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September 8, 1988

Mr. James Lieberman, Director
Office of Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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

References (1): C.M. Allen letter to USNRC dated January 18, 1988 transmitting Reload Licensing Package for Unit 1.

(2): NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel", Revision 8.

Dear Sir:

Enclosed for your information and use is LaSalle County Station Unit 1 Cycle 3 Startup Test Report Summary. This report is submitted in accordance with Technical Specification NPF-11, Section 6.6.A.1.

LaSalle Unit 1 Cycle 3 began commercial operation on June 16, 1987 following a refueling and maintenance outage. The Unit 1 Cycle 2 core loading consisted of 224 fresh GE 8x8EB bundles and 540 relaod bundles. The new fuel has an option for multiple lattice types (i.e., axial zoned guidelines).

The startup test program was satisfactorily completed on August 6, 1987. 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.

A startup test report is required to be submitted to the Nuclear Regulatory Commission (NRC) within 90 days following resumption of commercial power operation.

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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

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

Very truly yours,

C M Allen

C. M. Allen
Nuclear Licensing Administrator

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Attachments

cc: Regional Administrator - RIII
NRC Resident Inspector - LSCS
Paul Shemanski - NRR

5075K

LTP-1700-1, CORE VERIFICATION

PURPOSE

The purpose of this test is to visually verify that the core is loaded as intended for Cycle 3 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 3 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 scan, for comparison to the appropriate tag boards and Cycle Management documentation. On June 8, 1988, the core was verified as being properly loaded and consistent with the General Electric Cycle 3 Cycle Management Report. On June 9, 1988, 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 (transmitted to the Nuclear Regulatory Commission as Attachment F to Reference 1) assumed in the reload licensing analysis in that the core loading did not utilize twenty (20) 8CRB176 fuel assemblies. These 20 assemblies were replaced with 20 8CRB219 fuel assemblies in accordance with General Electric procedures. General Electric re-examined the six parameters specified in Section 3.4.3 of Reference 2. General Electric 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 0.709% 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 3 is 0.329% ΔK/K, with the minimum SDM occurring at 5,000 MWD/ST into the cycle.

RESULTS AND DISCUSSION

On July 4, 1988, the local SDM demonstration was successfully performed using control rods 18-55 and 22-51. Control rod 22-51 is diagonally adjacent to 18-55, the strongest rod at beginning-of-cycle. General Electric (GE) provided, in the Cycle Startup Package, rod worth information (for control rods 18-55 and diagonally adjacent rods 22-51 and 14-51) and moderator temperature reactivity corrections to support this test. Using the GE supplied information, it was determined that with control rod 18-55 at position 48 and rod 22-51 at position 16, a moderator temperature of 158°F, and the reactor subcritical, a SDM of 0.752% ΔK/K was demonstrated. The SDM demonstrated exceeded the 0.709% ΔK/K required to satisfy the test criteria, and maintained sufficient margin to the GE calculated SDM for the core at beginning-of-cycle (1.598% ΔK/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 0.709% $\Delta K/K$ (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 difference between the core's beginning-of-cycle SDM and the minimum SDM for the cycle. The R value for Cycle 3 is 0.329% $\Delta K/K$, with minimum SDM occurring at 5,000 MWD/ST into the cycle.

RESULTS AND DISCUSSION

The beginning-of-cycle SDM was successfully determined from the initial critical data. The initial Cycle 3 critical occurred on July 4, 1988, on control rod 34-55 at position 18, using an A-2 sequence. The moderator temperature was 155°F and the reactor period was 87 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 1.228% $\Delta K/K$ (see Table 1). The SDM demonstrated exceeded the 0.709% $\Delta K/K$ required to satisfy Technical Specification 3.1.1.

TABLE 1

SHUTDOWN MARGIN CALCULATION

Critical Rod = 34-55 @ 18

Worth of Strongest Rod = 0.02756 $\Delta K/K$ (1)

Worth of Control Rods Withdrawn to Obtain Criticality:

- 24 Group 1 rods at 48 = 0.03626 $\Delta K/K$ (2)
- 4 Group 2 rods at 48 = 0.00527 $\Delta K/K$ (3)
- 1 Group 2 rod at 18 = 0.00096 $\Delta K/K$ (4)

Temperature Correction = -0.0020 $\Delta K/K$ (5)
for Tm = 155°F

Period Correction = 0.00065 $\Delta K/K$ (6)
for Period = 87 seconds

Shutdown Margin Keff:

SDM Keff = 1.0000 + (1) - (2) - (3) - (4) - (5) + (6)
= 0.98772 $\Delta K/K$

SDM = (1.000 - (SDM Keff)) * 100 = 1.228% $\Delta K/K$

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% $\Delta K/K$. 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 3 Startup Test Program, one from the initial critical and the second from steady-state, equilibrium conditions at approximately 87 percent of full power.

The initial critical occurred on July 4, 1988, with control rod 34-55 at position 18, using an A-2 sequence. The moderator temperature was 155°F and the reactor period was 87 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 -0.370% $\Delta K/K$ 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 on July 18, 1988 with Unit 1 at 86.9% power at a cycle exposure of 120 MWD/ST, at equilibrium conditions. The predicted notch inventory from the vendor supplied data was 990 notches. The actual notch inventory, corrected for power and flow values which were less than rated, was 1060 notches. Using the notch worth provided by the vendor, the resulting anomaly was -0.14% $\Delta K/K$. This value is within the 1% $\Delta K/K$ criteria of Technical Specification 3.1.2.

TABLE 2

INITIAL CRITICALITY COMPARISON CALCULATIONS

ITEM	ΔK/K
Keff with all rods in at 68°F	= 0.95646 *
Reactivity inserted by 24 group 1 rods at position 48	= 0.03626 *
Reactivity inserted by 4 group 2 rods at position 48	= 0.00527 *
Reactivity inserted by 1 group 2 rod at position 18	= 0.00096 *
Predicted Keff at actual critical rod pattern (68°F)	= 0.99895
Reactivity associated with the measured reactor period (period correction for 87 second period)	= 0.00065 *
Reactivity associated with moderator temperature (155°F actual, 68°F predicted)	= 0.002 *
Reactivity Anomaly = [(predicted Keff - 1) - (period correction) - (temperature correction)] * 100%	=-0.370% ΔK/K

* - "LaSalle Unit 1 Cycle 3 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 July 8, 1988 and July 9, 1988. 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	0.621	0.645
25	1.331	1.381
05	2.418	2.525

Maximum 90% scram time (position 05): CRD 18-51, 2.656 secs.

τ_{ave} (position 39) for Minimum Critical Power Ratio determination: 0.621 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 August 2, 1988 at 93.0% power, and then repeated on August 3, 1988 at 92% power. The average total TIP uncertainty from the two data sets was 3.487%, satisfying the criteria of the test (less than 8.7%). The average standard deviation was 4.932%.

Table 3 lists the symmetrical TIP pairs, their core locations, and their respective average deviations. The maximum deviation between symmetrical TIP pairs was 8.32% for TIP pair 33-43, satisfying the criteria of the test (less than 25%).

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:

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

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:

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

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 8-2-88 and 8-3-88 at 93% and 92% power, respectively.

Symmetrical TIP Pair Numbers (Core Location) a	b	Absolute Difference of BASE#	Percent TIP Pair Deviation*
1 (16-09)	6 (08-17)	1.14	1.40
2 (24-09)	13 (08-25)	3.08	2.90
3 (32-09)	20 (08-33)	6.73	5.61
4 (40-09)	27 (08-41)	4.71	4.54
5 (48-09)	34 (08-49)	2.19	3.43
8 (24-17)	14 (16-25)	2.41	1.95
9 (32-17)	21 (16-33)	0.07	1.31
10 (40-17)	28 (16-41)	2.17	1.73
11 (48-17)	35 (16-49)	4.04	3.99
12 (56-17)	40 (16-57)	0.20	1.36
16 (32-25)	22 (24-33)	3.32	2.83
17 (40-25)	29 (24-41)	1.49	1.50
18 (48-25)	36 (24-49)	8.14	6.84
19 (56-25)	41 (24-57)	5.40	6.45
24 (40-33)	30 (32-41)	4.11	3.55
25 (48-33)	37 (32-49)	8.42	7.21
26 (56-33)	42 (32-57)	3.91	3.82
32 (48-41)	38 (40-49)	6.96	5.82
33 (56-41)	43 (40-57)	6.31	8.32

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

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

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