James A. FitzPatrick Cycle 11 Start-up Testing Report

I. Cycle 11 operations commenced January 2, 1993. The startup testing program was completed February 3, 1993 after rated power with equilibrium conditions had been achieved. The test program was conducted in accordance with Reactor Analyst Procedure 7.3.30 "Cycle Startup Reactor Physics Test Program". Listed below is a summary of the tests referenced by RAP-7.3.30.

1. Core Loading and Verification:

A full core offload/onload was performed during the refuel outage in which four hundred and eight bundles were relocated, one hundred fifty two bundles were discharged, and one hundred fifty two new bundles were loaded. The Cycle 11 fuel bundle inventory breakdown is as follows:

	Fuel Type	Enrichment	Number	Installed
1.	GE-8	3.19%	76	Reload 7
2.	GE-8	3.39%	32	Reload 8
3.	GE-8	3.36%	152	Reload 8
4.	GE-10	3.22%	56	Reload 9
5.	GE-10	3.24%	88	Reload 9
6.	GE-11	3.02%	4	Reload 9
7.	GE-11	3.59%	80	Reload 10
8.	GE-11	3.56%	72	Reload 10

The final core loading pattern was verified to be correct in accordance with Reactor Analyst Procedures-7.2.4 "Reactor Core Fuel Verification" using an underwater television camera and video recorder. The videotape was independently verified by QA personnel.

2. Control Rod Replacements And Drive Mechanism Tests:

Eight control blades were replaced during the outage with four General Electric Marathon blades and four ABB blades. A total of seventy one of the one hundred and thirty seven original equipment control blades have now been replaced.

Prior to start-up, Surveillance Test ST-20N, "Control Rod Exercise/Timing/Stall Flow Test" was performed on all 137 control rods to demonstrate that each rod is coupled to its drive mechanism, to check that each control rod drive satisfies a travel timing test.

Control rod scram time testing was performed on all 137 control rod drives prior to reaching 40% rated core thermal power.

The results were as follows:

Notch	Tech Spec Limit	Average for 137 rods
46	.338 sec	.298 sec
38	.923 sec	.717 sec
24	1.992 sec	1.464 sec
04	3.554 sec	2.581 sec

The average of the scram insertion times of the three fastest operable control rods for all groups of four control rods in a two by two array were less than the maximum times allowed by the Technical Specifications.

3. <u>Shutdown Margin Test:</u>

Initial Criticality for Cycle 11 was achieved on January 3, 1993. Shutdown margin was demonstrated using the in-sequence critical method which showed the core to have a shutdown margin of 2.58% delta k/k which exceeds the Technical Specification requirement of 0.38% delta k plus R where R = 1.27% delta k for a total of 1.65% delta k.

4. <u>Control Rod Sequence:</u>

The control rod withdrawal sequence was prepared and loaded into the RWM program in accordance with the requirements of Reactor Analyst Procedure 7.3.32 "Reduced Notch Worth Procedure." Prior to reactor startup, surveillance test ST-20A was performed to demonstrate system operability.

5. <u>SRM Performance Check:</u>

SRM Functional Testing was performed prior to startup to demonstrate operability of the SRM monitoring instrumentation. During reactor startup, an SRM/IRM Overlap check was performed to demonstrate that each IRM was on scale before any SRM exceeded the rod block setpoint.

6. <u>Reactivity Anomaly Check:</u>

A comparison between the predicted and actual control rod density was performed at 100% rated core thermal power and 98.2% rated core flow. The actual rod inventory was 571 notches inserted, which was 99 notches less than the predicted notch inventory of 670 notches. A reactivity anomaly of ±1% delta k/k was equivalent to ±419 notches.

7. Core Power Distribution Measurements And LPRM Calibrations:

The core power distribution was monitored throughout the power ascension using 3D-Monicore software in conjunction with the Traversing Incore Probe System (TIP) and the Local Power Range Monitors (LPRMS). LPRM calibrations were performed at 50%, 75%, and 100% of rated power. All fuel power production parameters were maintained within Technical Specification limits.

8. Core Power Symmetry Calculations:

Bundle power symmetry was checked at 50%, 75%, and 100% rated core thermal power. The values calculated are shown below.

Test Plateau	Maximum % Difference	Average % Difference
50% Power	10.18%	3.93%
75% Power	11.62%	4.59%
100% Power	12.42%	3.72%

9. Manual Heat Balance:

A manual heat balance was performed per RAP-7.3.3 at 25%, 50%, 75%, and 100% rated power. In each case, reactor power was calculated to be within 24 MW (1% of rated) of the plant computer calculation. The results are tabulated below.

Test Plateau	Hand Calculation	Computer Calc
25% Power	600.5 MW	608.9 MW
50% Power	1249.3 MW	1230.5 MW
75% Power	1841.6 MW	1828.1 MW
100% Power	2435.4 MW	2428.7 MW

10. LPRM and TIP Response Test:

During scram time testing, when control rod insertions and withdrawals were performed, LPRM response testing was conducted on all operable detectors. This test verified that each operable detector is connected to the appropriate flux amplifier. Four LPRM assemblies were replaced during the outage. In addition, a TIP response test was conducted to verify that each TIP tube is connected to the appropriate LPRM assembly.

11. Core Flow Evaluation:

A Core Flow Evaluation was performed at 100% rated power per Reactor Analyst Procedure 7.3.7. The indicated flow of 75.55 Mlb/hr differed from the calculated flow of 75.75 Mlb/hr by .3%, well within the procedure acceptance criteria of 1.0%.

12. Determination of Rated Drive Flow:

A rated drive flow calculation was performed at 100% power, and the results show that a drive flow of 33.50×10^6 lb/hr produces the rated core flow of 77.0×10^6 lb/hr. The original design value for rated drive flow was 34.2×10^6 lb/hr.

13. TIP System Checkout:

The Traversing Incore Probe System was operated in accordance with Reactor Analyst Procedure-7.3.14. Axial alignment of TIP channels assigned to machines A and B were checked at full power. Machine C channels were not checked due to a failure of the machine following operation at 75% power. The ability of the 3-D Monicore software system to compensate for missing TIP data was used to allow plant operation at 100% power.

14. TIP Reading Uncertainty:

The standard deviation between BASE distributions of symmetrically located TIP strings was determined at 50, 75 and 100 percent power. The resulting TIP reading uncertainties were calculated to be 1.40% at 50% power, 1.47% at 75% power, and 2.22% at 100% power. These values are well below the 8.7% TIP reading uncertainty assumed in the Licensing Topical Report.

15. Core Thermal Hydraulic Stability:

Data was acquired from the APRMs and LPRM detectors at 25 and 75 percent power in accordance with ST-5S, "Neutron Instrumentation Noise Monitoring". This information will serve as baseline data for the operating cycle when Technical Specifications require performance of ST-5S.

16. APRM Calibrations

Numerous APRM calibrations were performed throughout the startup beginning at 10% of rated power.

II. The following Start-up tests, not controlled by RAP-7.3.3C, were performed to satisfy Technical Specification Requirements:

1. Chemical and Radiochemical Tests - Performed per PSP-1, "Reactor Water Sampling and Analysis", and PSP-16, "Guidelines for Start-up, Shutdown, and Scram" which ensures Technical Specification requirements with regard to reactor water chemistry are met.

2. Reactor Vessel Heatup - Performed in accordance with ST-26J, "Heatup and Cooldown Temperature Checks". The reactor vessel heatup was monitored in accordance with the requirements of ST-26J. Reactor vessel heatup rate was maintained less than 60 degrees per hour.

3. IRM Performance - Performed ST-5C, "IRM-APRM Instrument Range Overlap Check" which demonstrated that each APRM channel was on scale before any IRM exceeded the high IRM rod block setpoint.

4. Safety Relief Valves - Performed ST-22B, "Manual Safety Relief Valve Operation and Valve Monitoring System Functional Test". The acceptance criteria of ST-22B were satisfied which demonstrated (1) that each safety relief valve opens and closes fully through operation of control switches on 09-4 control room panel and the remote 02ADS-071 panel, (2) the valve monitoring system operated satisfactorily to indicate valve position, (3) opening of each safety relief valve was verified by observing a ten percent or greater closure of the turbine bypass valves.

5. Main Steam Isolation Valves - Performed ST-1B, "MSIV Fast Closure" which demonstrated that all MSIV's close within the Technical Specification and IST stroke time of 3 to 5 seconds.

6. RCIC System - Performed Reactor Core Isolation Cooling operability testing in accordance with Technical Specification requirements.

7. HPCI System - Performed High Pressure Core Injection operability testing in accordance with Technical Specification requirements.

III. Some of the start-up tests performed during the initial cycle startup were not performed due to the reasons specified below.

(A) Performance of the following tests challenge the reactor protection and safety systems of the plant, and/or places the plant in a degraded condition.

1. Turbine Trip and Generator Load Rejection Test: The purpose of this test was to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator. The turbine stop valves close, and the main generator breaker trips in such a way that a load imbalance trip occurs.

2. Simultaneous Closure of All MSIV'S: The purpose of this test was to (1) functionally check the MSIV's for proper operation, (2) determine the resultant reactor transient behavior, (3) determine valve closure time, and (4) determine the maximum power at which a single valve closure can occur without causing a reactor scram.

3. Loss of Turbine Generator and Offsite Power: The purpose of this test was to determine reactor transient performance during the loss of the main generator and all off-site power.

4. Shutdown From Outside the Control Room: This test demonstrated that using controls located outside the control room the reactor can be scrammed and MSIV's closed, and that operators can control vessel water level and pressure such that a reactor cooldown is initiated.

5. Recirculation Pump Trip Test: The purpose of this test was to evaluate the recirculation flow and reactor power level transients following a single and then dual pump trip.

6. Feedpump Trip Test: The purpose of this test was to check the automatic runback feature of the recirculation pumps.

(B) The following tests measured parameters which needed to be established or verified during the initial plant startup before the plant had any operating history.

1. <u>System Expansion Test:</u> The purpose of this test was to verify that the reactor drywell piping system is free and unrestrained with regard to thermal expansion.

2. <u>Turbine Bypass Valve Measurement Test:</u> The purpose of this test was to demonstrate the ability of the pressure regulator to minimize the reactor pressure disturbance during an abrupt change in steam flow by tripping open and closing a turbine bypass valve.

3. <u>Selected Process Temperatures:</u> The purpose of this test was to establish the minimum recirculation pump speed that ensures adequate mixing in the lower vessel plenum, and to assure that the measured bottom head drain temperature corresponds to the bottom head coolant temperature during normal operation.

4. <u>Vibration Measurements:</u> This test performed vibration measurements on various reactor components to demonstrate the mechanical integrity of the system to flow induced vibrations.

5. <u>Radiation Measurements:</u> This test determined pre-operational background radiation levels in the plant environs to assure protection of plant personnel during plant operation.

6. <u>Recirculation MG Set Speed Control:</u> The purpose of this test was to determine the speed control characteristics of the MG sets, obtain acceptable speed control system performance, and determine maximum allowable pump speed.

7. <u>Flux Response to Control Rods:</u> This test demonstrated the stability of the core local power/reactivity feedback mechanism with regard to small perturbations in reactivity caused by rod movement.

8. <u>RHR Steam Condensing Mode Demonstration:</u> This test demonstrated the RHR system capable of removing decay heat from the reactor by operating in the Steam Condensing Mode.

9. <u>Feedwater System:</u> This test (1) adjusted the feedwater control system for acceptable reactor water level control, (2) demonstrated stable reactor response to subcooling changes, (3) demonstrated capability of the automatic recirculation flow runback feature to prevent low water level scram following the trip of one feed pump, and (4) demonstrated reactor response to loss of feed water heating.

10. <u>Flow Control:</u> This test demonstrated plant response to recirculation flow changes.

11. <u>Reactor Water Cleanup System:</u> This test demonstrated specific aspects of the mechanical operability of the RWCU system.

12. <u>Reactor Water Level:</u> The purpose of this test was to verify the calibration and agreement of the GEMAC and YARWAY water level instrumentation under various conditions. The instrumentation is presently calibrated in accordance with Technical Specifications.

13. <u>Pressure Regulator Test:</u> The purpose of this test was to determine the optimum setting for the pressure control loop by analysis of transients induced in the reactor pressure control system by means of the pressure regulators.

14. <u>Warranty Demonstration Test:</u> This test was performed to verify warranted condition of the Nuclear Steam Supply System.

15. <u>Reactor Moisture Separator And Dryer Efficiency Test:</u> This test calculated the efficiency of the moisture separator and dryer.