Commonwealth Edison Company
LaSalle Generating Station
2601 North 21st Road
Marseilles, IL 61341-9757
Tel 815-357-6761

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United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: LaSalle Unit 2 Cycle 7 Startup Test Report
NRC Docket Number 50-374

The Attachment to this letter presents the LaSalle Unit 2 Cycle 7 Startup Test Report. This report is being submitted in accordance with Technical Specification 6.6.A.1. Additional startup test results are available at LaSalle Station.

If there are any questions or comments concerning this letter, please refer them to me at (815) 357-6761, extension 2212.

Respectfully,

D.J. Ray
Station Manager
LaSalle County Station

cc: H. J. Miller, Regional Administrator, Region III
W. D. Reckley, Project Manager, NRR
P. G. Brochman, Senior Resident Inspector, LaSalle
D. L. Farrar, Nuclear Regulatory Services Manager, NORS
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LaSalle Unit 2 Cycle 7 Startup Test Report

SUMMARY

LaSalle Unit 2 Cycle 7 began commercial operation on May 16, 1995 following a refueling and maintenance outage. The Unit 2 Cycle 7 core loading consisted of 176 fresh fuel bundles (96 GE9B-PCWB322-11-GZ-100M-150-T and 80 GE9B-P8CWB320-9GZ3-100M-150-T), and 588 reload bundles. The Cycle 7 reload bundle is the same bundle design that was previously loaded in Unit 2 Cycle 6. In addition, 6 LPRM strings were replaced with General Electric NA-300 LPRM strings. No control blades were replaced for Unit 2 Cycle 7, however, 44 control blades were shuffled. 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:

- 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.
- baseline stability data acquisition.

The startup test program was satisfactorily completed on June 30, 1995. 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 review 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 Test (In-sequence critical)
- Reactivity Anomaly Calculation (Critical and Full Power)
- Scram Insertion Times
- Core Power Distribution Symmetry Analysis

FINDINGS AND RECOMMENDATIONS

Based upon the preceding discussion and the review of the startup test report, the "LaSalle County Nuclear Power Station Unit 2 Cycle 7 Startup Test Report" is submitted to the NRC in accordance with Technical Specification 6.6.A.1.

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 7 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 7 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 Core loading map and Cycle Management documentation. On April 9, 1995, the core was verified as being properly loaded and consistent with Commonwealth Edison Nuclear Fuel Services LaSalle 2 Cycle 7 Design Basis Loading Plan.

A serial number inventory was also performed prior to core load on the Unit 2 Discharge Queue on March 14, 1995 to verify that the discharge queue contained the proper bundles. The discharge queue 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 7 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 7 of Unit 2 was determined by Nuclear Fuel Services to be rod 50-39. On April 10, 1995, with a Unit 2 moderator temperature of 84 degrees Fahrenheit, rod 50-39 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 84 degrees Fahrenheit. This information is documented on LTP-1600-30, Attachment A.

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 April 11, 1995. Continuous insert friction traces were obtained for all 185 CRDs. All rods 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) to give indication of seal wear. This change might avoid adjustments of the CRD velocity during rated reactor operation.

RESULTS AND DISCUSSION

All CRDs testing was complete on May 5, 1995. Control rods 02-35, 34-51, 54-35, 58-35, 58-39 and 30-23 had withdrawal times faster than 46 seconds (but greater than 38.4 seconds) due to degraded drive seals. The above listed control rods directional control valves were tested and found to be operating properly. These control rods are scheduled to be replaced during the next refueling outage.

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 7 is 0.0% delta K/K, so a SDM of 0.38% 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 7 critical occurred on May 16, 1995 on control rod 26-55 at position 12, using an A-2 sequence. The moderator temperature was 133.8 degrees F and the reactor period was 356 seconds. Using rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by Nuclear Fuel Services (in the Cycle Startup Package), the beginning-of-cycle SDM was determined to be 0.915% delta K/K (see Table 1). The SDM demonstrated exceeded the 0.38% delta K/K required to satisfy Technical Specification 3.1.1.

TABLE 1

SHUTDOWN MARGIN CALCULATION

Critical Rod = 26-55 @ 12

Predicted Keff at Critical Rod Pattern

(1) 0.9990 K_{eff}

Period Reactivity Correction From CMR

(2) 0.00015 delta K/K

Moderator Temperature Reactivity Correction

(3) -0.0016 delta K/K

Keff with strongest rod out from CMR

(4) 0.9881 delta K/K

Shutdown Margin Keff:

SDM Keff = [(1) - (2) + (3) - (4)] x 100
= 0.915 % 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 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% delta K/K. If the difference does exceed 1% delta 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

Two reactivity anomaly calculations were successfully performed during the Unit 2 Cycle 7 Startup Test Program, one from the in-sequence critical and one from steady-state, equilibrium conditions at approximately 100 percent of full power.

The initial critical occurred on May 16, 1995, on control rod 26-55 at position 12, using an A-2 sequence. The moderator temperature was 133.8 degrees F and the reactor period was 356 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 -0.275% delta K/K of the predicted critical (see Table 2). The anomaly determined is within the 1% delta K/K allowed by Technical Specification 3.1.2.

The reactivity anomaly calculation, for power operation, was performed using data from June 28, 1995 at 99.1% power at a cycle exposure of 148.5 MWD/ST, at equilibrium conditions. The predicted notch inventory supplied by Nuclear Fuel Services was 837.6 notches.

The actual corrected notch inventory was 794.34 notches. Using the notch worth provided by Nuclear Fuel Services, the resulting anomaly was 0.11% 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

Predicted Keff at actual Critical pattern

(1) 0.9990 K_{eff}

Reactivity Period Correction

(2) 0.00015 delta K/K

Moderator Temperature Correction

(3) -0.0016 delta K/K

Reactivity Anomaly

RA = [((1) - 1.0) - (2) + (3)] x 100
= -0.275 % delta K/K

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 completed on April 30, 1995. 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	Maximum Average Scram Times of all CRDs (secs.)	Average Scram Times in a Two-by-Two Array (secs.)
45	0.329	0.344
39	0.627	0.653
25	1.347	1.416
05	2.434	2.563

Tau Ave (position 39) for Minimum Critical Power Ratio Limit determination: 0.627 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 a full core TIP set (OD-1) performed on June 21, 1995 at approximately 100% power. The TIP uncertainty was 3.1% with an average standard deviation of 4.38% which is within the 8.7% acceptance criteria.

The maximum deviation between symmetrical TIP pairs was 13.42% for TIP pair 05-34, satisfying the criteria of the test (less than 25%). This particular pair includes one Westinghouse and one GE string. The Westinghouse LPRM is constructed with thicker metal so deviation is greater for this pair. All of the other strings were under 6% deviation.