ATTACHMENT B

LaSalle County Nuclear Power Station

Unit 2 Cycle 4 Startup Test Report

LTP-1700-1, CORE VERIFICATION

PURPOSE

The purpose of this test is to visually verify that the core is loaded as intended for Unit 2 Cycle 4 operation.

CRITERIA

The as-loaded core must conform to the cycle core design used by the Core Management Organization (General Electric) in the reload licensing analysis. The core verification must be observed by a member of the Commonwealth Edison Company audit staff. Any discrepancies discovered in the loading will be promptly corrected and the affected areas reverified to ensure proper core loading prior to unit startup.

Conformance to the cycle core design will be documented by a permanent core serial number map signed by the audit participants.

RESULTS AND DISCUSSION

The Unit 2 Cycle 4 core verification consisted of a core height check performed by the fuel handlers and two videotaped passes of the core by the nuclear group. The height check verifies the proper seating of the assembly in the fuel support piece while the videotaped scans verify proper assembly orientation, location, and seating. Bundle serial numbers and orientations were recorded during the videotaped scans, for comparison to the appropriate tag boards and Cycle Management documentation. On May 16 and 17, 1990 the core was verified as being properly loaded and consistent with the General Electric Cycle 4 Cycle Management Report and the Final Station Use Loading Plan. A discrepancy was noted during the inventory. The channel fastener on fuel assembly LYF209 at core location 41-04 was observed to be bent. Consequently, at the completion of the inventory, that assembly and an adjacent assembly were removed from the core and inspected for damage. The channel fastener on LYF209 was subsequently replaced and the two fuel assemblies reloaded into the core. The affected core cell was then reverified to be properly loaded. On May 17, 1990 the videotapes were reviewed by the Assistant Lead Nuclear Engineer and the Nuclear Materials Custodian to reverify all bundle ID's, orientation, and seating.

A serial number inventory was also performed on the Unit 2 fuel pool on May 17, 1990 to verify that the fuel pool contained the proper bundles. The fuel pool contained no bundles which should have been loaded into the Unit 2 reactor.

LTP-1600-30, Single Rod Subcritical Check

PURPOSE

The purpose of this test is to demonstrate that the Unit 2 Cycle 4 core will remain subcritical upon the withdrawal of the analytically determined strongest control rod.

CRITERIA

The core must remain subcritical, with no significant increase in SRM readings, with the analytically determined strongest rod fully withdrawn.

RESULTS AND DISCUSSION

The analytically determined strongest rod for the Beginning of Cycle 4 of Unit 2 was determined by General Electric Core Management to be rod 06-35. On May 17, 1990, with a Unit 2 moderator temperature of 79 degrees Fahrenheit (as read from 2C33-R607, Reactor Bottom Head Drain, Position No. 5), rod 06-35 was single notch withdrawn to the full out position (48) and the core remained subcritical with no significant increase in SRM readings. The satisfactory completion of LTP-1600-30, Single Rod Subcritical Check, allows single control rod withdrawals for control rod testing provided moderator temperature is greater than or equal to 79 degrees Fahrenheit. This information is documented on LTP-1600-30, Attachment B, Unit Instructions for Single Control Rod Movement, of which a copy was given to the Unit 2 NSO and the Shift Engineer.

Subsequent to the performance of the Single Rod Subcritical Check all control rods were withdrawn individually to the full out position and the core remained subcritical with no significant increase in SRM readings at any time.

LTP-700-2, CONTROL ROD FRICTION AND SETTLE TESTING

PURPOSE

The purpose of this test is to demonstrate that excessive friction does not exist between the control rod blade and the fuel assemblies during operation of the control rod drive (CRD) following core alterations.

CRITERIA

With the final cell loading complete for the fuel assemblies in a control cell, the differential pressure across the CRD drive piston should not vary by more than 15 psid during a continuous insertion.

If the drive piston differential pressure during a continuous insert varies by more than 15 psid, an individual notch (insert) settling pressure test must be performed on the CRD. The differential settling pressure for an individual notch test should not be less than 30 psid, nor should it vary by more than 10 psid over a full stroke.

RESULTS AND DISCUSSION

Control Rod Drive (CRD) Friction testing was commenced after the completion of the core load verification and single rod subcritical check, and was completed on May 18, 1990. Continuous insert friction traces were obtained for all 185 CRDs. No control rods indicated excessive friction and accordingly, no notch by notch insert tests were required.

LOS-RD-SR5, CONTROL ROD DRIVE TIMING

PURPOSE

The purpose of this test is to check and set the insert and withdrawal times of the Control Rod Drives (CRDs). In addition, this surveillance will provide verification that each control rod blade is coupled to it's respective CRD mechanism.

CRITERIA

The insert and withdrawal times of a CRD should be 48 +/- 9.6 seconds (between 38.40 and 57.60 seconds). However, General Electric recommended that LaSalle change this criteria to 40 to 56 seconds for insert times and 46 to 58 seconds for withdrawal times in the cold shutdown conditions (depressurized). This change might avoid adjustments of the CRD velocity during rated reactor operation.

RESULTS AND DISCUSSION

All CRDs were tested between 06-08-90 and 06-09-90. General Electric recommended that the insert and withdraw times in a cold depressurized condition be set between 40-56 seconds and 46-58 seconds, respectively. Using this criteria could avoid timing adjustments at rated conditions. Control rod drives 46-35 and 26-27 had withdraw times of 42.6 and 38.5 seconds, respectively. Both rods are operated fully withdrawn for the entire cycle. The drives demonstrated normal scram times during the performance of LTS-1100-4, Scram Insertion Times. A coupling check was also successfully performed on each drive during the timing process.

LTS-1100-14, SHUTDOWN MARGIN (SDM) SUBCRITICAL DEMONSTRATION

PURPOSE

The purpose of this test is to demonstrate, using the adjacent rod subcritical method, that the core loading has been limited such that the reactor will be subcritical throughout the operating cycle with the strongest control rod in the full-out position (position 48) and all other rods fully inserted.

CRITERIA

If a SDM of 0.38% $\Delta K/K$ (0.38% $\Delta K/K + R$) cannot be demonstrated with the strongest control rod fully withdrawn, the core loading must be altered to meet this margin. R is the reactivity difference between the core's beginning-of-cycle SDM and the minimum SDM for the cycle. The R value for Cycle 4 is 0.131% $\Delta K/K$, with the minimum SDM occurring at 7000.0 MWD/ST into the cycle.

RESULTS AND DISCUSSION

On June 10, 1990, the local SDM demonstration was successfully performed using control rods 06-35 and 10-39. Control rod 10-39 is diagonally adjacent to 06-35, the strongest rod at beginning-of-cycle. General Electric (GE) provided, in the Cycle Startup Package, rod worth information (for control rods 06-35 and diagonally adjacent rods 10-39 and 10-31) and moderator temperature reactivity corrections to support this test. Using the GE supplied information, it was determined that with control rod 06-35 at position 48 and rod 10-39 at position 24, a moderator temperature of 125°F, and the reactor subcritical, a SDM of 0.583% $\Delta K/K$ was demonstrated. The SDM demonstrated exceeded the 0.511% $\Delta K/K$ required to satisfy Technical Specification 3.1.1, and maintained sufficient margin to the GE calculated SDM for the core at beginning-of-cycle (1.338% $\Delta K/K$) to avoid criticality during the test.

LTS-1100-1, SHUTDOWN MARGIN TEST

PURPOSE

The purpose of this test is to demonstrate, from a normal in-sequence critical, that the core loading has been limited such that the reactor will be subcritical throughout the operating cycle with the strongest control rod in the full-out position (position 48) and all other rods fully inserted.

CRITERIA

If a shutdown margin (SDM) of .38% $\Delta K/K$ (0.38% $\Delta K/K$ + R) cannot be demonstrated with the strongest control rod fully withdrawn, the core loading must be altered to meet this margin. R is the reactivity difference between the core's beginning-of-cycle SDM and the minimum SDM for the cycle. The R value for Cycle 4 is 0.131% $\Delta K/K$, so a SDM of 0.511% $\Delta K/K$ must be demonstrated.

RESULTS AND DISCUSSION

The beginning-of-cycle SDM was successfully determined from the initial critical data. The initial Cycle 4 critical occurred on June 10, 1990 on control rod 34-55 at position 16, using an A-2 sequence. The moderator temperature was 130°F and the reactor period was 60 seconds. Using rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by General Electric (in the Cycle Startup Package), the beginning-of-cycle SDM was determined to be 1.246% $\Delta K/K$ (see Table 1). The SDM demonstrated exceeded the 0.511% $\Delta K/K$ required to satisfy Technical Specification 3.1.1.

TABLE 1

SHUTDOWN MARGIN CALCULATION

Critical Rod = 34-55 @ 16

Worth of Strongest Rod = 0.02653 $\Delta K/K$ (1)

Worth of Control Rods Withdrawn to Obtain Criticality:

24 Group 1 rods at 48 = 0.03695 $\Delta K/K$ (2)
4 Group 2 rods at 48 = 0.00356 $\Delta K/K$ (3)
1 Group 2 rod at 16 = 0.000782 $\Delta K/K$ (4)

Temperature Correction = -0.0014 $\Delta K/K$ (5)
for Tm = 130 F

Period Correction = 0.0009 $\Delta K/K$ (6)
for Period = 60 seconds

Shutdown Margin Keff:
SDM Keff = 1.0000 + (1) - (2) - (3) - (4) - (5) + (6)
= 0.9875 $\Delta K/K$

SDM = (1.000 - (SDM Keff)) * 100 = 1.246% $\Delta K/K$

LTS-1100-2, CHECKING FOR REACTIVITY ANOMALIES

PURPOSE

The purpose of this test is to compare the actual and predicted critical rod configurations to detect any unexpected reactivity effects in the reactor core.

CRITERIA

In accordance with Technical Specification 3.1.2, the reactivity equivalence of the difference between the actual control rod density and the predicted control rod density shall not exceed 1% ΔK/K. If the difference does exceed 1% ΔK/K, the Core Management Engineers (General Electric Company and Commonwealth Edison Company) will be promptly notified to investigate the anomaly. The cause of the anomaly must be determined, explained, and corrected for continued operation of the unit.

RESULTS AND DISCUSSION

Three reactivity anomaly calculations were successfully performed during the Unit 2 Cycle 4 Startup Test Program, one from the initial critical, and the second and third from steady-state, equilibrium conditions at approximately 86 and 100 percent of full power.

The initial critical occurred on June 10, 1990, with control rod 34-55 at position 16, using an A-2 sequence. The moderator temperature was 130°F and the reactor period was 60 seconds. Using rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by General Electric (in the Cycle Startup Package), the actual critical was determined to be within -0.0918% ΔK/K of the predicted critical (see Table 2). The difference determined is within the 1% ΔK/K criteria of Technical Specification 3.1.2.

The first reactivity anomaly calculation for power operation was performed using data from June 22, 1990 with Unit 2 at 86.2% power at a cycle exposure of 113.0 MWD/ST, at equilibrium conditions. The predicted notch inventory from the vendor supplied data was 529 notches. The actual notch inventory was 485 notches. Using the notch worth provided by the vendor, the resulting anomaly was 0.099% ΔK/K. This value is within the 1% ΔK/K criteria of Technical Specification 3.1.2.

The second reactivity anomaly calculation for power operation was performed using data from July 3, 1990 with Unit 2 at 99.[illegible]% power at a cycle exposure of 250.9 MWD/ST, at equilibrium conditions. The predicted notch inventory from the vendor supplied data was 538 notches. The actual notch inventory was 502 notches. Using the notch worth provided by the vendor, the resulting anomaly was 0.079% ΔK/K. This value is within the 1% ΔK/K criteria of Technical Specification 3.1.2.

TABLE 2

INITIAL CRITICALITY COMPARISON CALCULATIONS

ITEM	$\Delta K/K$
Keff with all rods in at 68°F	= 0.96009 *
Reactivity inserted by 24 group 1 rods at position 48	= 0.03695 *
Reactivity inserted by 4 group 2 rods at position 48	= 0.00356 *
Reactivity inserted by 1 group 2 rod at position 16	= 0.000782*
Predicted Keff at actual critical rod pattern (68°F)	= 1.001382
Reactivity associated with the measured reactor period (period correction for 60 second period)	= 0.0009 *
Reactivity associated with moderator temperature (130°F actual, 68°F predicted)	= -0.0014 *
Reactivity Anomaly = [(predicted Keff - 1) - (period correction) + (temperature correction)] * 100%	= -0.0918% $\Delta K/K$

* - "LaSalle Unit 2 Cycle 4 Startup Package", supplied by General Electric Company.

LTS-1100-4, SCRAM INSERTION TIMES

PURPOSE

The purpose of this test is to demonstrate that the control rod scram insertion times are within the operating limits set forth by the Technical Specifications (3.1.3.2, 3.1.3.3, 3.1.3.4).

CRITERIA

The maximum scram insertion time of each control rod from the fully withdrawn position (48) to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

The average scram insertion time of all operable control rods from the fully withdrawn position (48), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following.

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

The average scram insertion time, from the fully withdrawn position (48), for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

RESULTS AND DISCUSSION

Scram testing was successfully performed between June 13, 1990 and June 14, 1990. All control rods were scram timed from full out. All control rod scram timing acceptance criteria were met during this test. The results of the testing are given below.

Position	Average Scram Times of all CRDs (secs.)	Maximum Average Scram Times in a Two-by-Two Array (secs.)
45	0.329	0.373
39	0.630	0.680
25	1.360	1.453
05	2.472	2.656

Maximum 90% scram time (position 05): CRD 38-39, 2.896 secs.

τ_{ave} (position 39) for Minimum Critical Power Ratio Limit determination: 0.630 seconds.

LTP-1600-17, CORE POWER DISTRIBUTION SYMMETRY ANALYSIS

PURPOSE

The purpose of this test is to verify the core power symmetry and the reproducibility of the TIP readings.

CRITERIA

The total TIP uncertainty obtained by averaging the uncertainties for all data sets must be less than 8.7%

The gross check of the TIP signal symmetry should yield a maximum deviation between symmetrically located pairs of less than 25%.

RESULTS AND DISCUSSION

Core power symmetry calculations were performed based upon data obtained from two full core TIP sets (OD-1). The initial TIP set was performed on July 3, 1990 at 99.9% power and the second on July 3, 1990 at 99.5% power. The average total TIP uncertainty from the two data sets was 3.607%, satisfying the criteria of the test (less than 8.7%). The average standard deviation was 3.34%.

Table 3 lists the symmetrical TIP pairs, their core locations, and their respective average deviations. The maximum deviation between symmetrical TIP pairs was 10.34% for TIP pair 48-25, satisfying the criteria of the test (less than 25%).

The results of the Random Noise Uncertainty and Geometric Noise were 0.944% and 3.48%, respectively.

A discussion of the calculational methodology is provided below.

The method used to obtain the uncertainties consisted of calculating the average of the nodal BASE ratio of TIP pairs by:

$$\bar{R} = \frac{1}{18n}\left[\sum_{i=5}^{22}\sum_{j=1}^{n} R_{i,j}\right]$$

where R_{ij} = the BASE ratio for the ith node of TIP pair j,
n = number of TIP pairs = 19.

Next, the standard deviation (expressed as a percentage) of these ratios is calculated by the following equation:

$$\sigma_R(\%) = \left[\frac{\sum_{i=5}^{22}\sum_{j=1}^{n}(R_{i,j}-\bar{R})^2}{(18n-1)}\right]^{1/2} * 100$$

The total TIP uncertainty (%) is calculated by dividing σ_R (%) by $\sqrt{2}$ because the uncertainty in one TIP reading is the desired parameter, but the measured uncertainty is the ratio of two TIP readings.

TABLE 3

TIP SIGNAL SYMMETRY RESULTS

All numbers shown are averages from two OD-1 data sets (from 7-03-90 and 7-03-90 at 99.9% and 99.5% power, respectively).

Symmetrical TIP Pair Numbers (Core Location) a	b	Absolute Difference of BASE#	Percent TIP Pair Deviation*
1 (16-09)	6 (08-17)	0.04	0.04
2 (24-09)	13 (08-25)	1.32	1.24
3 (32-09)	20 (08-33)	10.15	9.59
4 (40-09)	27 (08-41)	0.24	0.24
5 (48-09)	34 (08-49)	4.84	7.51
8 (24-17)	14 (16-25)	4.93	4.26
9 (32-17)	21 (16-33)	0.80	0.69
10 (40-17)	28 (16-41)	7.94	6.86
11 (48-17)	35 (16-49)	0.33	0.36
12 (56-17)	40 (16-57)	3.29	5.16
16 (32-25)	22 (24-33)	2.62	2.10
17 (40-25)	29 (24-41)	1.54	1.31
18 (48-25)	36 (24-49)	11.81	10.34
19 (56-25)	41 (24-57)	6.57	7.14
24 (40-33)	30 (32-41)	2.44	2.06
25 (48-33)	37 (32-49)	3.34	3.02
26 (56-33)	42 (32-57)	2.29	2.48
32 (48-41)	38 (40-49)	0.34	0.31
33 (56-41)	43 (40-57)	0.51	0.58

- where : Absolute Difference of BASE = $\left| \overline{BASE}_a - \overline{BASE}_b \right|$

and $\overline{BASE}_i = \frac{1}{16} \sum BASE_i(K)$

* - where : % Deviation = $\left[\left| \frac{BASE_a - BASE_b}{0.5(BASE_a + BASE_b)} \right| \right] * 100$