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Subject: JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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CYCLE 10 START-UP TESTING REPORT

Gentlemen:

Enclosed you will find the Cycle 10 Start-up Testing Report for the James A. FitzPatrick Nuclear Power Plant, which is submitted to you in accordance with the reporting requirements of section 6.9.A.1 of the Plant Technical Specifications.

We trust you will find this information satisfactory. However, should you desire more information, please contact Mr. David Burch at (315) 349-6311.

Very Truly Yours,

WILLIAM FERNANDEZ

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James A. FitzPatrick Cycle 10 Start-up Testing Report

I. Cycle 10 on-line operations commenced June 24, 1990, ending a 85 day refueling outage. The startup testing program commenced June 19, 1990 and was completed August 7, 1990. 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 performed in accordance with RAP-7.3.30.

1. Core Loading and Verification:

A fuel shuffle was performed during this outage in which four hundred and twelve bundles were relocated, one hundred forty-eight bundles were discharged, and one hundred forty-eight new bundles were loaded. The cycle 10 fuel bundle breakdown is as follows:

1. GE7 fuel, 2.99% enrichment, 40 bundles, Reload 6
2. Westinghouse LTA's, 2.99% enrichment, 4 bundles, Reload 7
3. GE-8 fuel, 3.19% enrichment, 184 bundles, Reload 7
4. GE-8 fuel, 3.39% enrichment, 32 bundles, Reload 8
5. GE-8 fuel, 3.36% enrichment, 152 bundles, Reload 8
6. GE-10 fuel, 3.22% enrichment, 56 bundles, Reload 9
7. GE-10 fuel, 3.24% enrichment, 88 bundles, Reload 9
8. GE-11 fuel, 3.02% enrichment, 4 bundles, Reload 9

The final core loading was verified in accordance with RAP-7.2.4 "Reactor Fuel Verification" using an underwater television camera and video recorder. The videotape was independently checked by QA personnel and documented by QA Surveillance Report# 1406.

2. Control Rod Drive Tests:

Eighteen control blades were replaced during the outage with General Electric Duralife 230 control blades. A total of sixty three of the one hundred and thirty seven original equipment control blades have now been replaced. Prior to start-up, Surveillance test ST-20K, "Control Rod Exercise/Venting" was performed on all 137 control rods to demonstrate that each rod is coupled to its drive mechanism, and 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	.295 sec
38	.923 sec	.709 sec
24	1.992 sec	1.452 sec
04	3.554 sec	2.566 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. Shutdown Margin Test:

Initial Criticality for cycle 10 was achieved on June 19,1990. Shutdown margin was demonstrated using the in-sequence critical method which showed the core to have a shutdown margin of 2.703% delta k/k which exceeds the Technical Specification requirement of 0.38% delta k plus R where R = .41% delta k for a total of .79% delta k.

4. Control Rod Sequence:

Sequence 10A2 was loaded into the RWM program in accordance with the requirements of the Reduced Notch Worth Procedure. Prior to startup, a surveillance test of the RWM was performed to demonstrate system operability.

5. SRM Performance Check:

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. Reactivity Anomaly Check:

A comparison between the predicted and actual control rod density was performed at 100% rated core thermal power and 99.1% rated core flow. The actual rod inventory was 288 notches inserted which is 65 notches less than the predicted notch inventory of 353 notches. A reactivity "anomaly" of ±1% delta k/k is equivalent to ±300 notches.

7. <u>Power Distribution Measurements:</u>

Core power distribution was monitored throughout the power ascension using the Traversing Incore Probe System (TIP) and the Local Power Range Monitors (LPRMS). LPRM calibrations were performed at 25%, 50%, 75%, and 100% of rated core thermal power. Fuel thermal limits were maintained within Technical Specification limits.

8. <u>Core Power Symmetry Calculations:</u>

Core power symmetry was checked at 50%, 75%, and 100% rated core thermal power. In all cases, the maximum percent difference in power level of symmetrically located fuel bundles was found to be less than 10%. The actual values calculated are shown below.

Test Plateau	Maximum % Difference	Average % Difference
50%	8.96%	1.84%
75%	5.73%	2.60%
100%	8.49%	2.45%

9. <u>Manual Heat Balance:</u>

A manual heat balance was performed per RAP 7.3.3 at 25%, 50%, 75%, and 100% rated core thermal power and in each case thermal power was found to be within 50 MW (2% of rated) of the plant computer thermal power calculation. The values are tabulated below.

Test Plateau	Hand Calculation	Computer Calc
25%	573.7 MW	623.4 MW
50%	1213.1 MW	1228.2 MW
75%	1723.4 MW	1714.9 MW
100%	2440.7 MW	2432.1 MW

10. <u>LPRM and TIP Response Test:</u>

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. Six 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. Plant Computer Checkout:

The computer databank for Cycle 10 operations was installed and verified to be correct per Reactor Analyst Procedure 7.3.17 "Core Monitoring Software and Database Changes". The plant computer calculations were compared with the offline core performance programs at 25%, 50%, 75%, and 100% rated thermal power. The results showed close agreement in location and magnitude of all thermal limits.

12. Core Flow Evaluation:

Core Flow Indication Calibrations were performed at 75% and 100% rated power per Reactor Analyst Procedure 7.3.7. In both evaluations the indicated flow matched the calculated flow within less than .5%, which is well within the ±2.5% tolerance assumed for the statistical uncertainty in the Licensing Topical Report.

13. Determination of Rated Drive Flow:

A rated drive flow calculation was performed at 100% power, and the results show that a drive flow of 32.56×10^6 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.

14. TIP System Checkout:

Prior to startup, the core top and bottom limits for each LPRM string were set by hand cranking the TIP probes to the top of each LPRM instrument tube. These limits were checked per Reactor Analyst Procedure 7.3.14 "TIP SYSTEM" at full power by checking the location of spacer dips on the flux traces.

15. TIP Reading Uncertainty:

The standard deviation between symmetrically located TIP strings was determined from BASE distributions obtained at 25, 50, 75 and 100 percent power. The resulting TIP reading uncertainties were calculated to be 3.2% at 25% power, 2.0% at 50% power, 3.1% at 75% power, and 2.6% at 100% power. These values are well within the 7.1% TIP reading uncertainty assumed in the Licensing Topical Report.

16. 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.

II. Other Start-up tests performed to satisfy Technical Specification Requirements included the following:

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 performed in accordance with the requirements of ST-26J which requires reactor coolant system pressure and temperature be at or to the right of curve C shown in Figure 3.6-1, and the maximum temperature change during any one hour equal to or less than 100 degrees fahrenheit.

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 ST-24A, "RCIC Pump and Valve Operability Test" which verified RCIC pump, motor, and valve operability. A Simulated Automatic Actuation test was performed in accordance with F-ST-24E which demonstrated the ability of the RCIC system to deliver a flow rate of 400 gpm.

7. HPCI System - Performed ST-4B, "HPCI Pump and MOV Operability Tests" to verify operability of the HPCI turbine and pump assembly, and associated motor operated valves. A simulated Automatic Actuation Test was performed in accordance with F-ST-4A which demonstrated the ability of the HPCI system to deliver a flow rate of 4250 gpm. In addition, Preoperational Test POT-23E "HPCI INJECTION TO REACTOR VESSEL" was conducted which verified the HPCI system met all applicable system design requirements. These included (1) demonstrating the HPCI pump discharge flow rate equals 4250 gpm in less than 30 seconds from automatic initiation at rated reactor pressure, (2) the HPCI turbine does not trip or isolate during the test, and (3) the decay ratio of any HPCI system related variable is not greater than 0.25.

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 test challenges 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 is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator. The turbine stop valves are closed, and the main generator breaker tripped in such a way that a load imbalance trip occurs.

2. Simultaneous Closure of All MSIV'S: The purpose of this test is to (1) functionally check the MSIV's for proper operation, (2) to 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 is 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 is to evaluate the recirculation flow and reactor power level transients following a single and then dual pump trip.

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

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

2. Turbine Bypass Valve Measurement Test: The purpose of this test is 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. Selected Process Temperatures: 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. Vibration Measurements: This test performed vibration measurements on various reactor components to demonstrate the mechanical integrity of the system to flow induced vibrations.

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

6. Recirculation MG Set Speed Control: 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. Flux Response to Control Rods: The purpose of this test is to demonstrate the stability of the core local power/reactivity feedback mechanism to small perturbations caused by rod movement.

8. RHR Steam Condensing Mode Demonstration: This test demonstrates the RHR system is capable of removing decay heat from the reactor by operating in the Steam Condensing Mode.

9. Feedwater System: 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. Flow Control: The test demonstrates plant response to recirculation flow changes.

11. Reactor Water Cleanup System: This test demonstrates specific aspects of the mechanical operability of the RWCU system.

12. Reactor Water Level: 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. Pressure Regulator Test: the main 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.