# THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

## PERRY NUCLEAR POWER PLANT UNIT 1

## POST-REFUELING STARTUP TEST SUMMARY REPORT

## CYCLE 4

## 1.1 INTRODUCTION

This report presents a su ary of the results from the post-refueling startup tests which were conducted in preparation for Cycle 4 at Unit 1 of the Perry Nuclear Power Plant. This report is submitted pursuant to Technical Specification 6.9.1.1.

## 1.2 PLANT DESCRIPTION

The Perry Nuclear Power Plant is operated by the Cleveland Electric Illuminating Company (CEI) and is located near Lake Erie in Lake County, Ohio. Unit 1 has a Boiling Water Reactor (BWR) nuclear steam supply system as designed and supplied by the General Electric Company (GE) and designated BWR/6, with a Mark III containment. The balance of plant was designed by Gilbert Associates, Inc., Reading, Pennsylvania, as architect-engineer.

The rated core thermal power is 3579 MWt with a gross electrical output of 1250 MWe. The turbine is an 1800 rpm tandem compound, six flow, reheat unit consisting of one double flow high pressure stage in tandem with three low pressure stages. The generator is a direct coupled 60 Hz, 22 KV, three-phase unit with a water cooled stator and hydrogen cooled rotor.

## 1.3 SUMMARY OF ACTIVITIES DURING REFUELING OUTAGE 3

The third refueling outage at Unit 1 of the Perry Nuclear Power Plant began on March 21, 1992 (main generator off-line) and was completed on June 13, 1992 (main generator on-line). The outage duration was 84 days.

Key activities of this refueling outage were: the performance of a core shuffle, replacement of 204 fuel bundles (out of 748), replacement of 12 control rod blades (out of 177), the installation of over 50 design changes, the completion of a large number of corrective and preventive maintenance activities, and the performance of required Technical Specification surveillance tests.

| | |
|---|---|
| Start Date | 21 March, 1992 |
| Stop Date | 13 June, 1992 |
| Duration | 84 days |
| Work Orders | 1662 |
| Design Changes | 58 |
| Surveillances | 481 |
| Repetitive Tasks | 1109 |

## 2.1 DESCRIPTION OF DIFFERENCE IN FUEL DESIGNS

Unit 1 at the Perry Nuclear Power Plant used the following General Electric fuel designs for cycle 3:

| | |
|---|---|
| GE6B-P8SIB176-4GZ-120M-150-T | 140 bundles, |
| GE6B-P8SIB219-4GZ-120M-150-T | 64 bundles, |
| GE8B-P8SQB301-5GZ-120M-150-T | 136 bundles, |
| GE8B-P8SQB301-7GZ-120M-150-T | 136 bundles, |
| GE8B-P8SQB320-9GZ-120M-150-T | 104 bundles, |
| GE8B-P8SQB322-7GZ-120M-150-T | 168 bundles. |

and the following General Electric fuel designs for cycle 4:

| | |
|---|---|
| GE6B-P8SIB176-4GZ-120M-150-T | 5 bundles, |
| GE8B-P8SQB301-5GZ-120M-150-T | 136 bundles, |
| GE8B-P8SQB301-7GZ-120M-150-T | 135 bundles, |
| GE8B-P8SQB320-9GZ-120M-150-T | 104 bundles, |
| GE8B-P8SQB322-7GZ-120M-150-T | 164 bundles, |
| GE10-P8SXB306-10GZ2-120M-150-T | 68 bundles, |
| GE10-P8SXB306-11GZ3-120M-150-T | 136 bundles. |

Complete descriptions of these General Electric fuel designs are given in GESTAR II, General Electric Standard Application for Reload Fuel. The major differences in the GE10 fuel relative to the GE8 fuel include:

1) An increase in the number of gadolinium rods in the bundle. (10 or 11 vice maximum of 9 previously)
2) Use of a large central water rod instead of two small water rods.
3) Fewer fuel rods per bundle. (60 for GE10 vice 62 for GE8)
4) Use of an interactive channel with machined flow trippers.
5) Axial enrichment zoning of Uranium.
6) Ferrule spacer design and resulting improved bundle flow.
7) Expanded and offset lattice design.

## 3.1 POST-REFUELING STARTUP TEST PROGRAM

During refueling operations and the subsequent return to power, activities were controlled under normal administrative programs rather than a separate formally defined post-refueling startup test program. These administrative programs cover areas of normal operation such as:

| | |
|---|---|
| Design Changes/Post-modification Testing | Post-maintanence Testing |
| Technical Specification Surveillances | Inservice Inspection |
| Special Nuclear Material Control | Periodic and Special Tests |
| Computer Software Modification | Radiation Control |

The acceptance criteria for these tests were derived from the requirements of the administrative programs. The reactor conditions for conducting the tests were guided by requirements in the appropriate administrative program.

## 3.2 POST-REFUELING STARTUP TEST REPORTS

As required by Technical Specification 6.9.1.2, this report addresses each of the startup tests identified in USAR Subsection 14.2.12.2. Each test was evaluated by the Responsible System Engineer who determined whether the test was impacted by any refueling activity. Those tests determined not to be impacted by any refueling activity are listed in Table 3.2-1.

For those tests which were impacted by refueling activities, this report lists:

1. A description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications;

2. A description of any corrective actions that were required to obtain satisfactory operation;

3. Any additional specific details required in license conditions based on other commitments.

3.3 POST-REFUELING STARTUP TEST REPORTS -- SUMMARIES

14.2.12.2.3 Test Number 3 - Fuel Loading

Test Objective

The purpose of this test is to load fuel safely and efficiently.

Discussion

Fuel unloading and loading was conducted in Operational Condition 5 under IOI-9, "Refueling." Fuel movement followed a predetermined plan in accordance with a Fuel Movement Checklist per FTI-D09, "Use of the Fuel Movement Checklist." A core shuffle (vice an offload/reload) was performed to minimize fuel movement.

Fuel movement was performed in several phases. First, fuel was offloaded from control cells whose control rod drives or blades were to be replaced. After these bundles were offloaded, the control rod drive and blade work commenced. Control rod drives were replace to enhance performance during the upcoming cycle. Control rod blades were replaced to ensure that the depletion limits would not be exceeded during the cycle. During the time at which any control rod was withdrawn and/or removed, no fuel was loaded into the reactor. By the completion of the control rod work, all control rods were fully inserted. The core shuffle was then started. Discharge fuel was removed from the core, new fuel inserted, and reload fuel moved within the reactor vessel. All fuel moves involving the reactor were analyzed for shutdown margin, within a minimum analytical margin of 1% delta-k/k. This ensured the acceptance criteria of at least 0.38% delta-k/k was met for all core configurations.

14.2.12.2.4 Test Number 4 - Full Core Shutdown Margin

Test Objective

The purpose of this test is first to demonstrate the reactor is subcritical throughout the fuel cycle with any single control rod fully withdrawn and second to determine quantitatively the shutdown margin of the as-loaded core.

DISCUSSION

Full core shutdown margin and reactivity anomaly were demonstrated to be within their Technical Specification requirements during Cycle 4 startup (reactor startup Number 55). SVI-B13-T0001, "Insequence Critical Shutdown Margin Calculation," was performed at 0% power and 143 degrees F moderator temperature. The minimum shutdown margin for Cycle 4 was measured to be 1.18% delta-k/k, compared to the Technical Specification minimum of 0.38% delta-k/k. The measured reactivity anomaly at the beginning of cycle was 0.245% delta-k/k, compared to the Technical Specification maximum of 1.0% delta-k/k.

14.2.12.2.5 Test Number 5 - Control Rod Drive System

Test Objective

The purposes of the control rod drive system tests are to demonstrate that the Control Rod Drive (CRD) system operates properly over the full range of reactor coolant temperatures and pressures from ambient to operating, and to determine the initial operating characteristics of the entire CRD system.

Discussion

Control rod insert/withdraw timing was performed in accordance with PTI-C11-P0005, "CRD Speed and Drift Testing," on all control rods. This evolution was performed following completion of core alterations and before the initial criticality of Cycle 4. All rods were verified to have insert and withdraw times of 40 to 60 seconds. All tests were performed at atmospheric pressure and various temperatures.

Twenty-two control rods were friction tested per PTI-C11-P0003, "Control Rod Friction Testing." Ten of these control rods had their associated control rod drive mechanisms replaced during the outage and the other 12 were control rod blade replacements. The absence of control rod blade interference for the remaining 155 control rods was confirmed by an evaluation of the results of the control rod insert/withdraw timing and the scram timing tests.

Control rod scram timing was performed for all control rods in accordance with SVI-C11-T1006, "Control Rod Maximum Scram Insertion Time." Control rod maximum scram insertion times were determined in accordance with Technical Specification 4.1.3.2.b for those rods whose associated control rod drive mechanisms were replaced, those control rods with blades replaced, and those rods whose hydraulic control units had maintenance during the outage. This testing was performed at rated temperature and pressure prior to entering Operational Condition 1. Scram timing was performed on the remaining rods in the core per Technical Specification 4.1.3.2.a prior to exceeding 40% of rated thermal power. Buffer times (the slowing down time at position 00 during a scram) were not measured because Perry has not experienced a control rod drive with an unacceptable buffer time in previous tests.

All rods were 'fast' per Technical Specification 3.1.3.2 on their first attempt, with the exception of control rod 54-27 which was determined to be 'slow' operable in accordance with Technical Specification 3.1.3.2.a. A work order was generated to replace the scram solenoid pilot valve for rod 54-27. The control rod was scram timed following replacement of the scram solenoid pilot valve and determined to be 'fast' prior to exceeding 40% rated thermal power.

Ganged rod timing was not performed because all control rods satisfied the individual insert/withdraw timing requirements.

No other hydraulic testing was performed on the control rod drive system as there were no changes to the system that would have affected these parameters.

The timing tests were considered physics tests in USAR 14.2.12.2.5.

14.2.12.2.6 Test Number 6 - SRM Performance and Control Rod Sequence

Test Objective

The purpose of this test is to demonstrate that the neutron sources, Source Range Monitor (SRM) instrumentation and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner.

Discussion

SRM B was inoperable for reactor startup. This SRM was identified as a potential Technical Specification impact on operation and was identified for corrective maintenance. The other three SRMs indicated greater than 0.7 cps with a signal to noise ratio greater than two to one for approach to criticality in accordance with IOI-1, "Cold Startup."

This test was considered a physics test in USAR 14.2.12.2.6.

14.2.12.2.7 Test Number 8 - Rod Sequence Exchange

Test Objective

The purpose of this test is to perform a representative sequence exchange of control rod patterns at a significant power level.

Discussion

Because cycle 4 utilizes a Control Cell Core fuel loading pattern, the reactor operates in the same sequence for the entire cycle. Therefore, no control rod sequence exchange was performed.

14.2.12.2.8 Test Number 10 - Intermediate Range Monitor Perfor .nce

Test Objective

The purpose of this test is to adjust the Intermediate Range Monitor (IRM) system to obtain an optimum overlap with the Source Range Monitor (SRM) and Average Power Range Monitor (APRM) systems.

Discussion

All eight IRMs were confirmed to have a half decade overlap with both the SRMs and the APRMs during performance of IOI-1, "Cold Startup."

This test was considered a physics test in USAR 14.2.12.2.8.

14.2.12.2.9 Test Number 11 - LPRM Calibration

Test Objective

The purpose of this test is to calibrate the Local Power Range Monitoring (LPRM) system and to verify the LPRM flux response.

Discussion

LPRM flux response was verified with the reactor power between 8% and 35% of rated power in accordance with PTI-C51-P0001, "Verification of Proper LPRM Connection." The testing procedure involves moving an adjacent control rod past each LPRM and observing the appropriate change (significant in magnitude and correct in direction) in the LPRM reading. Two of the LPRMs did not respond; work requests were initiated. The remaining 162 LPRMs responded satisfactorily.

Near 38% reactor power, Process Computer Program OD1, "Whole-Core LPRM Calibration and Base Distribution," was performed. Based on this program, the LPRM readings are internally adjusted by the Process Computer to be used in the core power distribution and thermal limits calculation. At this time, the gain adjustment factors (GAFs) were not manually adjusted. Although the GAF array ranged between 0.811 and 1.287 (outside the desired 10% band), the LPRMs outside the desired range were not considered inoperable because the Process Computer and Average Power Range Monitor (APRM) calibration correct for any variation. In addition, the GAFs were not adjusted because of the non-steady state power conditions and non-equilibrium xenon conditions.

After full power was reached, SVI-C51-T5351, "LPRM Calibration," was performed to manually adjust the LPRM GAFs. The 12 new LPRMs installed during RF03 were set with an initial calibration current of 700 microamps. Eight of the 12 LPRMs were outside the SVI's acceptance criteria and required calibration per the SVI.

This test was considered a physics test in USAR 14.2.12.2.9.

14.2.12.2.10 Test Number 12 - APRM Calibration

Test Objective

The purpose of this test is to calibrate the Average Power Range Monitor (APRM) system.

Discussion

During startup following refueling, the APRMs were calibrated several times in accordance with SVI-C51-T0024, "APRM Gain and Channel Calibration" as the reactor was brought to full power. The APRMs were calibrated to read within 2% of actual core thermal power on the following dates at the identified power levels.

| Date | % Rated CTP |
|---|---|
| 14-June-92 | 27 |
| 16-June-92 | 38 |
| 17-June-92 | 38, 59 |
| 21-June-92 | 58 |
| 22-June-92 | 93, 94, 100 |

Technical Specification and fuel warranty limits on APRM scram and rod block were not exceeded. The startup APRM scram functions were checked in the SVI-C51-T0030 series, "APRM Channel Calibration."

This test was considered a physics test in 14.2.12.2.10.

14.2.12.2.11 Test Number 13 - NSS Process Computer

Test Objective

The purpose of this test is to verify the performance of the NSS Process Computer and on-line NSS computer programs under plant operating conditions.

Discussion

The Traversing In-Core Probe (TIP) tubing undervessel was removed/installed during the refuel outage per Work Order 91-3460. The TIP alignment was checked, after installation of the tubing, per ICI-C-C51-5, "TIP System Mechanical Drive System Calibration."

The NSS software for Cycle 4 was updated and tested via Computer Program Modification Request 92-09 in accordance with PAP-0506, "Computer Access and Software Control." The Periodic NSS Core Performance Log (P1) calculations for thermal limits were compared to those from an independent copy of the NSS software to establish the NSS software's acceptability for Technical Specification application. The GE program Backup Core Thermal Limits Evaluation (BUCLE) is no longer used as the backup thermal limit calculation method for Perry, therefore, no BUCLE results were compared during the startup tests for Cycle 4. The LPRM calibration factors calculated by OD1 were verified by manual calculations.

In preparation for the above comparisons, the input data and intermediate calculations were evaluated. At the end of Cycle 3, records were generated for the NSS database for fuel constants, exposure and isotopics, LPRM exposures, and control rod exposures. While the plant was shutdown for reactor refueling, the NSS database was updated to reflect the new core design and a statistically significant number of entries in selected arrays were checked for reasonableness. During the reactor startup from 0% to 25% of rated thermal power, the process computer values for control rod positions, LPRM readings, and thermal hydraulic parameters were compared against readings from appropriate plant sensors. While slightly less than 25% of rated thermal power, the process computer calculation of reactor thermal power was compared against a manual calculation per FTI-B05, "Core Heat Balance," and verified to agree to within 1% of rated thermal power. Actual agreement was within 0.5% of rated thermal power. After validation of the core thermal power calculation, the Computer Outage Recovery Monitor program (OD15) was initiated to start the Ten-Minute Core Energy Increment program (P4) which was checked via manual calculations. Reactor power was increased and the Whole-Core LPRM Calibration and Base Distribution program (OD1) was performed to calibrate the LPRMs for the process computer. The process computer calculations for TIP normalization factor, LPRM calibration constants, and LPRM substitute values were verified against appropriate manual calculations. Prior to actually physically adjusting the amplifier gains on the LPRMs, the power, time, flow, and run flags of the P1 program were checked for reasonableness; and, the thermal limits from P1 were compared against those from the independent source code listing. The results of the independent source code are identified as P1-PCjr in the following tables. The results of that comparison are as follows:

| SOURCE | | P1 | | P1-PCjr |
|---|---|---|---|---|
| **TEST CONDITIONS:** | | | | |
| Date | | 14-JUN-92 | | 14-JUN-92 |
| Core Thermal Power (MWt) | | 1319.3 | | 1315.8 |
| Core Flow (Mlb/hr) | | 35.39 | | 35.43 |
| Core Inlet Subcooling (Btu/lb) | | 32.04 | | 31.93 |
| Reactor Dome Pressure (psia) | | 982.6 | | 982.6 |
| **NSS SOFTWARE RESULTS:** | | | | |
| MCPR | Location: 33-20 | 2.370 | | 2.376 |
| | Difference (%) | | 0.252 | |
| LHGR | Location: 25-30-5 (kW/ft) | 5.74 | | 5.74 |
| | Difference (%) | | 0.000 | |
| MAPLHGR | Location: 19-34-8 (kW/ft) | 4.92 | | 4.89 |
| | Difference (%) | | 0.613 | |

The Thermal Data in Specified Fuel Bundle program (OD6) calculation of thermal limits was compared against the corresponding P1 calculations with no differences being identified. At this point the OD1, P1 and OD6 NSS programs were considered operational. A second confirmatory check of these programs was performed at reactor conditions closer to rated core power and rated core flow. In this second check the power, time, flow, and run flags of the P1 program were again checked for reasonableness; and, the thermal limits from P1 were again compared against those from the independent source code. The results of this comparison are as follows:

| SOURCE | | P1 | | P1-PCjr |
|---|---|---|---|---|
| **TEST CONDITIONS:** | | | | |
| Date | | 24-JUN-92 | | 24-JUN-92 |
| Core Thermal Power (MWt) | | 3577.5 | | 3574.4 |
| Core Flow (Mlb/hr) | | 90.51 | | 90.39 |
| Core Inlet Subcooling (Btu/lb) | | 23.85 | | 23.88 |
| Reactor Dome Pressure (psia) | | 1039.6 | | 1039.6 |
| **NSS SOFTWARE RESULTS:** | | | | |
| MCPR | Location: 33-42 | 1.366 | | 1.366 |
| | Difference (%) | | 0.000 | |
| LHGR | Location: 35-48-12 (kW/ft) | 12.61 | | 12.60 |
| | Difference (%) | | 0.079 | |
| MAPLHGR | Location: 33-48-12 (kW/ft) | 10.64 | | 10.63 |
| | Difference(%) | | 0.094 | |

The generator and core energy accumulation features of the P1 program were checked. After P1 was considered operational, the daily generator and core energy accumulation and the LPRM exposure accumulation features of the P2 program were checked. After the Daily Core Performance Summary program (P2) was considered operational, the total generator and core energy accumulation and the control rod exposure accumulation features of the Monthly Core Performance Summary program (P3) were checked. Finally, the LPRM drift detection and reset feature of the Drifting LPRM Diagnostic program (P5) was checked. At this point, all programs were considered operational.

14.2.12.2.15 Test Number 18 - Core Power Distribution

Test Objective

The purpose of this test is to determine the reproducibility of the Traversing In-Core Probe (TIP) system readings.

Discussion

The licensing basis TIP uncertainty for the MCPR Safety Limit is 8.7% (standard deviation), as identified in Technical Specification 3.3.7.7.b. This value is based on a model uncertainty of 4.6%, LPRM uncertainty of 3.4%, and a TIP measurement uncertainty of 6.6%. The TIP measurement uncertainty is based on 6.4% geometry uncertainty and 1.2% random noise uncertainty.

FTI-A16, "Total TIP Uncertainty," was performed in an octant symmetric control rod pattern at approximately 100% of rated thermal power to determine the TIP measurement uncertainty. The TIP measurement uncertainty was determined to be 1.513%. A comparison to the 6.6% value for TIP measurement uncertainty assumed in the licensing basis showed the TIP reading reproducibility to be acceptable.

This test was considered a physics test in USAR 14.2.12.2.15.

## 14.2.12.2.16 Test Number 19 - Core Performance

### Test Objective

The purposes of this test are to evaluate the core thermal power and core flow and to evaluate the following core performance parameters:

1. Maximum Linear Heat Generation Rate (MLHGR).
2. Minimum Critical Power Ratio (MCPR).
3. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR).

### Discussion

The fuel thermal limits were monitored on a shiftly basis under the Technical Specification Rounds Instruction. The core thermal power was continuously monitored and controlled under IOI-3, "Power Changes." During post refueling testing, there were no fuel thermal limit violations and no steady-state operation above the more limiting of rated thermal power or the bounding licensed load line. The following lists the average thermal power (CMWTA) and the most limiting values for each of the thermal limits for every day in June that reactor power was above 25% of rated thermal power.

| Date | CMWTA (MWt) | MFLCPR | MFLPD | MAPRAT |
|---|---|---|---|---|
| 14-JUN-92 | 1133 | 0.891 | 0.436 | 0.637 |
| 15-JUN-92 | 1130 | 0.792 | 0.394 | 0.571 |
| 16-JUN-92 | 1262 | 0.816 | 0.521 | 0.684 |
| 17-JUN-92 | 1782 | 0.810 | 0.650 | 0.791 |
| 18-JUN-92 | 1533 | 0.781 | 0.565 | 0.704 |
| 19-JUN-92 | 1378 | 0.813 | 0.391 | 0.555 |
| 20-JUN-92 | 1785 | 0.811 | 0.585 | 0.713 |
| 21-JUN-92 | 2064 | 0.761 | 0.570 | 0.705 |
| 22-JUN-92 | 3346 | 0.875 | 0.914 | 0.929 |
| 23-JUN-92 | 3572 | 0.896 | 0.951 | 0.961 |
| 24-JUN-92 | 3576 | 0.888 | 0.916 | 0.925 |
| 25-JUN-92 | 3576 | 0.922 | 0.874 | 0.883 |
| 26-JUN-92 | 3576 | 0.914 | 0.867 | 0.875 |
| 27-JUN-92 | 3576 | 0.907 | 0.869 | 0.877 |
| 28-JUN-92 | 3562 | 0.919 | 0.873 | 0.881 |
| 29-JUN-92 | 3576 | 0.906 | 0.871 | 0.879 |
| 30-JUN-92 | 3577 | 0.924 | 0.870 | 0.879 |

This test was considered a physics test in USAR 14.2.12.2.15.

14.2.12.2.18 Test Number 21 - Core Power - Void Mode Response

Test Objective

The purpose of this test is to measure the stability of the core power-void dynamic response and to demonstrate that its behavior is within specified limits.

Discussion

This test was not performed during the return to power since the low pressure drop spacers used in the new GE10 fuel increases reactor stability. This stability is bounded by the older fuel designs in the reactor.

This test was considered a physics test in USAR 14.2.12.2.18.

14.2.12.31 Test Number 35 - Recirculation System Flow Calibration

Test Objective

The purpose of this test is to perform complete calibration of the installed recirculation system flow instrumentation.

Discussion

The jet pump flow instrumentation was checked at steady state conditions with core flow indicative of an operating core flow for the cycle. As part of the the NSS software verification via Computer Program/Modification Request 92-09 under PAP-0506, "Computer Access and Software Control," the measured core flow was verified against an established core flow/drive flow correlation. The comparison indicated good agreement and no adjustment of the instrumentation was performed.

At approximately 83% of rated core flow, FTI-A13, "Core Flow Calibration," was performed and the instrumentation was calibrated because the measured core flow did not show good agreement with the previously established correlation as required by FTI-A13, "Core Flow Calibration." The gain adjustment factors for the summers were determined from the ERIS computer program. Following the adjustment of the gain on the B jet pump loop summer, the gain adjustment factors were recalculated using ERIS and were determined to be within 1%.

| Date | | | 09 Jul 92 | 15 Jul 92 |
|---|---|---|---|---|
| Jet Pump Loop Flow A: | | | | |
| Recorder | B33-R612A | | 44 | 44 |
| Summer | B33-K611A | Measured | 44.00 | 43.58 |
| | | Calculated | 43.49 | 43.62 |
| Gain Adjustment Factor | | | 0.993 | 0.997 |
| Jet Pump Loop Flow B: | | | | |
| Recorder | B33-R612B | | 44 | 42 |
| Summer | B33-K611B | Measured | 43.59 | 41.73 |
| | | Calculated | 42.59 | 41.80 |
| Gain Adjustment Factor | | | 0.984 | 1.001 |
| Total Core Flow: | | | | |
| Recorder | B33-R613 | | 87 | 85.5 |
| Summer | B33-K613 | Measured | 87.16 | 84.88 |
| | | Calculated | 86.08 | 85.42 |
| Gain Adjustment Factor | | | 0.993 | 1.003 |

Except for the Gain Adjustment Factor, the values above ar[illegible] Mlb/hr.

The electronics for the APRM flow-bias instrumentation [illegible] calibrated in SVI-C51-T0030, "APRM Channel Calibration for 1C51-K605," [illegible]ies and the flow transmitters were calibrated in the SVI-C51-T0029, "APRM Flow Reference Transmitter 1B33-N014 and 1B33-N024 Calibration," series. The flow biased APRM system was verified to be within the established core flow/drive flow correlation in SVI-C51-T0026, "APRM Flow Biased Power-Flow Verification."

Table 3.2-1

The following tests from USAR 14.2.12.2 were evaluated by the Responsible System Engineer and determined not to have been impacted by any refueling activities.

| | | |
|---|---|---|
| 14.2.12.2.1 | Test Number 1 | - Chemical and Radiochemical |
| 14.2.12.2.2 | Test Number 2 | - Radiation Monitoring |
| 14.2.12.2.12 | Test Number 14 | - RCIC System |
| 14.2.12.2.13 | Test Number 16A | - Selected Process Temperatures |
| 14.2.12.2.13.1 | Test Number 16B | - Water Level Reference Leg Temperature |
| 14.2.12.2.14 | Test Number 17 | - System Expansion |
| 14.2.12.2.17 | Test Number 20 | - Steam Production Startup Test |
| 14.2.12.2.19 | Test Number 22 | - Pressure Regulator |
| 14.2.12.2.20.1 | Test Number 23A | - Feedwater System |
| 14.2.12.2.20.2 | Test Number 23B | - Loss of Feedwater Heating |
| 14.2.12.2.20.3 | Test Number 23C | - Feedwater Pump Trip |
| 14.2.12.2.20.4 | Test Number 23D | - Maximum Feedwater Runout Capability |
| 14.2.12.2.21 | Test Number 24 | - Turbine Valve Surveillance |
| 14.2.12.2.22.1 | Test Number 25A | - Main Steam Isolation Valves Function Tests |
| 14.2.12.2.22.2 | Test Number 25B | - Full Reactor Isolation |
| 14.2.12.2.22.3 | Test Number 25C | - Main Steamline Flow Venturi Calibration |
| 14.2.12.2.23 | Test Number 26 | - Relief Valves |
| 14.2.12.2.24 | Test Number 27 | - Turbine Trip and Generator Load Rejection |
| 14.2.12.2.25 | Test Number 28 | - Shutdown From Outside the Control Room |
| 14.2.12.2.26.1 | Test Number 29A | - Recirculation Flow Control - Valve Position Control |
| 14.2.12.2.26.2 | Test Number 29B | - Recirculation Flow Control - Flow Loop |
| 14.2.12.2.27.1 | Test Number 30A | - One Pump Trip |
| 14.2.12.2.27.2 | Test Number 30B | - RPT Trip of Two Pumps |
| 14.2.12.2.27.3 | Test Number 30C | - System Performance |
| 14.2.12.2.27.4 | Test Number 30D | - Test Deleted |
| 14.2.12.2.27.5 | Test Number 30E | - Recirculation System Cavitation |
| 14.2.12.2.28 | Test Number 31 | - Loss of Turbine-Generator & Offsite Power |
| 14.2.12.2.29 | Test Number 33 | - Drywell Piping Vibration |
| 14.2.12.2.30 | Test Number 34 | - Vibration Measurement |
| 14.2.12.2.32 | Test Number 36 | - Isolated Reactor Stability |
| 14.2.12.2.33 | Test Number 70 | - Reactor Water Cleanup System |
| 14.2.12.2.34 | Test Number 71 | - Residual Heat Removal System |
| 14.2.12.2.35 | Test Number 74 | - Offgas System |
| 14.2.12.2.36 | Test Number 99 | - Emergency Response Information System |
| 14.2.12.2.37 | Test Number 100 | - Integrated HVAC |
| 14.2.12.2.38 | Test Number 113 | - Service Water System |
| 14.2.12.2.39 | Test Number 114 | - Emergency Closed Cooling System |
| 14.2.12.2.40 | Test Number 115 | - Nuclear Closed Cooling System |
| 14.2.12.2.41 | Test Number 116 | - Turbine Building Closed Cooling System |
| 14.2.12.2.42 | Test Number 117 | - Emergency Service Water |
| 14.2.12.2.43 | Test Number 118 | - Circulating Water System |
| 14.2.12.2.44 | Test Number 119 | - Suppression Pool Cleanup System |
| 14.2.12.2.45 | Test Number 120 | - Feedwater System |
| 14.2.12.2.46 | Test Number 121 | - Extraction Steam System |
| 14.2.12.2.47 | Test Number 122 | - BOP Piping Expansion and Vibration |
| 14.2.12.2.48 | Test Number 123 | - Concrete Temperature Survey |
| 14.2.12.2.49 | Test Number 124 | - Main and Reheat Steam System |
| 14.2.12.2.50 | Test Number 125 | - Condensate System |

| | | |
|---|---|---|
| 14.2.12.2.51 | Test Number 126 | - Main, Reheat Extraction and Misc. Drains |
| 14.2.12.2.52 | Test Number 127 | - LP/HP Heater Drains and Vents |
| 14.2.12.2.53 | Test Number 128 | - Condensate Demineralizer System |
| 14.2.12.2.54 | Test Number 129 | - Steam Seal System |
| 14.2.12.2.55 | Test Number 130 | - Condenser Air Removal System |
| 14.2.12.2.56 | Test Number 131 | - Offgas Vault Refrigeration System |
| 14.2.12.2.57 | Test Number 132 | - Turbine Plant Sampling |
| 14.2.12.2.58 | Test Number 133 | - Loose Parts Monitoring System |