ATTACHMENT A

Summary of Unit 2 Cycle 5 Startup Test Program

SUMMARY

LaSalle Unit 2 Cycle 5 began commercial operation on April 12, 1992 following a refueling and maintenance outage. The Unit 2 Cycle 5 core loading consisted of 224 fresh fuel bundles (192 GE9B-P8CWB302-9GZ-100M-150-T and 32 GE9B-P8CWB300-9GZ-100M-150-T) and 540 reload bundles. The same bundle design being loaded for Cycle 5 was previously loaded for Unit 2 Cycle 4 operation. Unit 2 Cycle 5 had 21 LPRM strings replaced with General Electric NA-300 LPRM strings. No control blades were replaced for Unit 2 Cycle 5, however, 34 General Electric control blades were shuffled to optimize control blade lifetime. All applicable test results (neutron instrument calibration, computer monitoring results) indicate expected core performance with the new fuel design.

A comprehensive startup testing program was performed during startup and power ascension. The startup program included:

- local and in-sequence shutdown margin tests.
- reactivity anomaly calculations at initial critical and full power.
- nuclear instrument performance verifications (SRM, IRM, APRM response and overlap checks).
- instrument calibrations (LPRM, APRM, TIPs, core flow).
- control rod drive friction and full core scram timing.
- LPRM responses to control rod movement.
- process computer verification, comparison to off-line calculation.
- recirculation system performance data.
- baseline stability data acquisition.

The startup test program was satisfactorily completed on June 5, 1992. All test data was reviewed in accordance with the applicable test procedures, and exceptions to any results were evaluated to verify compliance with Technical Specification limits to ensure the acceptability of subsequent test results.

A startup test report must be submitted to the Nuclear Regulatory Commission (NRC) within 90 days following resumption of commercial power operation (in accordance with Technical Specification 6.6.A.1). The startup test report presented in this report (Attachment B) contains results (evaluations) from the following tests:

- Core Verification
- Single Rod Subcritical Check
- Control Rod Friction and Settle Testing
- Control Rod Drive Timing
- Shutdown Margin Subcritical Demonstration
- Shutdown Margin Test (In-sequence critical)
- Reactivity Anomaly Calculation (Critical and Full Power)
- Scram Insertion Times
- Core Power Distribution Symmetry Analysis

A full evaluation of the startup test program is included with the evaluation of LTP-1600-37 (On-Site Review 92-22), Unit Startup Test Program. Data from each startup is available at LaSalle Station.

ATTACHMENT B

LaSalle County Nuclear Power Station

Unit 2 Cycle 5 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 5 operation.

CRITERIA

The as-loaded core must conform to the cycle core design used by the Core Management Organization (Nuclear Fuel Services) in the reload licensing analysis. The core verification must be observed by a member of the Commonwealth Edison Company Nuclear Fuel Services 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 5 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 March 13, 1992 the core was verified as being properly loaded and consistent with Commonwealth Edison Nuclear Fuel Services Cycle 5 Cycle Management Report and the Final Station Use Loading Plan. On March 15, 1992 a partial inventory was performed on four fuel bundles that were re-channeled when friction testing (LTP-700-2) showed excessive friction between control rod 30-03 and the four surrounding bundles. On March 15, 1992, the videotapes were reviewed by the Lead Nuclear Engineer to reverify all bundle ID's, orientation, and seating.

A serial number inventory was also performed on the Unit 1 and Unit 2 fuel pools on March 16, 1992 and concluded on March 23, 1992 to verify that the fuel pool contained the proper bundles. The fuel pools 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 5 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 5 of Unit 2 was determined by Nuclear Fuel Services to be rod 22-31. On March 13, 1992, with a Unit 2 moderator temperature of 75.87 degrees Fahrenheit (as read from computer point B741, cleanup system inlet temperature), rod 22-31 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 75.87 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 commenced after the completion of the core load verification and single rod subcritical check, and was completed on March 16, 1992. Continuous insert friction traces were obtained for all 185 CRDs. Control rod 30-03 exhibited high friction during the test. The surrounding four bundles were rechanneled and the Control rod was tested satisfactorily.

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 03-25-92 and 04-01-92. All control rod drives demonstrated normal times during the performance of this test. 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% Δ K/K (0.38% Δ 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 5 is 0.0 % Δ K/K, with the minimum SDM occurring at 0.0 MWD/ST into the cycle.

RESULTS AND DISCUSSION

On April 2, 1992, the local SDM demonstration was successfully performed using control rods 22-31 and 26-35. Control rod 26-35 is diagonally adjacent to 22-31, the strongest rod at beginning-of-cycle. Nuclear Fuel Services provided, in the Cycle Startup Package, rod worth information (for control rods 22-31 and diagonally adjacent rod 26-35) and moderator temperature reactivity corrections to support this test. Using the supplied information, it was determined that with control rod 22-31 at position 48 and rod 26-35 at position 16, a moderator temperature of 160.0 degrees F, and the reactor subcritical, a SDM of 0.617% Δ K/K was demonstrated. The SDM demonstrated exceeded the 0.38% Δ K/K required to satisfy Technical Specification 3.1.1, and maintained sufficient margin to the calculated SDM for the core at beginning-of-cycle (2.082% Δ 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% Δ K/K (0.38% Δ 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 5 is 0.0% Δ K/K, so a SDM of 0.38% Δ K/K must be demonstrated.

RESULTS AND DISCUSSION

The beginning-of-cycle SDM was successfully determined from the initial critical data. The initial Cycle 5 critical occurred on April 2, 1992 on control rod 22-43 at position 08, using an A 2 sequence. The moderator temperature was 165 degrees F and the reactor period was 72 seconds. Using rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by Nuclear Fuel Services (in the Cycle Start Package), the beginning-of-cycle SDM was determined to be 2.6[illegible] Δ K/K (see Table 1). The SDM demonstrated exceeded the 0.38% Δ K/K required to satisfy Technical Specification 3.1.1.

The calculation was also performed for the April 8, 1992 critical. The reactor went critical on control rod 14-27 at 8, a moderator temperature of 165 degrees F and a reactor period of 184 seconds. The Shutdown margin was calculated to be 2.682% Δ K/K.

TABLE 1

SHUTDOWN MARGIN CALCULATION

The Following data is from the April 2, 1992 critical.

Critical Rod = 22-43 @ 08

Worth of Strongest Rod	= 0.02604 Δ K/K		(1)

Worth of Control Rods Withdrawn to Obtain Criticality:

24 Group 1 rods at 48 =	0.03556	Δ K/K	(2)
24 Group 2 rods at 48 =	0.01747	Δ K/K	(3)
6 Group 3 rods at 04 =	0.0005	Δ K/K	(4)
18 Group 3 rods at 08 =	0.00175	Δ K/K	(4)

Temperature Correction for Tm = 165°F	= -0.00165 Δ K/K	(5)
Period Correction for Period = 72 seconds	= 0.00075 Δ K/K	(6)

Shutdown Margin Keff:

SDM Keff = 1.0000 + (1) - (2) - (3) - (4) - (5) + (6)
= 0.97316 Δ K/K

SDM = (1.000 - (SDM Keff)) * 100 = 2.684% Δ 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 trends.

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% Δ K/K. If the difference does exceed 1% Δ K/K, the Core Management Engineers (Nuclear Fuel Services) 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 5 Startup Test Program, two from in-sequence criticals and a third from steady-state, equilibrium conditions at approximately 100 percent of full power.

The initial critical occurred on April 2, 1992, on control rod 22-43 at position 08, using an A-2 sequence. The moderator temperature was 165 degrees F and the reactor period was 72 seconds. Using rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by Nuclear Fuel Services (in the Cycle Startup Package), the actual critical was determined to be within 0.602% Δ K/K of the predicted critical (see Table 2). The anomaly determined is within the 1% Δ K/K allowed by Technical Specification 3.1.2.

The calculation was also performed for the April 8, 1992 critical. The reactor went critical on control rod 14-27 at 8, a moderator temperature of 165 degrees F and a reactor period of 184 seconds. The calculated Reactivity Anomaly was 0.600% Δ K/K.

The third reactivity anomaly calculation, for power operation, was performed using data from May 6, 1992 at 100% power at a cycle exposure of 308.1 MWD/ST, at equilibrium conditions. The predicted notch inventory supplied by Nuclear Fuel Services was 149 notches. The actual corrected notch inventory was 105.4 notches. Using the notch worth provided by Nuclear Fuel Services, the resulting anomaly was 0.148% Δ K/K. This value is within the 1% Δ K/K criteria of Technical Specification 3

TABLE 2

INITIAL CRITICALITY COMPARISON CALCULATIONS

ITEM		Δ K/K
Keff with all rods in at 68 degrees F	=	0.95314 *
Reactivity inserted by 24 group 1 rods at position 48	=	0.03556 *
Reactivity inserted by 24 group 2 rods at position 48	=	0.01747 *
Reactivity inserted by 6 group 3 rod at position 04	=	0.0005
Reactivity inserted by 18 group 3 rods at position 08	=	0.00175
Predicted Keff at actual critical rod pattern (68°F)	=	1.00842

Reactivity associated with the measured reactor period (period correction for 73 second period) = 0.00075 *

Reactivity associated with moderator temperature (165°F actual, 68°F predicted) = -0.00165 *

Reactivity Anomaly = [(predicted keff - 1) - (period correction) + (temperature correction)] * 100% = 0.602% Δ K/K

* - "LaSalle Unit 2 Cycle 5 Startup Package", supplied by Nuclear Fuel Services.

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 April 14-15, 1992. All control rods were scram timed from full out. All control rod scram timing acceptance criteria were met during this test. Control rod 06-27 had its pilot valves replaced and tested satisfactorily. Control rod 38-11 had a work request written for preventive maintenance and tested satisfactorily on April 21, 1992. 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	0.324	0.341
39	0.618	0.637
25	1.327	1.400
05	2.404	2.520

Tau Ave (position 39) for Minimum Critical Power Ratio Limit determination: 0.618 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 first TIP set was performed on May 4, 1992 at approximately 100% power and the second on May 5, 1992 at approximately 100% power. The TIP uncertainty from the first data set at approximately 100% power was 3.774% with an average standard deviation of 5.338%. The TIP uncertainty from the second data set was 3.105% with an average standard deviation of 4.391%. Both data sets exhibited TIP uncertainties within the 8.7% acceptance criteria.

Table 3 lists the symmetrical TIP pairs, their core locations, and their respective average deviations. The maximum deviation between symmetrical TIP pairs was 14.32% for TIP pair 05-34, satisfying the criteria of the test (less than 25%).

Table 4 lists the data calculated to determine the Random Noise Uncertainty and Geometric Noise. The Random Noise Uncertainty was determined to be 0.867% and the Geometric Noise was determined to be 3.328%.

Additionally, two full core TIP sets were performed at approximately 100% power on June 12, 1992. Although they were not officially part of the Unit 2 Cycle 5 Startup Test Program, core power symmetry calculations were reverified from the results of these TIP sets. The calculations yielded a TIP uncertainty of 3.207% with an average standard deviation of 4.535% for the first TIP set and a TIP uncertainty of 3.153% with an average standard deviation of 4.459% for the second TIP set.

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 Rij = 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:

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

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 5-4-92 and 5-5-92 at 99.7% and 99.5% power, respectively).

Symmetrical TIP Pair Numbers (Core Location) a	Symmetrical TIP Pair Numbers (Core Location) b	Absolute Difference of BASE#	Percent TIP Pair Deviation*
1 (16-09)	6 (08-17)	3.02	4.09
2 (24-09)	13 (08-25)	0.67	0.63
3 (32-09)	20 (08-33)	0.82	0.29
4 (40-09)	27 (08-41)	2.36	3.47
5 (48-09)	34 (08-49)	5.08	14.32
8 (24-17)	14 (16-25)	0.10	0.27
9 (32-17)	21 (16-33)	1.73	1.61
10 (40-17)	28 (16-41)	1.63	1.26
11 (48-17)	35 (16-49)	4.38	4.18
12 (56-17)	40 (16-57)	2.50	1.95
16 (32-25)	22 (24-33)	2.08	1.82
17 (40-25)	29 (24-41)	5.81	5.93
18 (48-25)	36 (24-49)	1.36	1.29
19 (56-25)	41 (24-57)	0.42	4.09
24 (40-33)	30 (32-41)	0.41	0.97
25 (48-33)	37 (32-49)	2.22	1.84
26 (56-33)	42 (32-57)	4.05	7.29
32 (48-41)	38 (40-49)	0.15	0.10
33 (56-41)	43 (40-57)	2.62	5.10

\# - where: Absolute Difference of BASE = $\overline{BASE_a} - \overline{BASE_b}$

and $\overline{BASE_i} = \frac{1}{18} \sum BASE_i(K)$

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

TABLE 4

RANDOM NOISE UNCERTAINTY AND GEOMETRIC NOISE DATA

Per LTP-1600-17, Attachment D, at approximately 99.9% thermal power, the Random Noise Uncertainty and Geometric Noise Data Analysis was performed. The results of the calculations are presented below. The Random Noise was determined to be 0.867%. The Geometric Noise was determined to be 3.328%.

Node	Avg. Base
5	124.71
6	121.26
7	116.36
8	116.38
9	114.36
10	112.65
11	111.93
12	112.87
13	111.77
14	107.40
15	109.20
16	111.29
17	106.65
18	105.56
19	103.01
20	93.80
21	82.31
22	72.16

ATTACHMENT C

List of References

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel", Latest Approved Revision.

2. LaSalle 2 Cycle 5 Cycle Management Report, DRF Number LS2-0006 Volume 2.