ATTACHMENT A

LaSalle Unit 1 Cycle 6 Startup Test Report

SUMMARY

LaSalle Unit 1 Cycle 6 began commercial power operation on January 30, 1993 following a refueling and maintenance outage. The Unit 1 Cycle 6 core loading consisted of 200 fresh GE8x8NB (GE9B) fuel bundles and 564 reload bundles. Twenty-two Local Power Range Monitor (LPRM) were replaced with General Electric NA-300 LPRM strings. Two control blades were replaced, one with a General Electric Standard Control Blade and the other an A.B.B CR-82B Control Blade.

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 March 7, 1993 with the final verification of all LPRM strings. Recirculation System Performance was completed on March 16, 1993. 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 Attachment B 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 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, Unit Startup Test Program. Data from each startup is available at LaSalle Station.

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ATTACHMENT B

LaSalle County Nuclear Power Station

Unit 1 Cycle 6 Startup Test Reports

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 6 operation.

CRITERIA

The as-loaded core must conform to the cycle core design used by the Core Management Organization (Commonwealth Edison 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. 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 6 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 20, 1992 the core was verified as being properly loaded and consistent with the Commonwealth Edison Nuclear Fuel Services Cycle 6 Cycle Management Report and the Final Station Use Loading Plan. No discrepancies were noted during the inventory. On December 22, 1992 twelve fuel assemblies were re-verified because they had been re-channeled. The videotapes were reviewed by the Lead Nuclear Engineer and a Nuclear Engineer to reverify all bundle ID's, orientation, and seating.

A serial number inventory was also performed on the Unit 1 fuel pool on January 5, 1993 to verify that the fuel pool contained the proper bundles. The fuel pool contained no bundles which should have been loaded into the Unit 1 reactor.

LTP-1600-30, Single Rod Subcritical Check

PURPOSE

The purpose of this test is to demonstrate that the Unit 1 Cycle 6 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 6 of Unit 1 was determined by Commonwealth Edison's Nuclear Fuel Services to be rod 46-47. On December 20, 1992 with a Unit 1 moderator temperature of 92 degrees Fahrenheit (as read from 1E12R601, Shutdown Cooling Suction Header), rod 46-47 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 92° Fahrenheit. This test was satisfactorily performed again on December 21, 1992 because moderator temperature dropped to 84° Fahrenheit. This information is documented on LTP-1600-30, Attachment 2, Unit Instructions for Single Control Rod Movement, of which a copy was given to the Unit 1 NSO and the Shift Engineer.

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 was commenced after the completion of the core load verification and single rod subcritical check, and was completed on December 23, 1992. Continuous insert friction traces were obtained for all 185 CRDs. Control rods 02-43, 42-59, and 58-31 exhibited high friction during the test. The surrounding four bundles were rechanneled and the control rods were 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 its 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). This change might avoid adjustments of the CRD velocity during rated reactor operation.

RESULTS AND DISCUSSION

All CRDs were tested between 01-02-93 and 01-22-93. All control rod drives demonstrated normal times during the performance of this test. A coupling check was also successfully performed on each drive during the timing process.

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%ΔK/K (0.38 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 6 is 0.0 % K/K, with the minimum SDM occurring at 0.0 MWD/ST into the cycle.

RESULTS AND DISCUSSION

On January 23, 1993 the local SDM demonstration was successfully performed using control rods 50-43 and 46-47. Control rod 50-43 is diagonally adjacent to 46-47, the strongest rod at beginning-of cycle. Commonwealth Edison's Nuclear Fuel Services (NFS) provided, in the Cycle Startup Package, rod worth information (for control rods 46-47 and diagonally adjacent rods 50-43 and 42-51) and moderator temperature reactivity corrections to support this test. Using the NFS supplied information, it was determined that with control rod 46-47 at position 48 and rod 50-43 at position 14, a moderator temperature of 140°F, and the reactor subcritical, a SDM of 0.564%ΔK/K was demonstrated. The SDM demonstrated exceeded the 0.38%ΔK/K required to satisfy Technical Specification 3.1.1, and maintained sufficient margin to the NFS calculated SDM for the core at beginning-of-cycle (1.700%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 insequence 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 6 is 0.0%ΔK/K, so a SDM of 0.38%ΔK/K must be demonstrated.

RESULTS AND DISCUSSION

The beginning-of-cycle SDM was successfully determined from the initial critical data. The initial Cycle 6 critical occurred on January 23, 1993 on control rod 34-39 at position 14, using an A-2 sequence. The moderator temperature was 148.4.°F and the reactor period was 198 seconds. Using rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by Commonwealth Edison's Nuclear Fuel Services (NFS) in the Cycle Startup Package, the beginning-of-cycle SDM was determined to be 2.295%ΔK/K (see Table 1). The SDM demonstrated exceeded the 0.38%ΔK/K required to satisfy Technical Specification 3.1.1.

TABLE 1

SHUTDOWN MARGIN CALCULATION

Critical Rod = 34-39 @ 14

Worth of Strongest Rod = 0.02786ΔK/K (1)

Worth of Control Rods Withdrawn to Obtain Criticality:

24 Group 1 rods at 48	= 0.03607ΔK/K	(2)
18 Group 2 rods at 48	= 0.01580ΔK/K	(3)
1 Group 2 rod at 14	= 0.000740ΔK/K	(4)

Temperature Correction = -0.00150ΔK/K (5)
for T_{mod} = 148.4°F

Period Correction = 0.00030ΔK/K (6)
for Period = 198 seconds

Shutdown Margin Keff:
SDM Keff - 1.0000 + (1) - (2) - (3) - (4) - (5) + (6)
- 0.977050ΔK/K

SDM = (1.000 - (SDM Keff)) * 100 = 2.2950%ΔK/R

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%ΔK/K. If the difference does exceed 1%ΔK/K, the Core Management Engineers (Commonwealth Edison's 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 1 Cycle 6 Startup Test Program, one from the initial critical, and a second from steady-state, equilibrium conditions at approximately 100 percent of full power.

The initial critical occurred on January 23, 1993, with control rod 34-39 at position 14, using an A-2 sequence. The moderator temperature was 148.4°F and the reactor period was 198 seconds. Using rod worth information, moderator temperature reactivity corrections, and period reactivity corrections supplied by Commonwealth Edison's Nuclear Fuel Services (in the Cycle Startup Package), the actual critical was determined to be within 0.595%ΔK/K of the predicted critical (see Table 2). The difference determined is within the 1% a ΔK/K criteria of Technical Specification 3.1.2.

The reactivity anomaly calculation for power operation was performed using data from March 19, 1993 with Unit 1 at 99.8% power at a cycle exposure of 574.9 MWD/ST, at equilibrium conditions. The predicted notch inventory from the vendor supplied data was 357 notches. The actual notch inventory was 355 notches. Using the notch worth provided by the vendor, the resulting anomaly was 0.005%ΔK/K. This value is within the 1% ΔK/K criteria of Technical Specification 3.1.2.

TABLE 2

INITIAL CRITICALITY COMPARISON CALCULATIONS

ITEM	Δ K/K
Keff with all rods in at 68°F	= 0.95514 *
Reactivity inserted by 24 group 1 rods at position 48	= 0.03607 *
Reactivity inserted by 18 group 2 rods at position 48	= 0.01580 *
Reactivity inserted by 1 group 2 rod at position 14	- 0.00074 *
Predicted Keff at actual critical rod pattern (68°F)	= 1.00775
Reactivity associated with the measured reactor period (period correction for 198 second period)	= 0.0003 *
Reactivity associated with moderator temperature (148.4°F actual, 68°F predicted)	= -0.0015 *
Reactivity Anomaly = [(predicted Keff - 1) - (period correction) + (temperature correction)] * 100%	= 0.595%ΔK/K

* "LaSalle Unit 1 Cycle 6 Startup Package", supplied by Commonwealth Edison's Nuclear Fuel Services.

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 deenergization 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

SEE ATTACHED

Position	Average Scram Times of all CRDs (secs.)	Maximum Average Scram Times in a Two-By-Two Array (secs.)
45	0.325	0.357
39	0.623	0.653
25	1.337	1.384
05	2.472	2.501

Average (position 39) for Minimum Critical Power Ratio Limit determination = 0.623.

PURPOSE

During the course of the last refuel outage (L1R05) scram insertion time testing was performed on all maintenance drives (110) during the vessel inservice leakage test (HYDRO). The remaining drives (75) were scram time tested with the reactor at approximately 25% power. During the course of scram time testing at 25% reactor power ten percent of those drives tested during the vessel HYDRO (11) were retested. The purpose of this retest is to verify that the times generated during the vessel HYDRO were conservative to those times generated with the reactor at power.

This report will address the results achieved during these tests. It will further define Tech Staffs position on continuing the process of scram time testing during the vessel inservice leakage test for all maintenance and non-maintenance control rod drives.

DISCUSSION

Scram insertion time testing is a Tech Spec requirement (TS 3.1.3.2) and performed to satisfy Tech Spec surveillance requirements of 4.1.3.2. This requires the maximum scram insertion time to notch position "05" of each control rod shall not exceed 7.0 seconds. Additional time criteria established for the control rod is governed by Tech Specs 3.1.3.3 and 3.1.3.4. These Tech Specs provide the time requirement for the average scram insertion times for all operable control rods and the average scram insertion time for the three fastest control rods in each group of four control rods arranged in a two-by-two array. An additional crucial factor governed by Tech Specs is Tau-average. This variable is the sum of the scram insertion times to notch position "39" for all 185 control rods divided by 185 (TS 3.2.3 as found in LAP-1200-16). These Tech Spec requirements were used in determining the most conservative times in the comparing the scram insertion times received during hydro and while at power. The usual method of performing scram insertion time testing on maintenance drives following a refuel outage is to perform the test on all 185 control rods after the reactor has achieved approximately 25% power.

It was discovered that scram insertion testing during the vessel hydrostatic leak test was a common practice throughout the nuclear industry. It was decided that scram time testing would be done during the vessel hydro at LaSalle. General Electric was contacted with several questions (LaSalle Letter #TSL 91-169) regarding possible consequences of taking a full reactor scram while in the process of performing the scram insertion time procedure. Additional questions were inquired about any possible consequences to Tech Spec control rod scram insertion time requirements. GE addressed these questions (CHRON #182877) and stated that the change in scram performance during the vessel HYDRO was insignificant. The control rod will meet the required time specifications under all conditions of plant operation.

Station Operations Department, Planning Department, and Tech Staff concurred that scram insertion time testing would be performed during the course of the Unit 1 refuel outage hydrostatic leakage test but only on the maintenance drives (110). This would leave 75 drives to scram time test with the reactor at approximately 25% power. These departments agreed that an additional 10% of those rods tested the vessel hydro would be retested at power. These ten percent would be used to compare the times to verify that the scram times achieved during the vessel hydro were conservative to those achieved with the reactor at 25% power.

The following is an analysis of the scram insertion times achieved during the hydro and times achieved with the reactor at power. The main focus will be on the important notch positions "39" used in the calculation of Tau-average for the core, and "05", used in Tech Spec 3.1.3.2.

RESULTS

A look at the data in Table 1 shows that seven of the ten rods tested yielded scram insertion times that definitely have more conservative times during hydro conditions than at power conditions (30-31, 34-03, 58-43, 42-55, 42-59, 22-35, 46-43). The hydro scram insertion times for 18-55 were also considered to be the conservative times since the overall scram insertion time to notch position "05" was greater during the hydro than at power (2.416 vs 2.384 a change of +0.032 sec). The scram insertion time to position "39" was not significantly different for either plant condition (0.616 vs 0.624 a change of -0.008 sec).

Two rods had conservative scram insertion times with reactor at power rather than during the hydro (54-27, 46-11). These times can be explained by reviewing other variables involved in the scram insertion times (i.e. reactor pressure and accumulator pressure).

Rod 54-27 has scram insertion times identical up to notch position "25" for both test conditions, however, the rod was slower from notch "25" to notch "05" for the scram insertion test performed with the reactor at power. This time difference is not significant since the rod was well below the 7 second time criteria established by Tech Spec 3.1.3.2 as well as the time criteria for the average time for all 185 control rods, 3.70 seconds, established by Tech Spec 3.1.3.4. This time difference can be explained by reviewing how the control rod drive mechanism (CRDM) and its associated hydraulic control unit work (HCU) during a scram.

The HCU contains high pressure water in a scram accumulator and scram valves to scram its associated control rod. The scram inlet valve directs water from the scram accumulator to the CRDM underpiston area and the scram outlet valve directs water from the overpiston area of the CRDM to the scram discharge volume (SDV). When a control rod is scrammed, high pressure water is applied to the the underpiston area and the overpiston area is vented to the SDV. As the rod moves in, pressure at the underpiston area decreases as nitrogen in the nitrogen cylinder expands into the scram accumulator. When pressure at the underpiston area decrease to less than reactor pressure, a ball check valve, located at the entrance of the underpiston area, lifts to open a flow path from the reactor to the underpiston area. This means that if nitrogen pressure decreases below reactor pressure, reactor pressure will take over as the driving force of the control rod.

Rod 54-27 had an identical accumulator pressure prior to receiving the scram signal during both the hydro scram test and the scram test at power (1100 psig). The reactor pressure however was greater during the Hydro scram test (973 vs 952 psig). This would lead to the results shown in Table 1, as during the initial phase of the scram the accumulator pressure is the driving force. Since the accumulator pressure was the same, 1100 psig, during both scram insertion tests, the times were the same during the initial portions of the stroke, position "45" to "25". These positions correspond to approximately half the length of travel for the control rod. During the course of the scram the nitrogen pressure decreased and reactor pressure became the driving force of the control rod. The fact that the hydro reactor pressure was greater than reactor pressure while at power led to the time being faster at the end of the stroke of the piston.

It is not known why the scram insertion times for rod 46-11 are so different from the other rods tested. There are many reasons for a rod to be slow one time and not another. The operator may not have placed the scram switches in the test position exactly at the same time, any hesitation would cause a slow initiation response time. There may have been a small particle of dirt in the air line which blocked the small air ports in the scram pilot valves momentarily. The fact that one rod out of the ten had scram insertion times greater during the scram time test with the reactor at power than for the scram time test during hydro conditions does not lead to the conclusion that by scram time testing during the hydro a true Tau-average and scram times can not be achieved. The following are calculations of the average scram times for all 10 rods to each position. These calculations will show that the scram insertion times taken during the reactor hydro will actually provide conservative scram times.

Tau-average calculated for the 10 drives used in the comparison is as follows:

$$\text{Tau-ave} = \frac{\text{sum of times at notch position 39}}{\text{total number rods timed}}$$

$$\text{Tau-ave(HYDRO)} = 6.424 / 10 = 0.642$$

$$\text{Tau-ave(Power)} = 6.264 / 10 = 0.626$$

Average Scram Time To Position "45"

HYDRO	AT POWER
0.339	0.330

Average Scram Time to Position "39"

HYDRO	AT POWER
0.642	0.626

Average Scram Time To Position "25"

HYDRO	AT POWER
1.367	1.338

Average Scram Time to Position "05"

HYDRO	AT POWER
2.477	2.434

CONCLUSION

The results of these average times indicates that the scram insertion times received during vessel hydro conditions were conservative to all notch positions. Ten rods were tested and used as comparison of these ten, nine had conservative or identical times during vessel hydro conditions. These facts and the position of GE stated in the 3/18/92 letter from A. Tsang to M. Wrightsman (CHRON #182877) indicate that the change in scram time performance is insignificant whether performed during the hydrostatic vessel leakage test or at rated power conditions. It is Tech Staffs position that Scram Insertion Time testing should be included in the outage schedule and affixed to the Vessel Hydrostatic Leakage Test. LOP-NB-01 has been revised to allow for this configuration, the special procedure written to perform scram insertion time testing during the hydro (LLP-92-215) will be converted to a LaSalle Technical Staff test (LTS) for future test.

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 three full core TIP sets (OD-1). The initial TIP set was performed on February 6, 1993 at 75% power, a second on March 18, 1993 at approximately 100% power. The average total TIP uncertainty from the three data sets was 3.588%, satisfying the criteria of the test (less than 8.7%). The average delta deviation of symetric strings was 3.97%.

Table 3 lists the symmetrical TIP pairs, their core locations, and their respective average deviations at 75 and 100% power. The maximum deviation between symmetrical TIP pairs was 15.54% for TIP pair 56-25, 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 BASF ratio of TIP pairs by:

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

Next, the standard deviation (expressed as a percentage) of these ratios is calculated by the following equation:

$$\sigma_R\ (\%) = \left[\sum_{i=5}^{22} \sum_{j=1}^{N} R_{i,j} \right]^{1/2} *100$$

The total TIP uncertainty (%) is calculated by dividing $\sigma_R(\%)\ \sqrt{2}$ 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 from the OD-1 data set (from 2-06-93 at 75% power and 3-18-93 at 100% power).

Symmetrical TIP Pair Numbers (Core Location) a	b	Absolute Difference of Base#	Percent TIP Pair Deviation*
1 (16-09)	6 (08-17)	3.03	3.10
2 (24-09)	13 (08-25)	1.74	1.58
3 (32-09)	20 (08-33)	1.14	1.06
4 (40-09)	27 (08-41)	6.41	6.16
5 (48-09)	34 (08-49)	1.52	2.90
8 (24-17)	14 (16-25)	2.36	1.96
9 (32-17)	21 (16-33)	4.79	4.16
10 (40-17)	28 (16-41)	2.38	2.00
11 (48-17)	35 (16-49)	0.78	0.71
12 (56-17)	40 (16-57)	1.75	1 75
16 (32-25)	22 (24-33)	1.55	1 30
17 (40-25)	29 (24-41)	6.06	5.14
18 (48-25)	36 (24-49)	8.86	7.35
19 (56-25)	41 (24-57)	11.54	15.54
24 (40-33)	30 (32-41)	5.33	4.52
25 (48-33)	37 (32-49)	0.94	0.80
26 (56-33)	42 (32-57)	2.90	3.61
32 (48-41)	38 (40-49)	3.90	3.20
33 (56-41)	43 (40-57)	5.84	8.54

where : Absolute Difference of BASE = $\overline{BASE_a} - \overline{BASE_b}$

and $\overline{BASE_i} = \frac{1}{18} \Sigma \ BASE_i \ (K)$

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

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ATTACHMENT C

List of References

1. Cycle Management Report, LaSalle 1 Cycle 6. DRF Number LS1-0011 Volume 1.