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United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

LaSalle County Station, Unit 1
Facility Operating License No. NPF-11
NRC Docket No. 50-373

Subject: LaSalle Unit 1 Cycle 11 Startup Test Report Summary

Enclosed for your information is the LaSalle County Station Unit 1 Cycle 11 Startup Test Report. This report is submitted in accordance with Technical Requirements Manual Section 5.0.b.

LaSalle County Station Unit 1 Cycle 11 began commercial operation on February 12, 2004, following a refueling and maintenance outage. The Unit 1 Cycle 11 core loading consisted of 290 fresh General Electric GE-14 fuel bundles and 474 reload bundles manufactured by Framatome-ANP Power Corporation. This is the first use of the GE-14 fuel type at LaSalle Station. Also installed in the Unit 1 Cycle 11 reactor were 10 new GE/Reuter-Stokes NA-300 Local Power Range Monitors (LPRM's), and 17 new General Electric Marathon control rod blades. The inlet mixing section for Jet Pump #19 was replaced.

Attached are the evaluation results from the following tests:

- Reactor 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
- TIP Measurement Uncertainty (Core Power Distribution Symmetry Analysis)
- Reactor Recirculation System Performance

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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.

Should you have any questions concerning this letter, please contact Mr. Glen Kaegi, Regulatory Assurance Manager, at (815) 415-2800.

Respectfully,

George P. Barnes

George P. Barnes
Site Vice President
LaSalle County Station

Attachment

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – LaSalle County Station

Reactor Core Verification

Purpose

The purpose of this test is to visually verify that the core is loaded as intended for Unit 1 Cycle 11 operation.

Criteria

The as-loaded core must conform to the cycle core design used by the Core Management Organization (General Electric & Nuclear Fuels) in the reload licensing analysis. Any discrepancies discovered in the loading will be promptly corrected and the affected areas re-verified to ensure proper core loading prior to unit startup.

A permanent core serial number map signed by the audit participants will document conformance to the cycle core design.

Results and Discussion

Core verification was performed concurrently with core load and shuffle per core verification guideline NF-AA-330-1001. The Unit 1 Cycle 11 core verification consisted of a core height, assembly orientation, assembly location, and assembly seating check performed by reactor services and reactor engineering. Bundle serial numbers and orientations were recorded during the videotaped scans for comparison to the appropriate core loading map and Cycle Management documentation. On January 29, 2004, through January 30, 2004, the core was verified as being properly loaded and consistent with the LaSalle 1 Cycle 11 Design Basis Loading Plan per Transmittal of Design Information (TODI) # NF0300069, Revision 1. This was documented in Work Order (WO) # 465411-01.

Single Rod Subcritical Check

Purpose

The purpose of this test is to demonstrate that the Unit 1 Cycle 11 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 Source Range Monitor (SRM) readings, with the analytically determined strongest rod fully withdrawn.

Results and Discussion

The analytically determined strongest rod for the Beginning of Cycle 11 for Unit 1 was determined by Nuclear Fuels to be rod 30-31 per TODI# NF0400003, Revision 0. On January 31, 2004, with a Unit 1 moderator temperature of 82.5 degrees Fahrenheit, rod 30-31 was withdrawn to the full out position (48) and the core remained subcritical with no significant increase in SRM readings. The satisfactory completion of procedure LTP-1600-30, Single Rod Subcritical Check, allows single control rod withdrawals for control rod testing. This information is documented on LTP-1600-30, Attachment A.

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 rod drift alarm shall not be received when moving the control rod from position 00 to 02, and then to 04.

Friction testing shall be performed on the respective control rod drives(s) when any condition listed below is applicable:

- After relocation or replacement of the Control Rod Drive Mechanism (CRDM).
- After relocation or replacement of Control Rod Blades.
- After maintenance or modification of an installed CRDM that could affect the performance of the drive.
- Prior to initial criticality of a new operating cycle, for any cell when any condition listed below is met:
  - A channel in the cell is beginning its third cycle in a peripheral location.
  - The combined peripheral residence time for any two channels in a control cell exceeds 4 cycles.
  - Bundle-average exposure for any fuel in the control cell exceeds 30 GWD/ST (~27.24 GWD/MT).
- The Unit Nuclear Engineer or CRD System Engineer determines that friction testing is appropriate.

Results and Discussion

CRD Friction Testing commenced after the completion of the core load verification and single rod subcritical check. There was no indication of excessive friction on the control rods tested as required by the above criteria since none of the rods tested produced a rod drift alarm. The testing was completed on February 3, 2004, per WO# 465415-01 and 564817-01.

Control Rod Drive Timing

Purpose

The purpose of this test is to check and set the Updated Final Safety Analysis Report (UFSAR) required insert and withdrawal speeds of the CRDs.

Criteria

LOS-RD-SR5, Control Rod Drive Timing, states the withdraw times from the full in (notch 00) to the full out (notch 48) position should be between 45 and 60 seconds and insert times from the full out to the full in position should be between 40 and 55 seconds.

Results and Discussion

LOS-RD-SR5 was performed satisfactory for all CRDMs requiring post maintenance testing. As found control rod speeds are satisfactory per the UFSAR requirements. Timing was completed by February 10, 2004, per WO#s 563171-01 and 465360-01.

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 remain subcritical throughout the operating cycle with the strongest worth control rod in the full-out position (position 48) and all other rods fully inserted.

Criteria

If a shutdown margin (SDM) of 0.38% delta K/K + R cannot be demonstrated with the strongest worth 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 11 is 0.0% delta K/K per TODI# NF0400012, Revision 0, 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 11 critical occurred on February 11, 2004, on control rod 26-07 at position 24, using an A-2 sequence. The moderator temperature was 139.3 degrees F and the reactor period was 128 seconds. Using rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by Nuclear Fuels in TODI# NF 0400012, the beginning-of-cycle SDM was determined to be 1.041% delta K/K. The SDM exceeded the 0.38% delta K/K that was required to satisfy the Technical Specifications.

Reactivity Anomaly Determination

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 the Technical Specifications, the reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall not exceed 1% delta K/K at full power steady state conditions. If the difference does exceed 1% delta K/K, the Core Management Engineers (Nuclear Fuel Management) 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 11 Startup Test Program. One reactivity anomaly calculation from the in-sequence critical and one from steady state, equilibrium conditions at approximately 100 percent of full power.

The initial critical occurred on February 11, 2004, on control rod 26-07 at position 24, using an A-2 sequence. The moderator temperature was 139.3 degrees F and the reactor period was 128 seconds. Using rod worth information, moderator temperature, reactivity corrections, and period reactivity corrections supplied by Nuclear Fuels, the actual critical was determined to be within 0.4% delta K/K of the predicted critical. The anomaly determined is within the 1% delta K/K required for BOC conditions as stated in procedure NF-AB-715. This was documented in NF-AB-715 Attachment 2.

The reactivity anomaly calculation for full power steady state operation was performed on February 16, 2004. The data used was from 99.9% power at a cycle exposure of 83.7 MWD/MT at equilibrium conditions. The expected Keff supplied by Nuclear Fuels was 1.0030. The actual Keff was 1.0043. The resulting anomaly was 0.13% delta K/K. This value is within the 1% delta K/K criteria of Technical Specifications.

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.

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.

No more than 12 OPERABLE control rods shall be "slow" in accordance with the below table. In addition, no more than 2 Operable control rods that are "slow" shall occupy adjacent locations.

When the scram insertion time of an operable control rod from the fully withdrawn position (48), based on de-energization of the scram pilot valve solenoids as time zero, exceeds any of the following, that control rod is considered "slow":

Notch Position	Fully Withdrawn Time (seconds)
45	0.52
39	0.80
25	1.77
05	3.20

Results and Discussion

Scram testing was successfully completed on February 13, 2004 per WO# 465417-01. All 185 rods were scram timed; most during the reactor pressure vessel leakage testing (Hydro) prior to startup; and, the remainder at power prior to exceeding 40% rated thermal power. Two control rods (42-07 and 34-27) were determined to be "slow." Work requests to repair these CRDMs have been written. The results of the testing are given below.

Notch Position	Core Average Scram Times of all CRDs (sec)
45	0.371
39	0.668
25	1.380
05	2.455

The τ_{ave} was calculated to be 0.668 seconds. These results also meet the Option B Scram Speeds referenced in the Unit 1 Cycle 11 Core Operating Limits Report (TRM Appendix I).

TIP Measurement Uncertainty

Purpose

The purpose of this test is to verify the core power symmetry.

Criteria

The χ^2 value of the total measured TIP uncertainty must be less than the critical value at the 1% confidence level (i.e., 36.19 for 19 TIP pairs).

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 February 19, 2004, at approximately 100% power. The TIP set was performed with all 5 TIP machines operable. All traces were obtained. The χ^2 value was 17.94, which satisfies the test criteria of 36.19 for 19 pairs. The maximum deviation between symmetrical TIP pairs was 10.72%, which is within the 25% acceptance criteria.

Recirculation System Performance

Purpose

The purpose of this procedure is to collect sufficient baseline data at the beginning of cycle to establish the following relationships:

- core thermal power vs. total core flow
- recirculation total drive flow vs. total core flow
- core plate flow vs. total core flow
- recirculation flow control valve position vs. loop drive flow
- jet pump readings vs. loop drive flow

Criteria

The performance curves used in conjunction with reactor recirculation system flow and differential pressure data will establish baseline data to determine if possible jet pump or recirculation pump degradation exists.

The established baseline performance curves will also be used to verify jet pump operability to determine if jet pump anomalies exist.

Results and Discussion

Reactor Recirculation (RR) data was collected from various power drops after L1C11 start up. Data was obtained from computer points for all the points of interest to evaluate the RR System performance. A small shift in the performance of Jet Pumps 19 and 20 was noted (due to the changes made to the #19 jet pump during L1R10, WO# 659181-01). The data collected was used to generate new curves for L1C11. Therefore, the RR system relationship curves were updated and documented in WO# 480822-01.