Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

April 11, 1990

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: LaSalle County Station Unit 1
Startup Test Report Summary
NRC Docket No. 50-373

Dear Sir:

Enclosed for your information and use is the LaSalle County Station Unit 1 Cycle 4 Startup Test Report Summary. This report is submitted in accordance with Technical Specification NFP-18, Section 6.6.A.1.

LaSalle Unit 1 Cycle 4 began commercial operation on January 10, 1990 following a refueling and maintenance outage. The Unit 1 Cycle 4 core loading consisted of 172 fresh GE 8x8 NB (GE9B) Fuel Bundles and 592 Reload Bundles. The new Fuel (GE 9B) has an option for multiple lattice types (i.e., axial zone enrichment and gadolinia).

The startup test program was satisfactorily completed on February 17, 1990. 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 and to ensure the acceptability of subsequent test results.

Attached are the evaluation results from the following tests:

- Core verification
- 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

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If you have any questions concerning this matter, please contact this office.

Very truly yours,

Wayne E. Morgan

W. E. Morgan
Nuclear Licensing Administrator

cc: A.Bert Davis-Regional Administrator, RIII
NRC Resident Inspector, LSCS
R. Pulsifer - Project Manager, NRC

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ATTACHMENT A

Summary of Unit 1 Cycle 4 Startup Test Program

LaSalle Unit 1 Cycle 4 Startup Test Report

SUMMARY

LaSalle Unit 1 Cycle 4 began commercial operation on January 10, 1990 following a refueling and maintenance outage. The Unit 1 Cycle 4 core loading consisted of 172 fresh GE8x8NB (GE9B) fuel bundles and 592 reload bundles. The new fuel (GE9B) has an option for multiple lattice types (i.e., axial zoned enrichment and gadolinia). The expected changes in the operating characteristics of the new fuel design are described in LOSR-89-072, which evaluated the incorporation of this fuel design. 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 January 25, 1990 with the exception of the Recirculation System Performance test, which was completed on February 17, 1990. 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 on-site review (Attachment B) contains results (evaluations) from the following test:

- Core Verification
- 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 90-05), Unit Startup Test Program. Data from each startup is available at LaSalle Station.

FINDINGS AND RECOMMENDATIONS

Based upon the preceding discussion and the review of the startup test report, On-Site Review recommends submittal of the "LaSalle County Nuclear Power Station Unit 1 Cycle 4 Startup Test Report" (Attachments B and C) to the NRC in accordance with Technical Specification 6.6.A.1.

ATTACHMENT B

LaSalle County Nuclear Power Station

Unit 1 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 1 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 1 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 December 5, 1989, 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. On December 6, 1989, the videotapes were reviewed by the Lead Nuclear Engineer to reverify all bundle ID's, orientation, and seating.

The core loading differed from the Reference Core Loading Pattern assumed in the reload licensing analysis (Reference 1) in that the core loading did not utilize one (1) BP8CRB299L fuel assembly which had been scheduled for use during Cycle 4. This change was reviewed and found to be acceptable against the requirements of Reference 2. The change was required as a result of the leaker fuel assembly which was identified during the Unit 1 third refuel outage. The leaker assembly was not loaded. The core pattern was shuffled and this assembly was replaced with a 8CRB219 fuel assembly in accordance with General Electric procedures. General Electric re-examined the parameters specified in Section 3.4.2 of Reference 2. They determined that only one parameter, cold shutdown margin, would be affected by the bundle substitutions. Since cold shutdown margin was recalculated for the Station Use Loading Plan (i.e., the as loaded core) and found to be within acceptable margins, the reload license analysis is not affected.

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 0.0% $\Delta K/K$, with the minimum SDM occurring at 0.0 MWD/ST into the cycle.

RESULTS AND DISCUSSION

On January 4, 1990, the local SDM demonstration was successfully performed using control rods 38-55 and 42-51. Control rod 42-51 is diagonally adjacent to 38-55, the strongest rod at beginning-of-cycle. General Electric (GE) provided, in the Cycle Startup Package, rod worth information (for control rods 38-55 and diagonally adjacent rods 42-51 and 34-51) and moderator temperature reactivity corrections to support this test. Using the GE supplied information, it was determined that with control rod 38-55 at position 48 and rod 42-51 at position 08, a moderator temperature of 135°F, and the reactor subcritical, a SDM of 0.502% $\Delta K/K$ was demonstrated. The SDM demonstrated exceeded the 0.38% $\Delta K/K$ required to satisfy the test criteria, and maintained sufficient margin to the GE calculated SDM for the core at beginning-of-cycle (1.684% $\Delta K/K$) 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% ΔK/K (0.38% ΔK/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 0.0% ΔK/K, i.e., the minimum SDM occurs at the beginning of cycle.

RESULTS AND DISCUSSION

The beginning-of-cycle SDM was successfully determined from the initial critical data. The initial Cycle 4 critical occurred on January 4, 1990 , on control rod 18-15 at position 16, using an A-2 sequence. The moderator temperature was 141 °F and the reactor period was 290 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 1.717% ΔK/K (see Table 1). The SDM demonstrated exceeded the .38% ΔK/K required to satisfy Technical Specification 3.1.1.

TABLE 1

SHUTDOWN MARGIN CALCULATION

Critical Rod = 18-15 @ 16

Worth of Strongest Rod = 0.02839 K/K (1)

Worth of Control Rods Withdrawn to Obtain Criticality:

- 24 Group 1 rods at 48 = 0.03525 K/K (2)
- 7 Group 2 rods at 48 = 0.01198 K/K (3)
- 1 Group 2 rod at 16 = 0.00024 K/K (4)

Temperature Correction = -0.0017 K/K (5)
for Tm = 141 F

Period Correction = 0.00021 K/K (6)
for Period = 290 seconds

Shutdown Margin Keff:

SDM Keff = 1.0000 + (1) - (2) - (3) - (4) - (5) + (6)
= 0.98283 K/K

SDM = (1.000 - (SDM Keff)) * 100 = 1.717% 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% $\Delta K/K$. If the difference does exceed 1% $\Delta K/K$, 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

Two reactivity anomaly calculations were successfully performed during the Unit 1 Cycle 4 Startup Test Program, one from the initial critical and the second from steady-state, equilibrium conditions at approximately 97 percent of full power.

The initial critical occurred on Janauary 5, 1990, with control rod 18-15 at position 16, using an A-2 sequence. The moderator temperature was 141 °F and the reactor period was 290 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 0.033% $\Delta K/K$ of the predicted critical (see Table 2). The difference determined is within the 1% $\Delta K/K$ criteria of Technical Specification 3.1.2.

The reactivity anomaly calculation for power operation was performed using data from January 22, 1990 with Unit 1 at 96.7% power at a cycle exposure of 189.5 MWD/ST, at equilibrium conditions. The predicted notch inventory from the vendor supplied data was 540 notches. The actual notch inventory was 446 notches. Using the notch worth provided by the vendor, the resulting anomaly was 0.18% $\Delta K/K$. This value is within the 1% $\Delta K/K$ criteria of Technical Specification 3.1.2.

TABLE 2

INITIAL CRITICALITY COMPARISON CALCULATIONS

ITEM	K/K
Keff with all rods in at 68°F	= 0.95477 *
Reactivity inserted by 24 group 1 rods at position 48	= 0.03525 *
Reactivity inserted by 7 group 2 rods at position 48	= 0.01198 *
Reactivity inserted by 1 group 2 rod at position 16	= 0.00024 *
Predicted Keff at actual critical rod pattern (68°F)	= 1.00224
Reactivity associated with the measured reactor period (period correction for 290 second period)	= 0.00021 *
Reactivity associated with moderator temperature (141°F actual, 68°F predicted)	= -0.0017 *
Reactivity Anomaly = [(predicted Keff - 1) - (period correction) + (temperature correction)] * 100%	= 0.033% ΔK/K

* - "LaSalle Unit 1 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 January 10, 1990 and January 11, 1990. All control rod scram timing acceptance criteria were met during this test. The results of the test 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	0.339
39	0.625	0.653
25	1.352	1.459
05	2.467	2.675

Maximum 90% scram time (position 05): CRD 42-11, 2.848 secs.

τ_{ave} (position 39) for Minimum Critical Power Ratio Limit determination: 0.625 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 January 18, 1990 at 86.1% power and the second on January 18, 1990 at 86.1% power. The average total TIP uncertainty from the two data sets was 3.586%, satisfying the criteria of the test (less than 8.7%). The average standard deviation was 4.05%.

Table 3 lists the symmetrical TIP pairs, their core locations, and their respective average deviations. The maximum deviation between symmetrical TIP pairs was 14.88% for TIP pair 19-41, satisfying the criteria of the test (less than 25%).

The results of the Random Noise Uncertainty and Geometric Noise were 0.859% and 3.96%, 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:

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

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:

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

The total TIP uncertainty (%) is calculated by dividing $\sigma_{\bar{R}}$ (%) 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 1-18-90 and 1-18-90 at 86.1% and 86.1% power, respectively.

Symmetrical TIP Pair Numbers (Core Location) a	b	Absolute Difference of BASE#	Percent TIP Pair Deviation*
1 (16-09)	6 (08-17)	2.82	2.61
2 (24-09)	13 (08-25)	5.67	4.89
3 (32-09)	20 (08-33)	5.95	5.32
4 (40-09)	27 (08-41)	2.92	2.57
5 (48-09)	34 (08-49)	0.62	0.79
8 (24-17)	14 (16-25)	4.12	3.43
9 (32-17)	21 (16-33)	8.15	6.21
10 (40-17)	28 (16-41)	1.63	1.46
11 (48-17)	35 (16-49)	5.43	4.94
12 (56-17)	40 (16-57)	2.72	3.88
16 (32-25)	22 (24-33)	5.41	4.97
17 (40-25)	29 (24-41)	0.93	0.77
18 (48-25)	36 (24-49)	1.50	1.25
19 (56-25)	41 (24-57)	13.86	14.88
24 (40-33)	30 (32-41)	7.10	6.23
25 (48-33)	37 (32-49)	4.69	3.71
26 (56-33)	42 (32-57)	3.06	3.45
32 (48-41)	38 (40-49)	0.32	0.29
33 (56-41)	43 (40-57)	4.77	5.30

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

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

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

<u>ATTACHMENT C</u>

List of References

1. Letter from G. J. Diederich to the Vice President of BWR Operations, Superintendent of Off-Site Reviews, and the Assistant Vice President of Quality Programs and Assessment, dated December 8, 1989, transmitting LaSalle On-Site Review 89-072.

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel", Revision 9.