

CYCLE 3 STARTUP TEST REPORT

Cycle 3 operations commenced December 5, 1978 with the withdrawal of the first control rod. The startup test program was conducted from December 3, 1978 through February 23, 1979, in accordance with Reactor Analyst Procedure (RAP) 7.1.17, titled Refuel Startup Program Revision 1. When reference is made to values of core thermal power and core flow, these are nominal values rather than exact percentages.

CONTROL ROD DRIVE TESTS

Control rod drive coupling checks were satisfactorily completed for all rods on December 11, 1978. In addition, the insert and withdrawal times for all rods were checked and adjusted as required.

Prior to reaching 40% rated core thermal power, control rod scram time testing was conducted in accordance with RAP 7.3.10 titled Control Rod Scram Time Evaluation Revision 2. This test requires that each control rod be scrammed from position 48 (full out) with reactor pressure >950 psig. The results of these tests are tabulated below.

Results:

<u>Control Rod Insertion (Percent)</u>	<u>Technical Specification (Seconds)</u>	<u>Average of 137 Rods (Seconds)</u>
5	0.375	0.28
20	0.900	0.74
50	2.000	1.53
90	3.500	2.62

Since all times for each amount of insertion for individual rods were less than the maximum allowed by the technical specifications, calculations were not performed for 2 X 2 arrays.

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SHUTDOWN MARGIN DEMONSTRATION

A shutdown margin (SDM) demonstration was performed December 5, 1978 in accordance with Reactor Analyst Procedure 7.3.9 Revision 1. The required SDM was $0.38\% \Delta k + R + \text{temperature defect}$. The fuel vendor has calculated that the most reactive point is at the beginning of the cycle, hence the value of R is 0.0. After calculating the temperature defect, the required SDM was determined to be $0.47\% \Delta k$. Sufficient control rods were withdrawn to demonstrate a SDM of $0.57\% \Delta k$.

INSEQUENCE CRITICAL

In order to eliminate one control rod sequence exchange during Cycle 3, the startup was conducted in sequence B rather than sequence A. It was necessary to withdraw five control rods (one of which was a low worth peripheral rod) more than estimated. This difference can be attributed to criticality occurring at a reactor water temperature of 174°F rather than 68°F , and a critical eigenvalue of 1.0093 instead of 1.006.

REACTIVITY ANOMALY CHECK

A comparison of the expected and actual control rod density was performed at 100% core thermal power (CTP) and 100% rated core flow. The control rod inventory was 282 notches which was in close agreement with the predicted value of 315 notches. The $\pm 1\%$ reactivity boundaries were 35 to 595 notches.

POWER DISTRIBUTION MEASUREMENTS

Core power distribution was monitored throughout the startup using the process computer. Following significant changes in control rod pattern and power level, a complete power distribution measurement was performed using the Traversing In-core Probe (TIP) system. Core parameters were maintained within technical specification limits.

TIP REPRODUCIBILITY

Successive TIP plots taken on each machine indicate a maximum difference of approximately 5.3%. This occurred adjacent to a fuel bundle spacer. In general, the plots were within 2% of each other.

CORE POWER SYMMETRY

Core power symmetry was checked at 25%, 50%, 75%, and 100% CTP. Mirror symmetric fuel assemblies checked at 100% CTP and 100% core flow using the process computer indicate a maximum difference of 4% for peripheral assemblies and less than 1% for interior, high power assemblies. Total TIP uncertainty is $\approx 5\%$, well below the 8.7% assumed by the vendor in the statistical analysis performed for the licensing topical report (NEDE-24011-P-A) for the reload fuel application. The highest individual asymmetries were observed in locations where the rod pattern was not rotationally symmetric. If the pattern had been rotationally symmetric, the uncertainty would have been even lower.

CORE LOADING

A copy of the final core loading is attached as Figure 1. Irradiated fuel returned to the core is designated EA or LJ5 or LJ6. There were 136 new fuel assemblies, designated LJB, (100 2.83 w/o U-235, 36 2.65 w/o U-235) loaded during the refueling outage. The new fuel was of the 8 X 8 R design with an active fuel length of 150 inches and received 100 mil channels.

Figure 2 shows the approximate irradiated bundle average exposure following refueling. A blank indicates a new fuel assembly.

All new fuel assemblies contain burnable poison in the form of GdO_3 . The concentration and location is proprietary to the fuel vendor.

Figure 3 shows the rod sequence control system (RSCS) designations for the A and B rod withdrawal sequences.

During the refueling operation, each fuel move was checked by an individual, other than the operator performing the move, and verified independently by a third individual. Two lines of communication were established between the refuel bridge and the control room. Following loading, a core verification was conducted (and video-taped and examined later by quality assurance personnel) to verify the correct placement and orientation of each assembly.

ADDITIONAL TESTS

1. Tests were performed in accordance with RAP 7.1.17 and F-ST-5C to verify that there is approximately one decade overlap between the source range monitor and intermediate range monitor (IRM) systems, and between the IRM and average power range monitor (APRM) systems.
2. Reactor core isolation cooling and high pressure coolant injection flow rate tests were performed in accordance with F-ST-24C and F-ST-4B and demonstrated compliance with the technical specifications.

3. Both the rod worth minimizer and rod sequence control system functioned properly during the startup test program.
4. Since some TIP tubing were replaced, the TIP alignment and logic limits were checked and adjusted as required.
5. New computer software was installed during the outage. Extensive testing was performed prior to and during the startup in accordance with the vendor's recommendations.
6. The APRM system was calibrated to core thermal power and satisfactorily tracked power changes.
7. Heat balances were calculated manually and used to verify the process computer calculations.
8. Process computer calculations of fuel assembly parameters, maximum average planar linear heat generation rate, minimum critical power ratio, and maximum fraction limiting power density compared satisfactorily with results obtained using off-line computer calculations.
9. RAP 7.3.18, "Pressure Regulator Tests," was performed satisfactorily when it was verified that an induced pressure transient of 10 psi was controlled by the electro-hydraulic control system pressure regulator. In addition, transfer from the primary to back-up pressure regulator was demonstrated following a simulated failure of the primary regulator.
10. RAP 7.3.7, "Core Flow Evaluation and Indication Calibration," was performed satisfactorily at 75% CTP. However, when performing the test at 100% CTP on February 2, 1979, operations and instrumentation and control personnel noted a step increase in the indicated core flow. The absolute value of the step increase was approximately 1 1/2 to 3% of indicated flow and this step increase resulted in a total indicated core flow of between 101.5% and 103% of rated flow. Operations personnel immediately reduced the recirculation pump speed until the indicated core flow was below 100% of rated flow.

Investigation revealed that the square root computation module for one of 20 jet pumps had failed at some earlier time. The first indication of this failure, however, was the step change in the output of the square root module which occurred when technicians "disturbed" the module by attaching test equipment to it. This disturbance caused the module output to return to its proper value, that is, the module began performing its square root function.

The square root computation module was replaced with an identical unit from inventory and the Core Flow Evaluation test was completed without further difficulty. This event was previously reported to the Nuclear Regulatory Commission in LER-79-10.

The test was repeated later at 100% CTP and 100% flow with satisfactory results.

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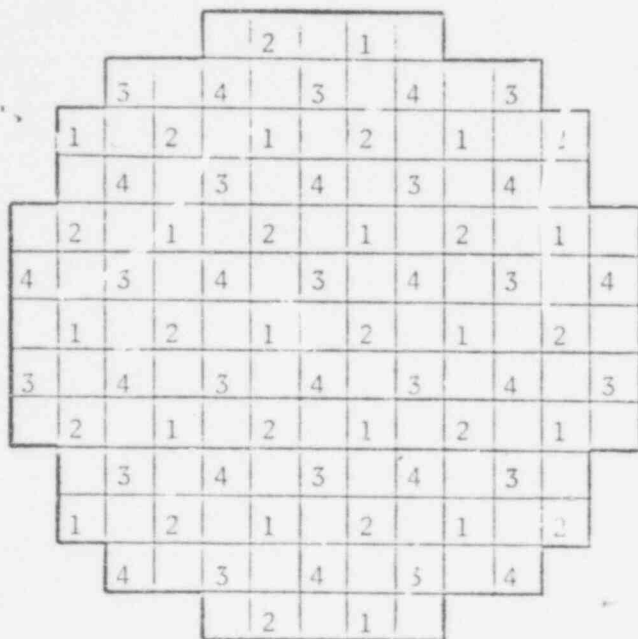
1 3 5 7 9 11 13 15 17 19 21 23 25 27 29 31 33 35 37 39 41 43 45 47 49 51

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FIGURE 1

ROD SEQUENCE CONTROL SYSTEM GROUP DESIGNATIONS

A SEQUENCE



02 06 10 14 18 22 26 30 34 38 42 46 50

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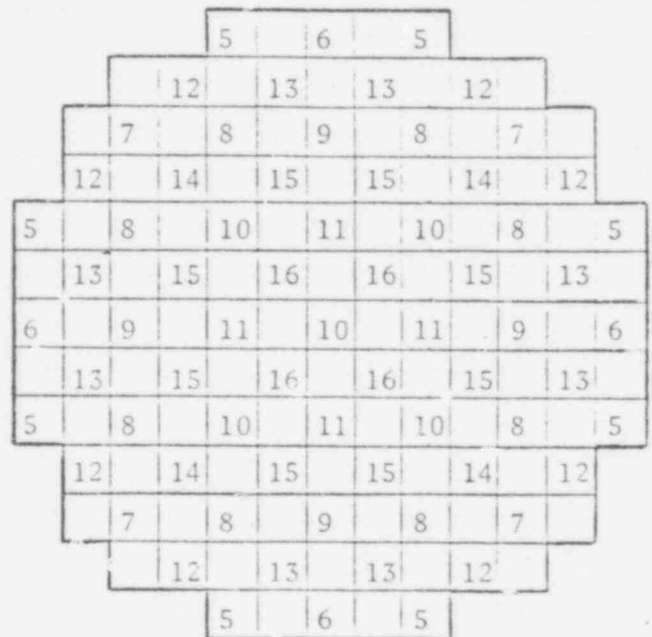
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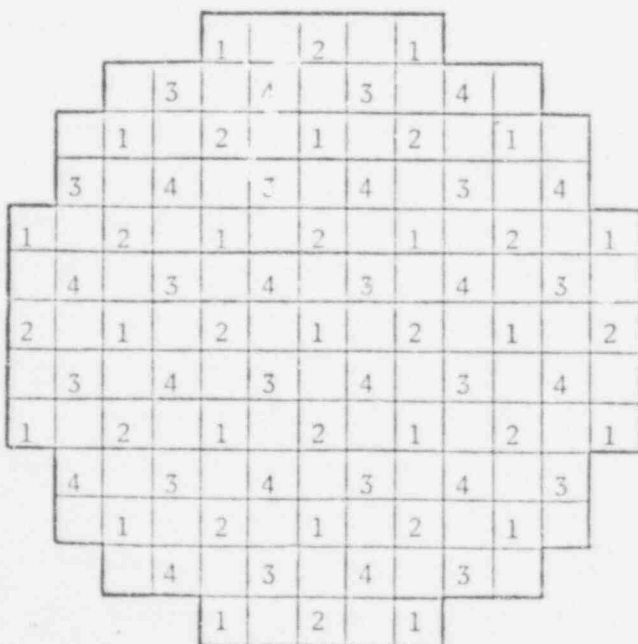
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02 06 10 14 18 22 26 30 34 38 42 46 50

B SEQUENCE



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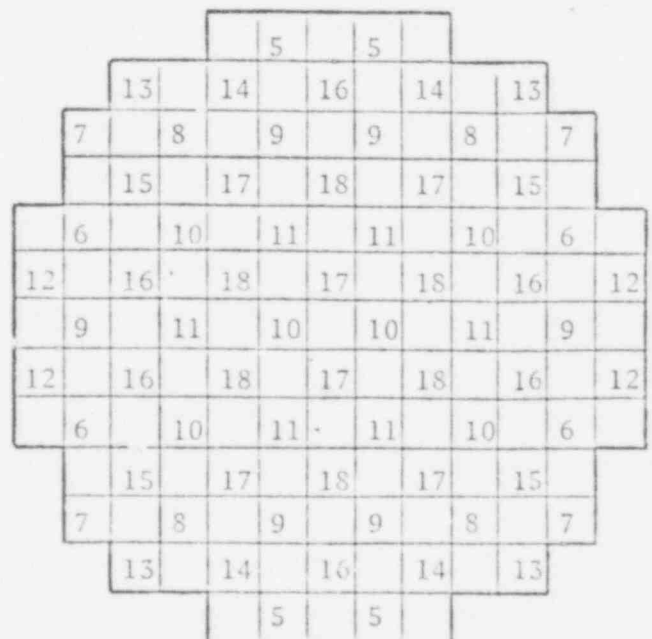
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Fig. 3