted to 50% prior to criticality. Hence, that trip was never greater than the LSSS. All PRNIs were recalibrated to the heat balance at the 30% power plateau. Heat balances conducted at higher powers yielded higher full power currents, which indicated that 30% power currents were conservatively under estimated. It does appear that one or more of the low setpoint high-flux trips for the PRNIs were greater that the LSSS early in the startup. Since those trips were recalibrated at the first opportunity to do so, and a good faith effort had been made to set the trips conservatively, that issue was not pursued further. A slightly more conservative prediction would have been made for the PRNIs if the calculation had used predicted power from the 0 MWD/MTU power distribution prediction vice the 150 MWD/MTU predicted distribution actually used. The licensee has not pursued the issue of the non-conservative estimates of full-power ion-chamber currents, nor has it tasked the fuel vendor, Westinghouse, to do so.

No violations or deviations were identified.

6. Measurement of Moderator Temperature Coefficient at the End of Cycle 2 for Unit 2 (61708)

88009-C (Revision 3), Moderator Temperature Coefficient Determination (EOL), was performed on November 26, 1991 at a nominal CBC of 300 ppmB, at full power. After corrections for changes in power defect, axial power redistribution, and xenon concentration (all small), the ITC measured during boration was -35.4 pcm/°F, and the ITC measured during dilution was -38.6 pcm/°F. The agreement between the two measurements is indicative of good measurement technique and good control of the process variables during the measurement. The resulting MTC was -36 pcm/°F, which compared well with the predicted MTC of -33 pcm/°F. The measured MTC was less negative than the TS limit.

No violations or deviations were identified.

7. Unit 1 Core Performance Surveillance Activities (61702, 61705)

The inspector reviewed the following Unit 1 surveillance procedures completed during the current operating cycle.

a. 88013-C (Revision 2), Overall Core Reactivity Balance, has been performed with at least 31-EFPD frequency throughout the cycle. The reactivity differences have ranged from -567 to +85 pcm, with no obvious trend toward the ± 1000 pcm limit.

b. 88016-C (Revision 0), Determination of RCS Delta T Power at 100% Rated Thermal Power, was first issued on February 25, 1992. The surveillance performed is required by TS 4.3.1.1 every 18 months. The licensee has opted to perform the surveillance quarterly in response to observed changes in radial power shape with burnup and concomitant changes in indicated hot leg temperatures because of hotleg streaming. The Delta T meters are rescaled when they differ from the heat balance by 1% or more. This is a good practice and initiative.

c. 88007-1 (Revision 1), Limiting Hot Channel Factor Determination, has been performed with 31-EFPD frequency throughout the current cycle. The limits on F_Q and $F_{\Delta H}$ were satisfied in all cases for both the LOPAR and VANTAGE5 fuel assemblies.

d. 88023-C (Revision 4), One-Point Incore/Excore Detector Calibration, has been performed at 31-EFPD intervals during the current cycle. Recalibrations were performed on two occasions in response to the surveillance observations.

A good feature of reactor engineering surveillance activities is that the parameters discussed above and others are routinely trended for both units.

No violations or deviations were identified.

8. Violation of Requirements to Review Temporary Changes to Procedures

All of the procedures reviewed during this inspection contained step 6.1.4, which is restated below.

6.1.4 If a procedural step cannot be completed successfully, clearly document the reason for not being able to complete the step on the "Comments and Observations" sheet before proceeding with the next step. Performance of the procedure may continue provided the omission of the step does not:

a. Affect the test results, or

b. Void the intent of the procedure, or

c. Result in the violation of a Technical Specification, or

d. Result in a reduction in the level of monitoring required by Technical Specifications (e.g. Special Test Exceptions).

If there is any doubt, consult the Reactor Engineering Supervisor, or terminate the test. If the test is to be terminated, immediately notify the Unit Shift Supervisor, if applicable. Document this on the "Comments and Observations" sheet and ensure that the appropriate restoration steps of Section 9.0 are completed.

Both TS 6.7.3 and administrative procedure 0052-C (Revision 7), Temporary Changes to Procedures, permit temporary changes to required procedures provided that the intent of the original procedure is not altered; the change is approved by two members of plant management, at least one of whom holds a Senior Operator license; and the change is documented and approved by the appropriate authority within 14 days of implementation. In the procedures reviewed in this inspection, step 6.1.4 gives performers of these procedures the latitude to make changes to procedures that are not pre-approved by two members of management (including one Senior Operator), or formally documented for review and approval by the appropriate authority.

Inclusion and implementation of step 6.1.4 in the procedures cited above has been identified as a violation of 10 CFR 50, Appendix B, Criterion V, and the licensee's accepted quality assurance program, Final Safety Analysis Report Section 17.2.5. (VIO 50-424 and 50-425/92-17-01: Step 6.1.4 in the 88xxx-x series of procedures violates requirements for review of temporary changes to procedures.)

9. Exit Interview

The inspection scope and findings were summarized on July 31, 1992, with those persons identified in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings. The licensee objected to the violation discussed in paragraph 8 and listed below. No additional dissenting comments were received from the licensee. Proprietary materials were provided to and reviewed by the inspector during this inspection, but were not incorporated into this report.

A telephone conference between Region II management and licensee representatives was held on September 2, 1992, to discuss the violation. The licensee agreed that procedure step 6.1.4 had the potential for violating requirements for temporary changes to procedures and described corrective action underway. After the conference, Region II management reconsidered and reaffirmed the violation.

The following item was discussed with the licensee.

VIO 50-424 and 50-425/92-17-01: Step 6.1.4 in the 88xxx-x series of procedures violates requirements for review of temporary changes to procedures - paragraph 8.

10. Acronyms and Initialisms Used in This Report

ARO	all rods out
BOC	beginning of cycle
CBC	critical boron concentration
COLR	Core Operating Limits Report
DRC	digital reactivity computer
EFPD	effective full power days
EOC(L)	end of cycle (life)
ERF	emergency response facility
F_Q	heat flux hot channel factor
F_{dH}	enthalpy rise hot channel factor
HZP	hot zero power
ICRR	inverse countrate ratio
IRNI	intermediate range nuclear instrument
ITC	isothermal temperature coefficient
LSSS	limiting safety system setting
MTC	moderator temperature coefficient
MWD/MTU	megawatt days per metric tonne of uranium
MWO	maintenance work order
NDR	WCAP-13119, The Nuclear Design Report for Vogtle Unit 2, Cycle 3.
NIS	nuclear instrument system
NR	nuclear recorder
pcm	percent millirho (reactivity)
ppmB	parts per million boron
PRNI	power range nuclear instrument
PRA	probabilistic risk assessment
RCP	reactor coolant pump
RCS	reactor coolant system
RSE	reload safety evaluation
RTP	rated thermal power
SRNI	source range nuclear instrument
TS	Technical Specifications
VCT	volume control tank