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

LaSalle County Station, Unit 2
Facility Operating License No. NPF-18
NRC Docket No. 50-374

Subject: LaSalle 2 Cycle 14 Startup Test Report Summary

Reference: Letter from Christopher Gratton, Sr., U.S. Nuclear Regulatory Commission to Michael Pacilio, Exelon Nuclear, dated September 16, 2010 "LaSalle County Station Units 1 and 2 - Issuance of Amendments Re: Measurement Uncertainty Recapture Power Uprate"

Enclosed for your information is the LaSalle County Station (LSCS) Unit 2 Cycle 14 Startup Test Report. This report is submitted in accordance with Technical Requirements Manual Section 5.0.b.

LaSalle County Station Unit 2 Cycle 14 commenced operation on March 7, 2011, following a refueling and maintenance outage. The Unit 2 Cycle 14 core loading consisted of 312 fresh AREVA Atrium-10 fuel bundles, 311 once-burned AREVA Atrium-10 fuel bundles, six once-burned AREVA Atrium-10XM fuel bundles, and 135 twice-burned AREVA Atrium-10 fuel bundles. Also installed in the Unit 2 Cycle 14 reactor were four new GE/Reuter-Stokes NA-300 Local Power Range Monitors (LPRMs), 34 new Westinghouse CR82M-1 Control Rod blades, six new General Electric Marathon C+ Control Rod blades, one used General Electric Duralife-215 Control Rod blade, and five used General Electric Marathon C+ Control Rod blades.

Attachment 1 contains 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
- Core Power Distribution Symmetry Analysis
- Reactor Recirculation System Performance

Also completed during the LaSalle 2 Cycle 14 Startup was the Measurement Uncertainty Recapture (MUR) Power Uprate (PU) Startup Testing. The referenced license amendment revised the Operating License and Technical Specification (TS) to implement an increase of approximately 1.65 percent in rated thermal power from the current licensed thermal power of 3489 megawatts thermal (MWt) to 3546 MWt.

The changes were based on increased feedwater flow measurement accuracy achieved by utilizing the Cameron International CheckPlus ™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation.

LSCS Unit 2 Cycle 14 MUR PU commenced operation on March 13, 2011 following successful testing conducted on March 13, 2011 per LSCS procedure LST-2010-009 "Unit 2 MUR Power Uprate Ascension Testing." A summary of LST-2010-009 is contained in Attachment 2.

All test data were 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. Terrence W. Simpkin, Regulatory Assurance Manager, at (815) 415-2800.

Respectfully,

David P. Rhoades
Site Vice President
LaSalle County Station

Attachments

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 2 Cycle 14 operation.

Criteria

The as-loaded core must conform to the cycle core design used by the Core Management Organization (AREVA & 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.

Conformance to the cycle core design will be documented by a permanent core serial number map signed by the audit participants.

Results and Discussion

Core verification was performed concurrently with core load and shuffle per NF-AA-330-1001 "Core Verification Guideline." The Unit 2 Cycle 14 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 and compared to the appropriate core loading map and Cycle Management Report. On March 2, 2011, the core was verified as being properly loaded and consistent with the LaSalle 2 Cycle 14 Core Loading Plan per Transmittal of Design Information (TODI) # NF1100041, Revision 2. Core verification was documented in Work Order (WO)# 1214746-01.

Single Rod Subcritical Check

Purpose

The purpose of this test is to demonstrate that the Unit 2 Cycle 14 core will remain subcritical upon the withdrawal of the analytically determined strongest control rod.

Criteria

In accordance with LTP-1600-30 "Single Rod Subcritical Check," the core must remain subcritical, with no significant increase in SRM readings and with the analytically determined strongest rod fully withdrawn.

Results and Discussion

The analytically determined strongest rod for the beginning Unit 2 Cycle 14 was determined by Nuclear Fuels to be control rod 22-15 per TODI# NF1100050, Revision 0. On March 3, 2011, with a Unit 2 moderator temperature of 88.3 °F, control rod 22-15 was withdrawn to the full out position (48) and the core remained subcritical with no significant increase in SRM readings. This information is documented in WO# 01321510-01.

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

Appropriate acceptance criteria are provided in LOS-RD-SR7 "Channel Interference Monitoring," and include limits on rod settle times (less than seven seconds), and if necessary, scram times from position 45 to position 05 and full stroke insertion time criteria (dependent on seal leakage and normal insertion time).

The control rod testing population will consist of control rods selected for testing per Reactor Engineering, Operations, and Nuclear Fuels.

Results and Discussion

CRD friction testing commenced after the completion of the core load verification and single rod subcritical check. All 185 control rods were tested and met the appropriate acceptance criteria (the settle times of all rods were less than 3.0 seconds). The testing was completed on March 4, 2011 and is documented in WO# 01214512-05.

Control Rod Drive Timing

Purpose

The purpose of this test is to check and set the insert and withdrawal speeds of the Control Rod Drives (CRDs).

Criteria

LOS-RD-SR5 "Control Rod Drive Timing" acceptance criteria for the withdraw times (full-in to full-out) and insert times (full-out to full-in) is between 40 and 60 seconds.

Results and Discussion

Control rod timing per LOS-RD-SR5 was performed for all 185 CRDs on March 4-5, 2011. All rods were within the FSAR allowed withdrawal speed. The testing was documented in WO# 01214486.

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 and all other control rods fully inserted.

Criteria

In accordance with LTS-1100-1 “Shutdown Margin Determination” and Technical Specifications, 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 14 is 0.00% $\Delta k/k$ per ANP-2987(P) “Startup and Operations Report LaSalle Unit 2 Cycle 14,” Revision 0, transmitted by NF TODI# NF1100081, 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 14 critical occurred on March 6, 2011 on Control Rod 34-55 at position 24, using an A-2 sequence. The moderator temperature was 155 °F and the reactor period was 216 seconds. Using control rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by Nuclear Fuels in TODI# NF1100081, Revision 0, the SDM was determined to be 1.415% $\Delta k/k$. This was documented in LTS-1100-1, Attachment A and WO 01403603-01. 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 NF-AB-715 "Critical Predictions with Powerplex III," NF-AB-760 "Reactivity Anomaly Determination," and 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% Δk/k at full power steady state conditions. If the difference exceeds 1% Δk/k, 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 14 Startup Test Program. One reactivity anomaly calculation is from the in-sequence critical and the other is from steady state, equilibrium conditions at approximately 100% full power.

The initial critical occurred on March 6, 2011 on Control Rod 34-55 at position 24, using an A-2 sequence. The moderator temperature was 155° F and the reactor period was 216 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 0.102% Δk/k of the predicted critical. The anomaly determined is within the 1% Δk/k required for beginning of cycle (BOC) conditions as stated in NF-AB-715. This was documented in NF-AB-715, Attachment 3 and WO# 1403600.

The reactivity anomaly calculation for full power steady state operation was performed on March 11, 2011. The data used was from 99.8% power at a cycle exposure of 73.7 MWD/MT at equilibrium conditions. The expected k_{eff} supplied by Nuclear Fuels was 1.0009. The actual k_{eff} was 1.0032. The resulting anomaly was 0.23% Δk/k. This value is within the 1% Δk/k criteria of Technical Specifications. This was documented in NF-AB-760 Attachment 1 and WO# 1403604.

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

In accordance with LOS-RD-SR12 "Scram Insertion Times" and Technical Specifications, 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 two 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	Scram Time to Notch Indicated (seconds)
45	0.52
39	0.80
25	1.77
05	3.20

Results and Discussion

Scram testing was successfully completed on March 3, 2011 per WO# 1214561. All 185 rods were scram timed during the reactor pressure vessel leakage testing (Hydro) prior to startup. No rods were classified SLOW or INOPERABLE, and the results of the testing are given below.

Notch Position	Core Average Scram Times of all CRDs (sec)
45	0.309
39	0.604
25	1.308
05	2.345

These results also meet the "Nominal" Scram Speeds referenced in the Unit 2 Cycle 14 Core Operating Limits Report (TRM Appendix I).

Core Power Distribution Symmetry Analysis

Purpose

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

Criteria

In accordance with NF-AB-716, 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 March 21, 2011, at approximately 100% power. The TIP set was performed with four of five TIP machines operable. All traces were obtained for the operable TIP machines. The χ^2 value was 13.00, which satisfies the test criteria of 36.19 for 19 pairs. The maximum deviation between symmetrical TIP pairs was 10.26%, which is within the 25% acceptance criteria. This was documented in WO# 1392106.

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

In accordance with LTP-1600-13 "Reactor Recirculation System Performance" and Technical Specifications, 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 data was collected during the L2C14 startup. Data was obtained from computer points for all the points of interest to evaluate the RR System performance. No significant changes from L2C13 were noted in the L2C14 RR performance curves. This was completed on April 1, 2011. This is documented in WO#1265873.

Measurement Uncertainty Recapture Power Uprate

Summary

The license amendment associated with the Measurement Uncertainty Recapture (MUR) Power Uprate (PU) revised the Operating License and Technical Specification (TS) for implementation of an increase of approximately 1.65 percent in rated thermal power from a thermal power of 3489 megawatts thermal (MWt) to 3546 MWt. This power increase was evaluated for its impact on the operating margin of various plant systems, structures, and components (SSCs) due to increased flow, temperature, and electrical output. An Operational and Technical Decision Making (OTDM) associated with Plant Margins was developed to identify and track SSCs with potentially degraded margin resulting from the MUR PU. To address any potential issues, MUR Task Reports were developed by General Electric Hitachi (GEH), Sargent & Lundy (S&L), Areva and General Electric Energy Services (GEES) for all impacted SSCs.

During implementation of the license amendment, all potential margin issues contained in the MUR Task Reports were again reviewed and re-evaluated against the results of the MUR PU testing, which was performed on March 13, 2011. This review took into consideration data obtained during testing, which are documented in LST-2010-009 "Unit 2 MUR Power Uprate Ascension Testing", and was also based on general observations from test participants (i.e., from Operations, Engineering, etc.). Re-evaluation of each of the potential margin issues determined that in some cases, further performance monitoring of some of the affected SSCs is recommended. Special station procedure LLP-2011-001 "Unit 2 MUR Power Uprate Performance Monitoring Plan," will be used for this purpose and this will augment normal system monitoring with a more focused long-term monitoring of SSCs with high impact on reliability.

Evaluation

<u>Feedwater Heaters</u>
There was no unexpected plant response/performance data noted during MUR PU testing. However, the 4th and 5th stage Feedwater (FW) Heaters will experience increased shell side flow, which is in excess of Heat Exchange Institute (HEI) guidelines. This could lead to elevated tube vibration due to vortex shedding. In addition, the 5th stage FW Heaters have high steam velocities such that they will have low margin that could potentially cause additional wear on steam inlet nozzles. Each of these issues will be managed with increased Eddy current tube inspections on the 4th and 5th stage heaters including inspection of two nozzles on the 5th stage heaters to identify, in a timely manner, any adverse wear trend.

<u>Main Condenser</u>
Margin to the main condenser high backpressure alarm will be reduced in the summer months as a result of the MUR PU. Although the testing conditions did not identify current adverse conditions, cleaning of the condenser tubes in L2R13 (Unit 2 refueling outage) would likely mitigate this concern; however, further evaluation is needed and a modification will be will be developed to raise the main condenser high backpressure setpoint to 6.5 HGa.

Reactor Pressure Vessel Flow Induced Vibration
No anomalies in reactor jet pump performance were noted during MUR PU testing. Jet pump main wedge wear is anticipated to continue or slightly increase as a result of the MUR PU. However, monitoring jet pump wedge wear within existing processes will continue (i.e., in-vessel inspection program). Long-term trends in cobalt concentration will also be monitored for indications of abnormal wear.

High Pressure (HP) Turbine Steam Control Valve Margin
The main Turbine Control Valve (TCV) positions were seen to be as predicted during the testing. The HP nozzle block was resized during L2R13, which kept the control point approximately the same (an increase of about 1.3% prior to L2R13). The control valve regulation point is about 55% open on Unit 2 versus 59% on Unit 1. This will continue to be monitored under the performance monitoring procedure LLP-2011-001.

Moisture Separator Reheater (MSR)
No significant data anomalies were noted during the MUR PU ascension testing. However, the existing degraded material condition of the MSRs may have an impact on long-term operation under increased rated power level. Both MSRs will be inspected during preceding outages following MUR PU implementation, and repair kits will be installed if necessary to maintain margin under MUR PU operation. Inspection of MSRs will continue in accordance with current plant practices and frequency.

Steam Separator/Dryer Performance
The testing limitations for moisture carryover were not approached based on chemistry samples taken during the testing. The MUR PU ascension test determined a carryover value of 0.012%, which corresponds to the carryover prior to the MUR PU implementation and is lower than the carryover value before L2R13 of 0.015%. The station will continue to periodically monitor and trend moisture carryover within the existing process.

Condensate Polisher (CP) Flow Capacity
The flow rate across the filter demineralizers was marginally acceptable under pre-MUR PU and will be slightly more limiting under MUR PU operations. System flow requirements can be met with six of the total seven CP vessels in service with essentially no remaining margin. This will be closely monitored under LLP-2011-001.

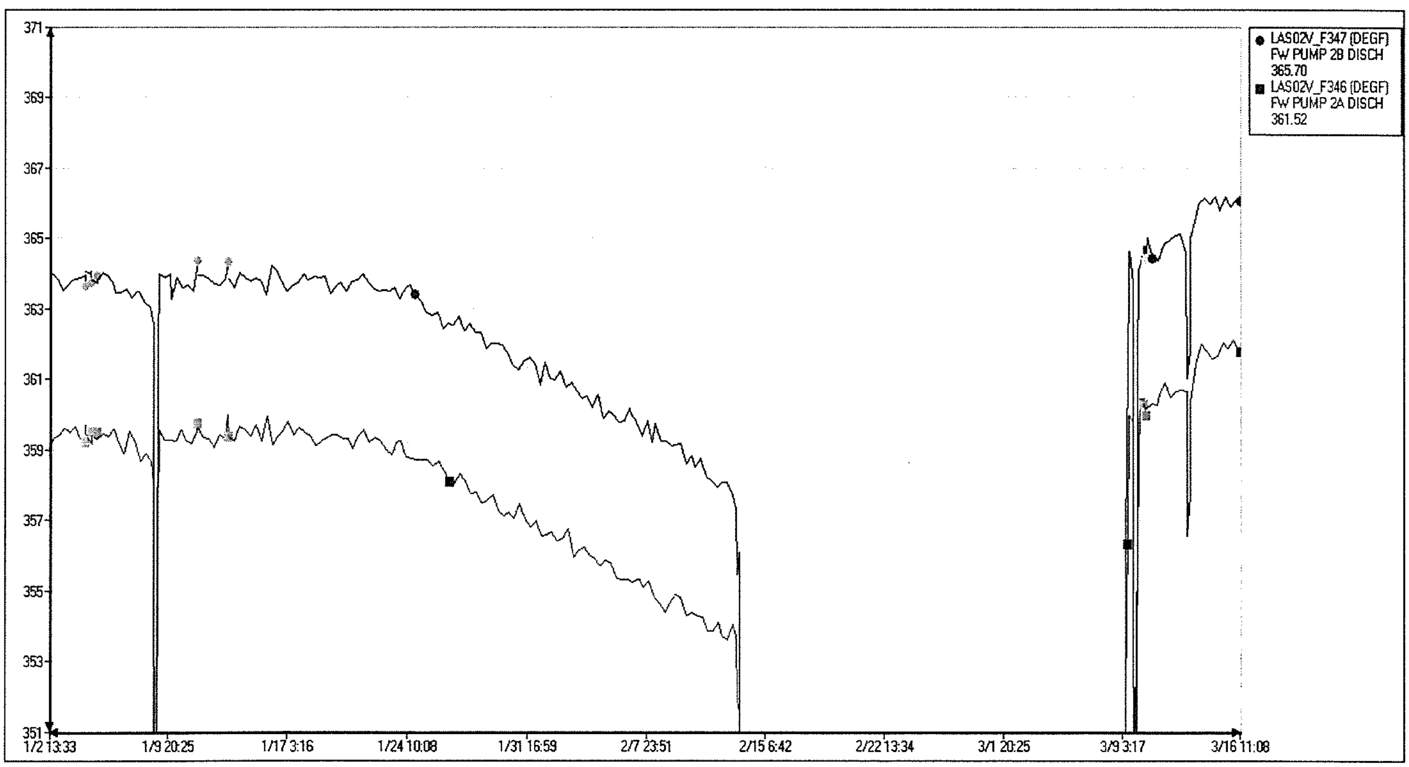
Steam Jet Air Ejector (SJAE) Temperature
The MUR PU ascension testing did not reveal any anomalous data. However, to ensure proper SJAE summer operation, condensate temperature should remain below 134°F. It is possible that this temperature could be approached in hot summer conditions under the MUR PU power level. SJAE performance, when condensate temperatures approach 134°F will be closely and continuously monitored during summer operations.

Test Exceptions

The only plant performance data, which was seen to trigger a Test Exception (TE #5) was with the 2B Turbine Driven Reactor Feed Pump (TDRFP) outlet temperature computer point (F347).

During performance of steps F.3.19 and F.4.19 of the testing procedure, LST-2010-009, temperatures of 365.3 °F and 365.7 °F were obtained, respectively, which exceeded the level 2 criteria. This elevated temperature was accepted as a test exception and the condition was documented in LSCS' Corrective Action Program under Issue Report (IR) Number 1186901. An action that has resulted from this IR is to have the FW System Engineer evaluate the appropriate alarm setpoint. Subsequently, the evaluation determined that the outlet temperature was approximately 1 °F higher than what the temperature was before L2R13, and that the implementation of the MUR PU further increased this outlet temperature by another 1 °F. This same trend was observed on the 2A TDRFP outlet temperature computer point (F346), and both temperatures have trended steady and with one another since the MUR PU implementation (see table and corresponding graph below). The evaluation has concluded that no further actions are needed with regards to this test exception.

	2A TDRFP Outlet Temp	2B TDRFP Outlet Temp
	Pt. F346	Pt. F347
Pre-L2R13	359.6	363.8
Post-L2R13	360.6	364.8
Post MUR	361.7	365.8



Conclusion

Overall plant performance during MUR PU ascension testing was as expected and predicted with no significant performance concerns, and the procedure LST-2010-009, as written, was performed successfully. All of the above noted SSCs will continue to be thoroughly monitored through special procedure LLP-2011-001, which will be used primarily for this purpose. The use of this SSC monitoring procedure will augment normal system monitoring with a more focused long-term monitoring of SSCs with high impact on reliability.