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United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
STARTUP TEST REPORT

Dear Sir or Madam:

In accordance with Technical Specifications 6.9.1.1 for the Harris Nuclear Plant, Carolina Power & Light Company (CP&L) provides the enclosed startup test report following steam generator replacement and power uprate modifications. This report fulfills the requirement for the issuance of a startup test report within 90 days following commencement of commercial operation.

Please refer any questions regarding this submittal to Mr. J. R. Caves at (919) 362-3137.

Sincerely,

R. J. Field

Richard T. Field
Manager-Regulatory Affairs
Harris Nuclear Plant

MSE/mse

Enclosures:
1. Startup Test Report

cc: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. John Goshen, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

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SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400/LICENSE NO. NPF-63
STARTUP TEST REPORT

Harris Nuclear Plant Unit 1, Cycle 11

Startup Test Report

March 28, 2002

CAROLINA POWER AND LIGHT COMANY

Executive Summary

The Harris Technical Specifications provide the following guidance for conditions specifically requiring a startup report and items that should be addressed in the startup report.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, **(2) amendment to the license involving a planned increase in power level,** (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and **(4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.**

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include 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. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

Reviewing the tests described in FSAR Chapter 14, see Table 4.7.3 and determining the tests impacted by SGR/PUR generated the scope of this report.

The Harris Plant is tracking two open testing items, at this time. These open items are as follows:

- EPT-287 Blowdown Flow Versus Differential Pressure (Delta-75 SG) [Reference 5.83]
- EPT-854T Steam Generator Moisture Carryover Test [Reference 5.84]

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1.0 Introduction

1.1 General

This startup report documents test results for Harris Nuclear Plant Unit 1, Cycle 11. This report primarily focuses on the results of the following evolutions:

- Component & Initial Operation Tests
- Operational and Power Ascension Tests

These evolutions were modeled after those described in Chapter 14 of the Harris FSAR. The evolutions were modified to eliminate testing that is no longer appropriate. Examples of tests that were judged to be inappropriate include low power flux mapping and boron worth measurements. In the cases of boron worth measurement alternate testing described in ANSI 19.6.1 [Reference 5.37] was performed. Plant response data demonstrates that HNP control systems can safely and effectively operate following steam generator replacement (SGR) and power uprate (PUR). The Startup Test Program, defined by PLP-632T [Reference 5.1] collected plant data from steady state operation and simulated transients to compare plant response with design predictions, specifications and accident analysis assumptions.

1.2 Cycle Description

Cycle 11 introduces the sixth reload of Siemens High Thermal Performance (HTP) fuel. Sixty-five (65) fresh HTP assemblies were loaded to replace 9 discharged Westinghouse LOPAR assemblies and fifty-six (56) discharged HTP assemblies. Cycle 11 also uses eight (8) Siemens High Thermal Performance (HTP) once burned fuel assemblies that were stored in the spent fuel pool during Cycle 10. The specifics for the core reload design are presented in the Startup and Operations Report for Cycle 11 [Reference 5.2].

1.3 Steam Generator Replacement

Harris Nuclear Plant recently replaced the original D4 (preheater style) steam generators with Δ75 (feed ring style) steam generators. The current steam generators share many operating characteristics with those installed at V.C. Summer Nuclear Power Plant.

1.4 Power Uprate

Harris Nuclear Plant recently uprated core power from 2775 MW_t (NSSS power = 2787.4 MW_t) to 2900 MW_t (NSSS power equals 2912.4 MW_t). All references to reactor power are in percent of rated thermal power. The overall electrical output of the unit was increased approximately 50 megawatts, when compared to the previous cycle.

1.5 T_{avg} Restoration

Harris Nuclear Plant restored T_{avg} to 588.8°F after operating at T_{avg} at 580.8°F during cycles 5 through 10, in order to reduce the degradation rate of the D4 steam generators. Satisfactory operation was verified after restoring T_{avg}. The Δ75 steam generators (SGs) were designed to operate with a T_{avg} of 588.8°F.

2.0 Summary

Safe operation at the increased reactor power is supported by a combination of testing and plant observations. During power ascension safe operation within the analyzed bases was assured by a combination of conservative scaling adjustments and reduced setpoints. A rigorous test program was used to verify that impacted components were capable of meeting design bases assumptions. The design bases bound the measured plant parameters at uprated conditions.

3.0 Component & Initial Operation Test Summaries

3.1 Protection System Engineered Safety Features Actuation Logic Test

There were no changes required to the actual Reactor Protection Logic due to any RFO-10 modifications. There were some changes to the utilization of certain SSPS slave relay contacts due to the removal of the feedwater preheater bypass valves, tempering lines, and large bore piping modifications [Reference .5.78, 5.79 and 5.80]. However, there were no changes to the actual SSPS logic cabinet. This logic was tested twice during the outage. MST-I0072 [Reference 5.43] and MST-I0001 [Reference 5.42] tested Train A and MST-I0073 [Reference 5.44] and MST-I0320 [Reference 5.45] tested Train B. There were some setpoint changes implemented into the PIC cabinets due to the new Lo-Lo Level (25%) and Hi-Hi Level (78%) setpoints required for the replacement steam generators (RSGs). There is an additional 5% operating margin (compared to the old SGs) from the normal operational setpoint of 57% to these setpoints. Also there were some changes implemented to the OTΔT/OPΔT setpoints and time constants due to RSG/PUR. This increased the margin to the turbine runback and reactor trip setpoints. An initial conservative value of ΔT of 62°F (versus a predicted 62.8°F nominal ΔT) was selected and implemented into the scaling, loop calibration procedures, and PICs. This value proved to be conservative during power ascension, particularly at the lower power levels prior to the initial calorimetric at 30% power. This value was used until the results of EPT-156 [Reference 5.36] were incorporated into the revised scaling/calibration procedures for RCS loops B and C. EPT-156 was conducted at 75%, 90% and 100% power. Although the test results showed that only loop B was outside the acceptance criteria, both loop B and C were rescaled/recalibrated to minimize any initial errors at the beginning of the cycle. It should be noted that this test is performed quarterly throughout the cycle to ensure the RCS loop temperatures are maintained within a conservative band. New steam flow/feed flow mismatch setpoints were implemented as well to accommodate the increase in the 100% feed and steam flow from a nominal 4.067 mpph to 4.241 mpph.

EPT-093 [Reference 5.41] was used to establish any new control or protection setpoints that use turbine first stage pressure as the basis for the setting. Plots of first stage turbine pressure versus reactor power and a linear regression curve were generated. The predicted first stage pressure versus reactor power matched the actual curve such that no changes to instrument setpoints were required. This is also substantiated by the fact that the T_{ref} curve, which is also based upon first stage pressure, did not require any instrument re-scaling/recalibration. The T_{ref} and T_{avg} signals are aligned within the tight specifications of the operating procedures.

Overlap testing was performed by the combination of PIC loop calibrations, surveillance testing, and SSPS testing.

3.2 Reactor Protection System Engineered Safety Features Actuation Response Time Test

Qualified Barton model 764 transmitters were used to replace all of the narrow range and wide range steam generator level transmitters. The steam flow transmitters were replaced with qualified Rosemount units. The new SG level transmitters were tested per MST-I0622 [Reference 5.46]. The remaining protection transmitters, which were scheduled to be time response tested during RFO-10, were tested per MST-I0651 [Reference 5.47]. All of the applicable time response related surveillance tests were revised to reflect the new setpoints and time constants; i.e., for the OTΔT/OPΔT channels. The following time response related testing was also completed:

- EST-300 [Reference 5.48]
- EST-301 [Reference 5.49]
- EST-302 [Reference 5.50]
- EST-303 [Reference 5.51]
- EST-304 [Reference 5.52]
- EST-305 [Reference 5.30]
- EST-306 [Reference 5.54]
- EST-307 [Reference 5.55]
- EST-308 [Reference 5.56]
- EST-309 [Reference 5.57]
- EST-310 [Reference 5.58]
- EST-311 [Reference 5.59]

- EST-312 [Reference 5.60]
- EST-313 [Reference 5.61]
- EST-314 [Reference 5.62]
- EST-315 [Reference 5.63]
- EST-318 [Reference 5.66]

Results of these tests show that the acceptance criteria are met for the reactor protection and ESF circuits and components.

One narrow range RTD (TE-432D) was replaced during RFO-10 due to the element's time response falling outside the acceptance criteria of EST-300 [Reference 5.48]. The replacement RTD was satisfactorily time response tested prior to starting the first plateau (350°F) for EST-104 [Reference 5.67]. New calibration curves were generated for the following RTDs:

- TE-422D
- TE-422B2
- TE-422B2-S

The test results of EST-104 [Reference 5.67] show that all RTDs met the acceptance criteria.

3.3 Auxiliary Feedwater System Test

Increased steam pressures, piping reroute due to different azimuthal location of SG nozzle and elimination of the feedwater bypass piping, impacted the auxiliary feedwater system. The impact on the system was demonstrated to be minimal analytically and by a combination of routine Technical Specification surveillances (EST-305 [Reference 5.30] and OST-1087 [Reference 5.32]) and an integrated test that was written to determine the turbine driven setpoint that verified minimal and maximum analyzed flow limits bound routine operation. The integrated test EST-230 [Reference 5.70] demonstrated that a setpoint of 31 psid would deliver the minimum flow for various accident analyses, without exceeding the maximum flow assumed in the steam generator tube rupture (SGTR) analysis.

Pumps Operating	Maximum Total Flow	Maximum Flow Per SG	Minimum Total Flow	Minimum Flow Per SG
TDAFW	N/A	N/A	414 gpm	N/A
TDAFW MDAFW A MDAFW B	1450 gpm	479 gpm	1194 gpm	404 gpm

3.4 Containment Integrated Leak Rate and Structural Integrity Test

Removal and reinstallation of the equipment hatch barrel weld was identified as a potential adverse impact to the containment integrated leak rate. The closure weld was tested via a post-modification retest procedure contained in ESR 97-00805 [Reference 5.71]. The test verified that the reinstalled equipment hatch closure weld is leak tight.

3.5 Piping Thermal Expansion and Dynamic Effects Test

The combination of steam generator replacement (SGR) and power uprate (PUR) impacts primary and secondary systems. Systems were visually observed at the revised operating conditions for adverse reactions from thermal expansion and dynamic effects. Special emphasis was placed on the systems listed in Table 3.6.1, for the stated reasons. The walkdown inspections were recorded using TMM-117 [Reference 5.72].

3.6 Metal Impact Monitoring System Test

The accelerometers that were located on the D4 steam generators were removed and new accelerometers that have the same fit and function were installed on the Δ75 steam generators. The preliminary alignment setpoints were established using conservative estimates. The digital metal impact monitoring system (DMIMS) baseline was established using EPT-012 [Reference 5.20] and data was collected and analyzed using EPT-023 [Reference 5.21]. The revised setpoints are recorded in OP-182 [Reference 5.82]. DMIMS has been restored to an operable condition.

3.7 Feedwater System Test

The following impacted the feedwater system operation:

- Head curve (impeller diameter) increased to overcome higher steam pressure
- Piping dynamics due to conversion from preheater to feedring SG
- Piping dynamics due to elimination of preheater bypass piping
- System dynamics due to level control band on new SGs

The feedwater pump was replaced during RFO-10 and a partial pump curve was verified using PPP-205 [Reference 5.38]. An integrated power ascension program, PLP-632T [Reference 5.1] coordinated this activity. During power ascension steady state data was recorded and analyzed at various power levels. In addition, transient data was recorded and analyzed during the simulated transients created during the performance of EPT-848T [Reference 5.33].

EPT-848T was performed on the main feedwater regulating valves (MFRVs) by injecting a ±5% SG level setpoint deviation at 30% power and 90% power. After making some initial controller adjustments at the 30% power plateau, excellent steam generator water level (SGWL) control response was achieved using the MFRVs. Data was taken during the second main feedwater pump (MFP) start and minimal controller overshoot was experienced and rapid (<10 seconds) recovery to the setpoint was achieved. The test results at 90% power were excellent with minimal overshoot ranging from 0.34% to 1.5% and rapid stabilization at the setpoint.

Testing on the feedwater regulating bypass valves (FRBVs) was not completed due to oscillations on the SGWL control system prior to the test and a manual reactor trip due to a failure of the loop C FRBV. The C FRBV was repaired (new positioner) and the valves were operated in manual for the remainder of power ascension, with no further problems.

Continuation of the power ascension program and eventually full power operation was based on meeting the applicable acceptance criteria. The feedwater design bases bound the steady state, transient system response and accident analyses assumptions.

3.8 Steam Generator Primary Side FOSAR Inspection

Foreign object search and retrieval was conducted on the primary side of the replacement steam generators (RSGs) prior to placing them into service. In addition to the normal debris created by the fabrication and installation processes it was necessary to remove some of the debris that was created by a partial decontamination of the RCS. A sponge-like blasting medium was used to clean away the corrosion product layer and perform a partial decontamination of the RCS piping. The resulting cleanliness level of the RCS piping met the established acceptance criteria (cleanliness level A) and accident analyses assumptions.

3.9 Steam Generator Secondary Side Internal Inspection

Foreign object search and retrieval was conducted on the secondary side of the replacement steam generators (RSGs) prior to placing them into service. The potential objects consisted of debris created by the fabrication and installation processes. The visual inspections were documented in EPT-856T [Reference 5.22]. The digital photographs of the SG secondary surfaces clearly indicate that there are no materials that would adversely impact the SG tubes. The resulting cleanliness level met the established acceptance criteria (cleanliness level B) and accident analyses assumptions.

3.10 Steam Generator Baseline

A pre-service inspection (PSI) of the replacement steam generators (RSGs) open tubes was performed in June 2001 using eddy current testing (ECT). This inspection provides a baseline of eddy current signals against which future ECT inspections will be compared against in order to detect tube degradation.

NEI 97-06 [Reference 5.69] contains the PWR industry's "Steam Generator Program Guidelines" document. CP&L committed to meet the intent of the guidance in NEI 97-06. PLP-651 [Reference 5.40] has extracted the pertinent information from the information from NEI 97-06 for a site SG Program. NEI 97-06, Section 3.1.3 / PLP-651, Section 5.2.3 states that tube integrity shall be assessed following each steam generator inspection.

The assessment of tube integrity from an eddy current test (ECT) inspection consists of two parts. Condition monitoring ("as-found condition) and operational assessment (addresses tube integrity until the next scheduled tube inspection) constitute the two parts. The guidance in NEI 97-06, and its referenced sub-tier EPRI documents, are focused on tube degradation during plant operation and do not detail guidance for pre-service inspection (PSI) for new tubing that has not been in service. Therefore, the following assessment will focus on the ECT inspection results from the PSI, highlighting indications that are attributable to the SG tube manufacturing process at the mill or the tube installation process at the SG fabrication facility.

The "conditioning monitoring" part of the integrity assessment is noted as "ensuring the performance criteria have been met for previous operating cycle" (NEI 97-06, Section 3.1.3). EPRI document TR-107569-V1R5 "PWR SG Examination Guidelines: Rev. 5", states condition monitoring involves an assessment of the "as found" condition of the tubing relative to the performance criteria (Section 5.3). EPRI has an additional document, TR-107621-R1 "SG Integrity Assessment Guidelines", that provides guidance for actually performing tube assessments (Section 8). This document states "the process of condition monitoring involves the evaluation of the inspection results at the end of the operating interval to infer the state of the steam generator tubing for the most recent period of operation.

To satisfy the guideline for performing a condition monitoring assessment following ECT, and realizing this tubing has not been in operational service, this plant modification process has documented the "as-found" integrity of the new tubing based on the PSI results.

The operational assessment portion of the tube integrity assessment addresses the ability of the tubing to meet integrity performance criteria until the next scheduled tube inspection (integrity performance criteria is located in NEI 97-06, Sections 2 and 3.1.3 / PLP-651 Sections 4 and 5.2.3). The industry guidance documents for operational assessment evaluations are oriented toward tubing that has been in service, not for new tubing in RSGs. The EPRI guidance document TR-017569-V1R5 indicates that this assessment is based in part on previous inspection results and repair criteria associated with each degradation mechanism. The new tubing does not have service-induced degradation mechanisms.

3.11 Component Cooling Water System Test

The component cooling water (CCW) system was revised extensively to support SGR, PUR and the activation of C & D spent fuel pools. The flow increase through the CCW heat exchanger resulted in flows that are 160% of the original design rating. Several other heat exchangers in the CCW system experienced similar flow increases. The new CCW pump impellers also develop substantially more head pressure, which caused several components to be re-rated at higher system pressures. Acceptable system performance was verified in EPT-847 [Reference 5.23]. The resulting system performance in each of the five system alignments met the established acceptance criteria and accident analyses assumptions.

Shutdown Cooling

Component	Acceptance Criteria	Measured A/B
RCP 1A Thermal Barrier	Greater than or equal to 40 gpm	40/40
RCP 1A Lower Cooler	Greater than or equal to 5 gpm	6/5
RCP 1A Upper & Lower Oil Cooler	Greater than or equal to 155 gpm	166/164
RCP 1B Thermal Barrier	Greater than or equal to 40 gpm	40/40
RCP 1B Lower Cooler	Greater than or equal to 5 gpm	5/5
RCP 1B Upper & Lower Oil Cooler	Greater than or equal to 155 gpm	162/160
RCP 1C Thermal Barrier	Greater than or equal to 40 gpm	41/40
RCP 1C Lower Cooler	Greater than or equal to 5 gpm	6/5.5
RCP 1C Upper & Lower Oil Cooler	Greater than or equal to 155 gpm	164/160
Letdown Heat Exchanger	Greater than or equal to 330 gpm	375/340
Seal Water Heat Exchanger	Greater than or equal to 235 gpm	250/248
Spent Fuel Pool 1&4A	Greater than or equal to 3425 gpm	3450/
Spent Fuel Pool 1&4B	Greater than or equal to 3425 gpm	/3450
Spent Fuel Pool 2&3A	Greater than or equal to 282 gpm	320/
Spent Fuel Pool 2&3B	Greater than or equal to 282 gpm	/300
RHR Heat Exchanger A	Greater than or equal to 5600 gpm	6041/
RHR Pump A Cooler	Greater than or equal to 5 gpm	5.5/
RHR Heat Exchanger B	Greater than or equal to 5600 gpm	/5965
RHR Pump B Cooler	Greater than or equal to 5 gpm	/5.0
Gross Failed Fuel Detector	Greater than 6 gpm	7/6.1
Reactor Coolant Drain Tank	Greater than or equal to 226 gpm	240/235
Excess Letdown Heat Exchanger	Greater than or equal to 250 gpm	255/250
CCW Heat Exchanger 1A-SA	Less than or equal to 12650 gpm	11518/
CCW Heat Exchanger 1B-SB	Less than or equal to 12650 gpm	/11467

Normal Operation

Component	Acceptance Criteria	Measured A/B
RCP 1A Thermal Barrier	Less than or equal to 60 gpm	53/52.5
RCP 1A Lower Cooler	Less than or equal to 10 gpm	7.5/6.5
RCP 1A Upper & Lower Oil Cooler	Less than or equal to 235 gpm	212/210
RCP 1B Thermal Barrier	Less than or equal to 60 gpm	55/55
RCP 1B Lower Cooler	Less than or equal to 10 gpm	5.75/7
RCP 1B Upper & Lower Oil Cooler	Less than or equal to 235 gpm	210/210
RCP 1C Thermal Barrier	Less than or equal to 60 gpm	53/52.5
RCP 1C Lower Cooler	Less than or equal to 10 gpm	7.5/7.5
RCP 1C Upper & Lower Oil Cooler	Less than or equal to 235 gpm	206/204
Letdown Heat Exchanger	Less than or equal to 1300 gpm	1150/1140
Seal Water Heat Exchanger	Less than or equal to 350 gpm	315/315
Spent Fuel Pool 1&4A	Less than or equal to 4900 gpm	4475/
Spent Fuel Pool 1&4B	Less than or equal to 4900 gpm	/4250
Spent Fuel Pool 2&3A	Less than 750 gpm	410/
Spent Fuel Pool 2&3B	Less than 750 gpm	/385
RHR Pump A Cooler	Less than or equal to 10 gpm	7.6/5.9
RHR Pump B Cooler	Less than or equal to 10 gpm	5.6/7.0
Primary Sample Panel	Less than or equal to 170 gpm	67.3/63.3
Gross Failed Fuel Detector	Less than 12 gpm	11/10.5
Reactor Coolant Drain Tank	Less than or equal to 360 gpm	300/295
Excess Letdown Heat Exchanger	Less than or equal to 355 gpm	330/327
CCW Heat Exchanger 1A-SA	Less than 8500 & 10600 gpm	7887 & 10501
CCW Heat Exchanger 1B-SB	Less than 8500 & 10600 gpm	7911 & 10537

Recirculation @ 243.5°F

Component	Acceptance Criteria	Measured A/B
RHR Heat Exchanger A	7850 to 8050 gpm	7910/
RHR Pump A Cooler	7 to 9 gpm	8.1/
RHR Heat Exchanger B	7850 to 8050 gpm	/7921
RHR Pump B Cooler	7 to 9 gpm	/8.2

Recirculation @ 200°F

Component	Acceptance Criteria	Measured
RHR Heat Exchanger A	Greater than or equal to 4029 gpm	6187
RHR Pump A Cooler	Greater than or equal to 5 gpm	5
RHR Heat Exchanger B	Greater than or equal to 3425 gpm	6187
RHR Pump B Cooler	Greater than or equal to 5 gpm	5
Spent Fuel Pool 1&4A	Greater than or equal to 3425 gpm	3650
Spent Fuel Pool 1&4B	Greater than or equal to 3425 gpm	3650
Spent Fuel Pool 2&3A	Greater than or equal to 282 gpm	310
Spent Fuel Pool 2&3B	Greater than or equal to 282 gpm	310

Min / Max Flow

Component	Acceptance Criteria	Measured A/B
CCW Heat Exchanger A Discharge Pressure	Greater than 57 psig	81
CCW Heat Exchanger B Discharge Pressure	Greater than 57 psig	79
CCW Heat Exchanger A Outlet Flow	12500 to 12650 gpm	12553
CCW Heat Exchanger B Outlet Flow	7850 to 7950 gpm	7919

Service water system flow balance was also verified EPT-250 [Reference 5.74] and EPT-251 [Reference 5.75].

3.12 Safety Injection System Flow Balance

The high head safety injection (HHSI) system was revised extensively to eliminate erosion of the throttle valves and eliminate a potential flow restriction inside the throttle valves [Reference 5.81]. Installing a flow-restricting orifice in each of the twelve HHSI flow paths modified the system resistance. This allowed the throttle valves to be repositioned to a position that significantly reduced the probability of erosion and flow blockage from particles that may pass through the safety injection sump screens. The design bases flow rate was verified with plant procedure EST-206 [Reference 5.77].

4.0 Operational and Power Ascension Test Summaries

4.1 Rod Drop Time Measurement Test

Rod drop tests were performed in accordance with plant procedure EST-724 [Reference 5.3] at hot full flow coolant conditions. Briefly, a bank is selected and pulled to the fully withdrawn position. Opening the reactor trip breakers, thus interrupting the circuit, then drops rods.

The acceptance criteria, from Technical Specifications, require that the rod drop time from the beginning of the drop to dashpot entry be no greater than 2.7 seconds at full core flow and operating temperatures. All rod drop tests were completed within the acceptance criteria. Results of the rod drop testing are included in Table 4.1.1 and Figure 4.2.1

4.2 Reactor Coolant System Flow Measurement Test

Reactor coolant system flow was measured using EST-709 [Reference 5.4]. The various RCS flows were established in the Final PCWG parameters [Reference 5.5]. The corresponding description and numerical values are as follows:

RCS Flow Description	Flow (gpm)
Thermal Design (low)	277,800
FSAR Chapter 15/Technical Specification (minimum)	293,540
Cycle 11 Measured "EST-709"	306,406
Nominal Predicted (best estimate)	306,600
Maximum (limiting) (high)	321,300

The HNP accident analyses are based on the most limiting RCS flow values (minimum or maximum). The RFO-10 (Cycle 11) measured EST-709 [Reference 5.4] flow value is bounded by the various accident analysis values.

Based upon the results of EST-709 [Reference 5.4], nine of the reactor coolant flow protection loops required re-scaling. This re-scaling was implemented into the applicable surveillance tests and these channels were recalibrated on 1/07/02 to 1/08/02. This re-calibration resulted in acceptable reactor coolant flow indication.

4.3 Calibration of Nuclear Instrumentation Test

The intermediate (IR) and power range (PR) detectors were adjusted after refueling (prior to startup), per procedure EPT-008 [Reference 5.6].

The IR adjustment factor (also referred to as the "R-factor") for Cycle 11 was calculated to be 1.1569. This pre-calculated value includes a bias multiplier of 1.1454 based on benchmark data [Reference 5.8] and a correction factor of 1.01 for restoration of T_{avg} from 580.8°F to 588.8°F. The Cycle 11 measured IR R-factor is determined from the N35 and N36 determined trip and rod stop setpoints between the last setpoint determination of Cycle 10 and the first setpoint determination of Cycle 11. The post-startup setpoint determination was performed under procedure EPT-009 [Reference 5.7]. These data are included in Table 4.3.1. The actual IR trip setpoint, prior to recalibration, was calculated as N35 = 21.65% and N36 = 19.16%. The Technical Specification maximum allowable limit is 30.9%.

The PR adjustment factor (also referred to as the "R-factor") for Cycle 11 was calculated to be 1.2442. This pre-calculated value includes a bias multiplier of 1.1963, based on benchmark data [Reference 5.8] and a correction factor of 1.04 for restoration of T_{avg} from 580.8°F to 588.8°F. The Cycle 10 measured PR R-factor is determined from the N41, N42, N43, and N44 top and bottom HFP normalized detector currents between the last incore/excore calibration of Cycle 10 (Flux Map 322) and the first incore/excore calibration of Cycle 11 (Flux Map 327). The incore/excore calibrations were performed under procedure EST-911 [Reference 5.15]. These data are included in Table 4.3.2.

4.4 Flux Distribution Measurement Test

Core power distributions for Cycle 11 are measured by processing moveable detector traces with the INPAX-W code, which is a module of the POWERTRAX core monitoring system. Power distribution maps for the power ascension flux maps are included as Figures 4.4.1 through 4.4.3.

The initial low power flux map is taken near 30% power to verify core loading is as designed. Map @30% was taken immediately after stabilizing power near 30% (before equilibrium xenon was established) for core verification. The maximum difference between measured and calculated powers was 6.7% (location B-08), as shown in Figure 4.4.1. The Map @100% indicated that the limiting fuel assembly (L-09) had an F-dh (peak pin) fraction of limit of 0.918, see Figure 4.4.3. The following flux maps passed acceptance criteria contained in FMP-200 [Reference 5.13].

- Map 325 @ 30% (verifying that the core was loaded as designed)
- Map 326 @ 75%
- Map 327 @ 100%

The core operating limits report (COLR) [Reference 5.11] requires a minimum of 44 measured traces for the core verification flux map; following the core verification flux map, all flux maps require a minimum of 38 measured traces.

4.5 Core Performance Test

The flux maps following core loading verification are taken to verify compliance with Technical Specification requirements and limits on hot channel factors, quadrant power tilts, and to establish allowed power limits for successive power ascension. The following flux maps were taken near 75% and 100% power, respectively.

- Map 326 @75%
- Map 327 @100%

All flux maps allowed full power operation with no additional intermediate power level maps other than those required per PLP-632T [Reference 5.1]. Table 4.5.1 includes pertinent statistics for evaluating map quality and core parameters, which must be monitored.

The flux maps allowed power ascension and then full power operation based on meeting the applicable acceptance criteria.

4.6 Power Coefficient Measurement Test

The RMAS reactivity computer is set up before LPPT by procedure EPT-026 [Reference 5.9]. Comparing period measurements to the startup rate indicated by the computer performed following initial criticality, checkouts the reactivity computer. The six-group constants input to the reactivity computers were provided by SPC and are listed in Table 4.6.1.

The reactivity computer checkout requires that the positive and negative reactivity insertion period checks agree within 5%. Results of the reactivity computer checkout are included in Table 4.6.1. The reactivity computer acceptance criteria for Cycle 11 were met.

The isothermal temperature coefficient (ITC) is measured at ARO, HZP to verify that Technical Specification requirements limiting the ARO moderator temperature coefficient (MTC) to less than or equal to +5 pcm/°F at HZP. Should the MTC exceed the acceptance criteria, rod withdrawal limits for startup and power ascension must be established. The MTC is derived from the measured ITC using the equation below, where the doppler temperature coefficient (DTC) is -1.54 pcm/°F [Reference 5.2].

$$ITC = MTC + DTC$$

The low power physics testing (LPPT) is performed under a single test procedure (EST-923 [Reference 5.10]). EST-923 covers:

- Initial criticality
- Reactivity computer period checks
- Test band determination (point of adding heat determination)
- ARO boron endpoint
- Temperature coefficient determination
- Rod swap

Results for Cycle 11 LPPT and the corresponding acceptance criteria are listed in Table 4.6.2. This table (Table 4.6.2) also contains test results from sections 4.7 and 4.8.

4.7 Control Rod Reactivity Worth Test

The worths of the control and shutdown banks are measured using the rod swap technique. The reference bank (for Cycle 11, control bank B) was measured via boron swap. The remaining banks were measured fully inserted in the presence of the reference bank in a critical configuration.

The **review criteria** for the rod worths are as follows:

1. The absolute value of the percent difference between measured and predicted integral worth of the reference bank is less than 10%.

2. For all banks other than the reference banks, the absolute value of the percent difference between measured and predicted worths is less than 15% or the absolute value of the reactivity difference between measured and predicted worths is less than 100 pcm, whichever is greater.

The **acceptance criteria** require that the sum of the measured worths be between 90% and 110% of the sum of the predicted worths.

Results for Cycle 11 LPPT and the corresponding acceptance criteria are listed in Table 4.6.2. Table 4.7.1 presents the integral and differential worth of the reference bank. Figures 4.7.1 and 4.7.2 graphically compare the predicted and measured integral and differential rod worths for the reference bank.

4.8 Boron Endpoint Measurement – All Rods Out Test

The boron endpoint is measured at ARO and again with the reference bank (control bank B) inserted. The acceptance criteria for boron endpoint measurement require the ARO endpoint to be within 50 ppm of the predicted value.

Results for Cycle 11 LPPT and the corresponding acceptance criteria are listed in Table 4.6.2.

4.9 RTD/TC Cross Calibration Test

EST-104 [Reference 5.67] is performed at three temperature plateaus, 350°F, 450°F, and approx 548°F. The data at the 350°F plateau was acceptable for all of the RTDs. However, at the 450°F plateau, TE-422D and TE-422B2 both showed low out of spec readings. Low insulation resistance was found to be the cause of the problem. A self-heating test was performed to dry the insulation. Follow-up insulation resistance testing at the 450°F, and approx 548°F plateaus were found to be acceptable.

4.10 Steam Generator Moisture Carryover Test

A moisture carryover test, using lithium, is planned for the summer of 2002. Operating for a short period will allow nominal SG tube fouling to occur before taking test data. The moisture carryover limit for the replacement steam generators is 0.10%.

The Harris calorimetric procedure, OST-1204 [Reference 5.73], currently uses a moisture carryover fraction of 0.00.

4.11 Load Swing Test

A planned load swing test EPT-849T [Reference 5.34] was eliminated from scope. Acceptable plant performance during a load swing is the result of several control systems (rod control, steam dumps, steam generator level, feedwater flow, etc.) working in an integrated manner. SGR/PUR impacted only the steam generator level and feedwater flow control systems. SGR/PUR setpoints established additional margin between the nominal and the expected value during a load swing. Simulator testing demonstrated that if the feedwater regulating valves are tuned to produce acceptable results during EPT-848T [Reference 5.33] then moderate (<25%) load swing tests were not challenging. After reviewing the level swing testing EPT-848T [Reference 5.33] results Westinghouse confirmed that the planned load swing test EPT-849T [Reference 5.34] would generate minimal additional data.

4.12 Reactor Coolant System Leakrate Test

The RCS was severed at the connections to the original SGs (D4 SGs) and narrow groove welds were utilized to reconnect the RSGs (Δ75 SGs). ASME Section XI code case N416-1 was invoked to substitute a normal in-service visual inspection in lieu of an ASME Section III hydrostatic test. The visual inspections of the RCS piping and components were documented in EST-201 [Reference 5.39]. The visual inspection was further confirmed by OST-1026 [Reference 5.19] leakage results during power ascension and subsequent operation.

4.13 Main Steam and Feedwater Systems Test

The integrated power ascension program was coordinated by PLP-632T [Reference 5.1]. During power ascension steady state data was recorded and analyzed at various power levels. In addition, data was recorded and analyzed during the simulated transients created during the performance of EPT-848T [Reference 5.33].

EPT-848T [Reference 5.33] was completed on the main feedwater regulating valves (MFRVs) by injecting a ±5% setpoint deviation at 30% power and 90% power. Also the FRV control response was checked during the second main feedwater pump (MFP) start. After making some initial controller adjustments at the 30% power plateau, excellent steam generator water level (SGWL) control response was achieved at the test plateaus and throughout power ascension.

Testing on the feedwater regulating bypass valves (FRBVs) was not completed due to oscillations on the SGWL control system and a manual reactor trip due to a failure of the loop C FRBV valve. During the subsequent restart the FRBV worked satisfactorily in the manual mode.

The MFRVs and FRBVs were stroke tested and the data recorded in EPT-120 [Reference 5.76]. Design bases assumptions, with respect to feedwater isolation, were validated.

EPT-852T [Reference 5.68] was completed with acceptable results. These loops were calibrated per the input of ESR 00-00262, which was based upon a "best estimate" calculation of the steam flow differential pressure across the new steam generator outlet restrictor orifice and the steam flow piping. Data was taken at power plateaus of 30%, 50%, 75%, 90%, and 100% power. The steam flow transmitter differential pressures (D/Ps) and the steam flow proportionality constant resulted in the steam flow instrumentation providing acceptable results without any instrument rescaling or calibrations required.

Continuation of the power ascension power and eventually full power operation was based on meeting the applicable acceptance criteria. The design bases bound the main steam/feedwater steady state and transient system response and accident analyses assumptions.

4.14 Plant Performance Test

The integrated power ascension program was coordinated by PLP-632T [Reference 5.1]. During power ascension steady state data was recorded and analyzed at various power levels. In addition, data was recorded and analyzed during the simulated transients created during the performance of EPT-848T [Reference 5.33]. The plant performance baseline data was reviewed to ensure that the plant performance met design bases assumptions.

4.15 DEH Changes

The digital electric hydraulic (DEH) system was modified to increase turbine blade reliability at low power levels. To accomplish this objective the arc of admission was increased from 180 degrees to 270 degrees. The DEH system was also tuned to increase controllability at higher steam pressures and higher power levels. The changes to the DEH system were verified to be acceptable in CM-C004 [Reference 5.27], ORT-8001 [Reference 5.26], and SCP-006 [Reference 5.53].

5.0 References

5.1 PLP-632T "Power Ascension Testing Program After R10 Refueling Outage (Steam Generator & Power Upgrade Modifications"
5.2 Startup and Operations Report for Cycle 11 "SOR"
5.3 EST-724 "Shutdown and Control Rod Drop Test Using Computer"
5.4 EST-709 "Reactor Coolant System Flow Determination By Calorimetric"
5.5 CQL-98-030 "Final PCWG Parameters for the SGR/Uprating Analysis and Licensing Project"
5.6 EPT-008 "Intermediate and Power Range Detector Setpoint Determination"
5.7 EPT-009 "Intermediate Detector Setpoint Determination"
5.8 HNP-F/NFSA-0010 "HNP BOC NI Adjustment"
5.9 EPT-026 "RMAS Setup and Operation"
5.10 EST-923 "Initial Criticality and Low Power Physics Testing"
5.11 PLP-106 "Core Operating Limits Report"
5.12 FMP-201 "Incore Flux Mapping Using POWERTRAX"
5.13 FMP-200 "Full Core Flux Map Review Checklist (POWERTRAX Version)"
5.14 EST-910 "Hot Channel Factors Test (POWERTRAX Version)"
5.15 EST-911 "Incore/Excore Detector Calibration Using POWERTRAX"
5.16 EST-216 "Steam Generator Tube Indication Tracking and Reporting Procedure"
5.17 EPT-244T "ABB/Combustion Engineering XS10370016, Temporary Procedure for Tube Plugging and Repair of Steam Generator Tubes (Expires 12/31/2001)"
5.18 ISI-202 "Safety-Related Component Support (Hangers and Snubbers) Examination and Testing Program"
5.19 OST-1026 "Reactor Coolant System Leakage Evaluation, Computer Calculation, Daily Interval, Modes 1-2-3-4"
5.20 EPT-012 "Digital Metal Impact Monitoring System (DIMIMS) Baseline Data Procedure"
5.21 EPT-023 "Digital Metal Impact Monitoring System Data Analysis Procedure"
5.22 EPT-856T "Steam Generator Secondary Side Internal Inspection"
5.23 EPT-847 "CCW Pump Performance Test and System Flow Balance"
5.24 EPT-243T "ABB/Combustion Engineering XS10370016, Temporary Procedure for Data Acquisition for Steam Generator Tube Examinations (Expires 12/31/2001)"
5.25 EPT-242T "ABB/Combustion Engineering XS10370016, Temporary Procedure for Eddy Current Examinations of Steam Generator Tubes (Expires 12/31/2001)"
5.26 ORT-8001 "DEH Computer System Dynamic Simulation Test"
5.27 CM-C004 "DEH Computer Reload and Restart"
5.28 OST-1825 "Safety Injection: ESF Response Time, Train A 18 Month Interval on a Staggered Test Basis Mode 5-6"
5.29 OST-1853 "Feedwater Isolation ESF Response Time Trains A and B, 18 Month Interval Modes 3, 5, 6"

5.30 EST-305 "Engineered Safety Features Response Time Evaluation Motor Driven Auxiliary Feedwater Pumps"
5.31 EPT-196 "Motor Driven Auxiliary feedwater Pumps Performance Test"
5.32 OST-1087 "Motor Driven Auxiliary Feedwater Pumps Full Flow Test Quarterly Interval"
5.33 EPT-848T "S/G Level Swings"
5.34 EPT-849T "Load/Swing Reject Test"
5.35 EST-919 "Incore Versus Excore Axial Flux Difference Comparison (POWERTRAX Version)"
5.36 EPT-156 "Reactor Coolant System (RCS) ΔT Scaling at 100% Reactor Power"
5.37 ANSI 19.6.1 "Reload Startup Physics Test for Pressurized Water Reactors"
5.38 PPP-205 "Main Feed Pump Performance Test"
5.39 EST-201 "ASME System Pressure Tests"
5.40 PLP-651 "Steam Generator Program"
5.41 EPT-093 "Turbine First Stage Pressure Data"
5.42 MST-I0001 "Train A Solid State Protection System Actuation Logic & Master Relay Test"
5.43 MST-I0072 "Train A 18 Month Manual Reactor Trip Solid State Protection System Actuation Logic & Master Relay Test"
5.44 MST-I0073 "Train B 18 Month Manual Reactor Trip Solid State Protection System Actuation Logic & Master Relay Test"
5.45 MST-I0320 "Train B Solid State Protection System Actuation Logic & Master Relay Test"
5.46 MST-I0622 "Bench Transmitter Response Time Test"
5.47 MST-I0651 "Transmitter Noise Analysis Time Response Test"
5.48 EST-300 "Reactor Trip response Time Evaluation"
5.49 EST-301 "Engineered Safety Features Response Time Evaluation Safety Injection"
5.50 EST-302 "Engineered Safety Features Response Time Evaluation Feedwater Isolation"
5.51 EST-303 "Engineered Safety Features Response Time Evaluation Containment Phase "A" Isolation"
5.52 EST-304 "Engineered Safety Features Response Time Evaluation Containment Ventilation Isolation"
5.53 SCP-006 "Throttle Valve and Governor Valve Calibration Procedure"
5.54 EST-306 "Engineered Safety Features Response Time Evaluation Emergency Service Water Pumps"
5.55 EST-307 "Engineered Safety Features Response Time Evaluation Containment Fan Coolers"
5.56 EST-308 "Engineered Safety Features Response Time Evaluation Steamline Isolation"
5.57 EST-309 "Engineered Safety Features Response Time Evaluation Containment Spray"
5.58 EST-310 "Engineered Safety Features Response Time Evaluation Phase-B Isolation"
5.59 EST-311 "Engineered Safety Features Response Time Evaluation Turbine Trip"
5.60 EST-312 "Engineered Safety Features Response Time Evaluation Turbine Driven AFW Pump"

5.61 EST-313 "Engineered Safety Features Response Time Evaluation Switchover to Recirculation Sumps with SI"
5.62 EST-314 "Engineered Safety Features Response Time Evaluation Switchover to Recirculation Sumps with Containment Spray"
5.63 EST-315 "Engineered Safety Features Response Time Evaluation Containment Purge Isolation"
5.64 EST-316 "Emergency Sequencing System 1A-SA Response Time Test"
5.65 EST-317 "Emergency Sequencing System 1B-SB Response Time Test"
5.66 EST-318 "Engineered Safety Features Response Time Evaluation Isolate AFW to the Affected SG"
5.67 EST-104 "Incore Thermocouple and RTD Cross Calibration Data Compilation"
5.68 EPT-852T "Temporary Procedure for Calibration of Steam Flow Instruments"
5.69 NEI 97-06 "Steam Generator Program Guidelines"
5.70 EST-230 "Determination of Turbine Driven Auxiliary Feedwater Pump Differential Pressure Controller Setpoint"
5.71 ESR 97-00805 "SGR Containment Modifications"
5.72 TMM-117 "System Walkdowns and Observations"
5.73 OST-1204 "Power Range Heat Balance, Manual Calculation, Daily Interval, Mode 1 (Above 15% Power)"
5.74 EPT-250 "A Train ESW Flow Verification/Balance"
5.75 EPT-251 "B Train ESW Flow Verification/Balance"
5.76 EPT-120 "Stroke Timing of the Main and Bypass Feedwater Regulating Valves"
5.77 EST-206 "ECCS Flow Balance"
5.78 ESR 99-00468 "RSG Blowdown and Wet Layout System setpoints"
5.79 ESR 98-00537 "Tempering Lines Modification"
5.80 ESR 97-00807 "SGR Large Bore Modifications"
5.81 ESR 99-00407 "ECCS Flow Balancing Orifice Installation"
5.82 OP-182 "Digital Metal Impact Monitoring System"
5.83 EPT-287 "Blowdown Flow Versus Differential Pressure (Delta-75 SG)"
5.84 EPT-854T "Steam Generator Moisture Carryover Test"

Table 3.6.1
Piping Thermal Expansion and Dynamic Effects
Special Emphasis Systems

System	Reason for Special Emphasis
Reactor Coolant System	The D4 steam generators were removed from the system and Δ75 steam generators installed. The replacement steam generators (Δ75) have a higher center of gravity and are predicted to have a slightly greater RCS flow.
Feedwater System (Inside Containment)	The feedwater piping was rerouted from the bottom of the steam generators (D4 - preheater style) to the top of the replacement steam generators (Δ75 – feedring style). The preheater bypass flow path was eliminated, increasing the feedwater flow through the main feedwater piping by 30% (prior to PUR). PUR further increased feedwater flow by an additional 4.5%.
Feedwater System (Outside Containment)	Feedwater flow has increased by 4.5%. This change impacts the feedwater system and various support systems (condensate, heater drains, etc.). The preheater bypass flow path and tempering line flow path was eliminated.
Steam System	Steam flow has increased by 4.5%. This change impacts the steam system and various support systems (moisture separators, heater drains, etc.).
Component Cooling Water	In conjunction with SGR, PUR, and activation of the C & D spent fuel pools Harris upgraded the CCW system to restore operating margin. Significant flow increases were experienced at the CCW heat exchangers (160% of original design flow rating) and similar increases were observed throughout the system. Larger CCW pump impellers increased the flow and pressure transient response associated with pump starts and other system transients.

Table 4.1.1
Control Rod Drop Times[1]

Control Banks			
Rod Bank	Core Location	Time to Dashpot Entry (sec)	Time to Bottom of Dashpot (sec)
CBA	F-02	1.52	2.02
	B-10	1.60	2.05
	K-14	1.66	2.09
	P-06	1.49	1.92
	K-02	1.56	1.97
	B-06	1.80	2.25
	F-14	1.54	1.93
	P-10	1.53	1.90
CBB	F-04	1.54	1.92
	D-10	1.53	1.88
	K-12	1.50	1.94
	M-06	1.54	1.88
	K-04	1.51	1.89
	D-06	1.54	1.94
	F-12	1.52	1.89
	M-10	1.51	1.86
CBC	D-04	1.52	1.94
	D-12	1.50	1.94
	M-12	1.48	1.93
	M-04	1.53	1.96
	H-06	1.53	1.93
	F-08	1.49	1.85
	H-10	1.52	1.90
	K-08	1.51	1.90
CBD	H-02	1.54	1.95
	B-08	1.65	2.08
	H-14	1.52	1.93
	P-08	1.56	2.00
	F-06	1.55	1.94
	F-10	1.54	1.91
	K-10	1.54	1.91
	K-06	1.53	1.90

Shutdown Banks			
Rod Bank	Core Location	Time to Dashpot Entry[2] (sec)	Time to Bottom of Dashpot (sec)
SBA	G-03		2.03
	C-09		2.02
	J-13		1.96
	N-07		1.99
	J-03		1.98
	C-07		2.08
	G-13		1.98
	N-09		1.98
SBB	E-05		2.01
	E-11		2.02
	L-11		1.98
	L-05		1.98
	G-07		1.97
	G-09		1.98
	J-09		1.97
	J-07		1.96
SBC	E-03		1.90
	C-11		1.91
	L-13		1.94
	N-05		1.90

1 Measured data obtained from Cycle 11 (RFO 10) performance of EST-724 [Reference 5.3].
2 Dashpot entry times were not recorded for Shutdown Banks. All rod bottom data are within TS Limit of 2.7 seconds for dashpot entry.

Figure 4.2.1
Control Rod Drop Times

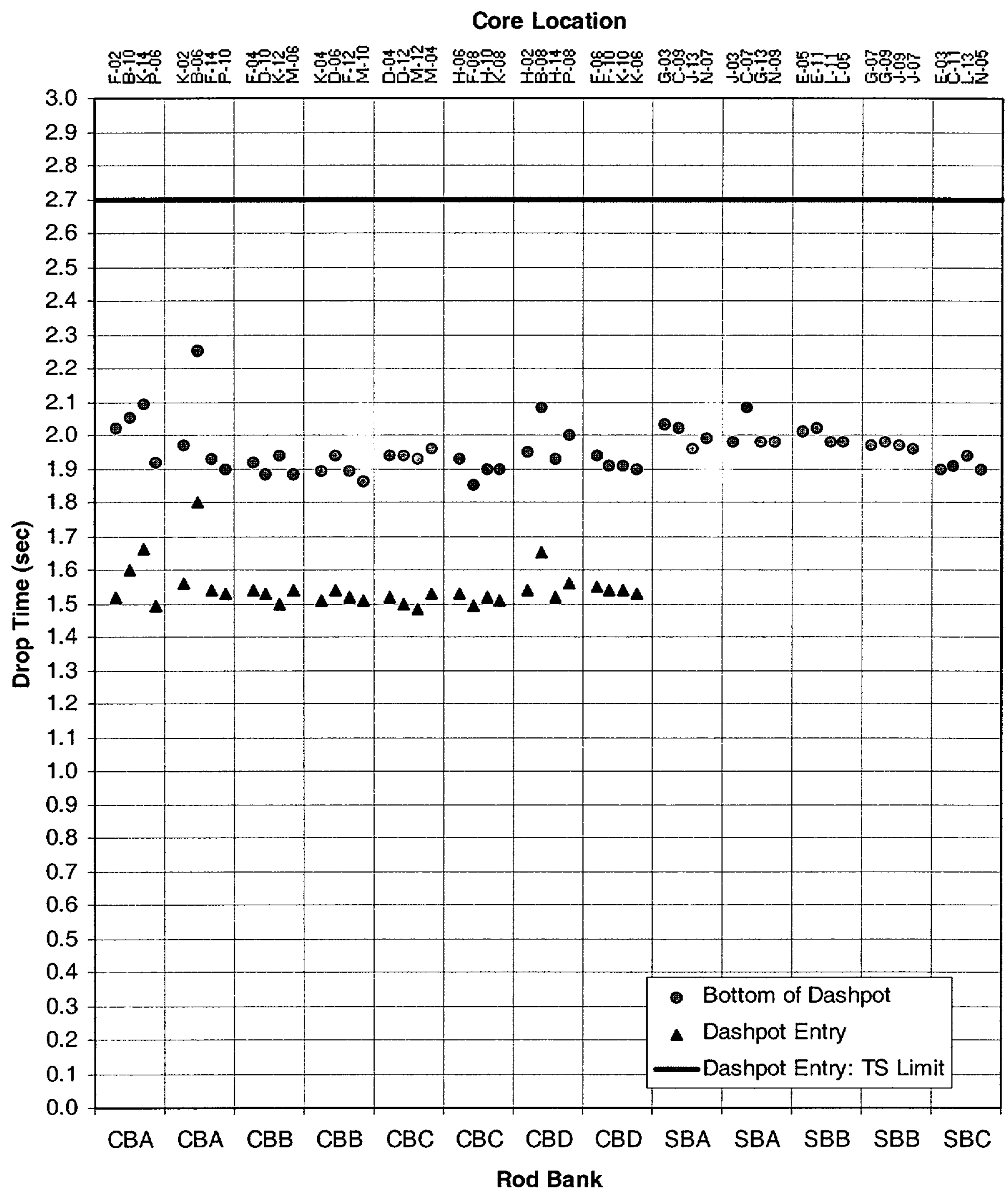


Table 4.3.1

Intermediate Range Detector R-factor Determination

Cycle		N35 Trip	N35 Rod Stop	N36 Trip	N36 Rod Stop	Average
10[1]	Startup 79	6.10E-5	4.88E-5	5.32E-5	4.26E-5	
11[2]	Startup 82	8.16E-5	6.53E-5	8.38E-5	6.70E-5	
C11 R-value	(C11/C10)	1.338	1.338	1.575	1.573	**1.457**

1 Cycle 10 data obtained from Startup 79 performance of Reference 5.7 (5/25/00)
2 Cycle 11 data obtained from Startup 82 performance of Reference 5.7 (1/11/02)

Table 4.3.2

Power Range Detector R-factor Determination

Detector		Cycle 10[1] (Flux Map 322)	Cycle 11[1] (Flux Map 327)	C9 R-value (C11 / C10)
N41	top	137.2	159.2	
	bottom	155.6	184.2	
	sum	292.8	343.4	1.1728
N42	top	149.2	177.4	
	bottom	171.6	208.2	
	sum	320.8	385.6	1.2020
N43	top	169.4	201.9	
	bottom	189.5	229.2	
	sum	358.9	431.1	1.2012
N44	top	136.5	160.8	
	bottom	164.3	199.2	
	sum	300.8	360.0	1.1968
Average				**1.1932**

1 Power Range Data taken from respective performance of EST-911 [Reference 5.15].

Figure 4.4.1
Flux Map 325 Measured vs. Calculated Powers

Measured Power
Calculated Power
Percent Difference

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							.297 .299 .7	.315 .323 2.5	.298 .302 1.3						
2					.386 .391 1.3	.998 1.005 .7	1.204 1.212 .7	.943 .962 2.0	1.203 1.217 1.2	1.003 1.008 .5	.392 .391 -.3				
3				.467 .473 1.3	1.188 1.214 2.2	1.238 1.247 .7	1.240 1.231 -.7	1.144 1.170 2.3	1.224 1.235 .9	1.247 1.248 .1	1.221 1.215 -.5	.473 .475 .4			
4			.463 .473 2.2	1.275 1.305 2.4	1.203 1.250 3.9	1.065 1.076 1.0	1.088 1.090 .2	1.220 1.249 2.4	1.082 1.087 .5	1.077 1.071 -.6	1.272 1.250 -1.7	1.321 1.309 -.9	.478 .475 -.6		
5		.387 .390 .8	1.181 1.211 2.5	1.236 1.247 .9	1.028 1.026 -.2	1.102 1.096 -.5	1.276 1.258 -1.4	1.182 1.168 -1.2	1.256 1.244 -1.0	1.088 1.069 -1.7	1.079 1.029 -4.6	1.267 1.253 -1.1	1.215 1.216 .1	.381 .391 2.6	
6		1.010 1.005 -.5	1.251 1.244 -.6	1.080 1.069 -1.0	1.101 1.067 -3.1	.885 .872 -1.5	1.077 1.051 -2.4	1.116 1.062 -4.8	1.052 1.040 -1.1	.877 .873 -.5	1.121 1.098 -2.1	1.088 1.077 -1.0	1.244 1.248 .3	.987 1.006 1.9	
7	.302 .300 -.7	1.228 1.213 -1.2	1.262 1.232 -2.4	1.106 1.085 -1.9	1.270 1.242 -2.2	1.058 1.039 -1.8	1.148 1.130 -1.6	1.000 .984 -1.6	1.110 1.130 1.8	1.065 1.051 -1.3	1.279 1.259 -1.6	1.102 1.090 -1.1	1.218 1.232 1.1	1.176 1.212 3.1	.289 .300 3.8
8	.315 .322 2.2	.968 .960 -.8	1.188 1.168 -1.7	1.270 1.247 -1.8	1.188 1.165 -1.9	1.081 1.061 -1.9	1.003 .984 -1.9	1.192 1.170 -1.8	1.001 .984 -1.7	1.112 1.061 -4.6	1.189 1.167 -1.9	1.251 1.248 -.2	1.146 1.170 2.1	.902 .962 6.7	.307 .323 5.2
9	.300 .299 -.3	1.221 1.209 -1.0	1.246 1.229 -1.4	1.106 1.088 -1.6	1.280 1.257 -1.8	1.069 1.050 -1.8	1.153 1.129 -2.1	1.002 .983 -1.9	1.151 1.130 -1.8	1.063 1.039 -2.3	1.254 1.243 -.9	1.081 1.086 .5	1.207 1.234 2.2	1.167 1.215 4.1	.289 .301 4.2
10		1.014 1.003 -1.1	1.261 1.245 -1.3	1.089 1.075 -1.3	1.110 1.095 -1.4	.881 .872 -1.0	1.066 1.039 -2.5	1.077 1.060 -1.6	1.061 1.050 -1.0	.873 .872 -.1	1.065 1.068 .3	1.058 1.069 1.0	1.219 1.245 2.1	.967 1.006 4.0	
11		.394 .390 -1.0	1.224 1.212 -1.0	1.262 1.249 -1.0	1.036 1.026 -1.0	1.076 1.067 -.8	1.255 1.242 -1.0	1.174 1.166 -.7	1.260 1.257 -.2	1.097 1.095 -.2	1.010 1.026 1.6	1.232 1.247 1.2	1.190 1.211 1.8	.381 .391 2.6	
12			.476 .473 -.6	1.311 1.305 -.5	1.251 1.246 -.4	1.069 1.068 -.1	1.082 1.084 .2	1.241 1.246 .4	1.076 1.088 1.1	1.057 1.075 1.7	1.230 1.249 1.5	1.296 1.305 .7	.466 .473 1.5		
13				.474 .473 -.2	1.208 1.210 .2	1.237 1.244 .6	1.218 1.232 1.1	1.147 1.168 1.8	1.194 1.228 2.8	1.194 1.244 4.2	1.179 1.212 2.8	.465 .473 1.7			
14					.386 .390 1.0	.992 1.005 1.3	1.189 1.213 2.0	.930 .960 3.2	1.161 1.209 4.1	.964 1.003 4.0	.378 .390 3.2				
15							.289 .301 4.2	.312 .323 3.5	.288 .299 3.8						

Figure 4.4.2

Flux Map 326 Measured vs. Calculated Powers

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							.306	.318	.309						
							.315	.347	.317						
							2.9	9.1	2.6						
2					.383	.967	1.182	.996	1.189	.976	.390				
					.391	.980	1.196	1.014	1.201	.983	.392				
					2.1	1.3	1.2	1.8	1.0	.7	.5				
3				.447	1.134	1.193	1.207	1.169	1.208	1.204	1.162	.460			
				.468	1.166	1.207	1.208	1.165	1.212	1.208	1.167	.470			
				4.7	2.8	1.2	.1	-.3	.3	.3	.4	2.2			
4			.458	1.215	1.165	1.057	1.083	1.213	1.082	1.059	1.215	1.231	.463		
			.468	1.245	1.209	1.066	1.087	1.238	1.084	1.061	1.209	1.248	.470		
			2.2	2.5	3.8	.9	.4	2.1	.2	.2	-.5	1.4	1.5		
5		.385	1.135	1.206	1.076	1.137	1.288	1.195	1.274	1.111	1.066	1.183	1.145	.384	
		.390	1.164	1.206	1.027	1.118	1.269	1.176	1.256	1.093	1.029	1.211	1.168	.391	
		1.3	2.6	.0	-4.6	-1.7	-1.5	-1.6	-1.4	-1.6	-3.5	2.4	2.0	1.8	
6		.972	1.197	1.064	1.111	.975	1.122	1.160	1.112	.980	1.138	1.063	1.192	.960	
		.980	1.205	1.059	1.091	.963	1.095	1.096	1.085	.964	1.120	1.067	1.208	.981	
		.8	.7	-.5	-1.8	-1.2	-2.4	-5.5	-2.4	-1.6	-1.6	.4	1.3	2.2	
7	.312	1.190	1.208	1.089	1.268	1.097	1.187	1.056	1.188	1.117	1.288	1.096	1.191	1.163	.305
	.316	1.198	1.209	1.083	1.254	1.084	1.175	1.027	1.176	1.095	1.269	1.087	1.209	1.197	.315
	1.3	.7	.1	-.6	-1.1	-1.2	-1.0	-2.7	-1.0	-2.0	-1.5	-.8	1.5	2.9	3.3
8	.335	1.003	1.164	1.245	1.183	1.111	1.048	1.249	1.066	1.135	1.194	1.238	1.138	.950	.329
	.346	1.011	1.162	1.236	1.174	1.095	1.027	1.212	1.028	1.096	1.176	1.237	1.164	1.013	.347
	3.3	.8	-.2	-.7	-.8	-1.4	-2.0	-3.0	-3.6	-3.4	-1.5	-.1	2.3	6.6	5.5
9	.312	1.193	1.211	1.097	1.295	1.114	1.202	1.064	1.255	1.108	1.268	1.080	1.185	1.151	.303
	.314	1.194	1.206	1.085	1.267	1.094	1.175	1.027	1.175	1.084	1.255	1.083	1.211	1.199	.316
	.6	.1	-.4	-1.1	-2.2	-1.8	-2.2	-3.5	-6.4	-2.2	-1.0	.3	2.2	4.2	4.3
10		.981	1.214	1.075	1.131	.970	1.102	1.117	1.126	.975	1.099	1.057	1.184	.943	
		.979	1.205	1.065	1.117	.962	1.084	1.095	1.094	.962	1.091	1.059	1.205	.981	
		-.2	-.7	-.9	-1.2	-.8	-1.6	-2.0	-2.8	-1.3	-.7	.2	1.8	4.0	
11		.393	1.173	1.216	1.029	1.095	1.259	1.176	1.279	1.133	1.033	1.211	1.150	.382	
		.390	1.165	1.208	1.027	1.091	1.254	1.174	1.267	1.117	1.026	1.206	1.164	.391	
		-.8	-.7	-.7	-.2	-.4	-.4	-.2	-.9	-1.4	-.7	-.4	1.2	2.4	
12			.476	1.249	1.207	1.057	1.077	1.229	1.079	1.058	1.212	1.271	.456		
			.468	1.244	1.205	1.058	1.082	1.236	1.085	1.064	1.208	1.244	.468		
			-1.7	-.4	-.2	.1	.5	.6	.6	.6	-.3	-2.1	2.6		
13				.469	1.160	1.197	1.195	1.144	1.178	1.162	1.147	.470			
				.468	1.163	1.204	1.209	1.162	1.206	1.205	1.164	.468			
				-.2	.3	.6	1.2	1.6	2.4	3.7	1.5	-.4			
14					.388	.968	1.176	.983	1.155	.947	.382				
					.390	.980	1.198	1.011	1.194	.978	.390				
					.5	1.2	1.9	2.8	3.4	3.3	2.1				
15							.306	.337	.305						
							.316	.346	.314						
							3.3	2.7	3.0						

Measured Power
Calculated Power
Percent Difference

Figure 4.4.3

Flux Map 327 Measured vs. Calculated Powers

Measured Power
Calculated Power
Percent Difference

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							.315 .322 2.2	.337 .361 7.1	.317 .325 2.5						
2					.381 .387 1.6	.948 .959 1.2	1.171 1.185 1.2	1.032 1.050 1.7	1.172 1.189 1.5	.951 .962 1.2	.385 .388 .8				
3				.446 .461 3.4	1.105 1.131 2.4	1.169 1.181 1.0	1.197 1.197 .0	1.154 1.165 1.0	1.190 1.200 .8	1.172 1.182 .9	1.123 1.133 .9	.449 .463 3.1			
4			.452 .461 2.0	1.174 1.203 2.5	1.144 1.182 3.3	1.053 1.061 .8	1.087 1.087 .0	1.217 1.233 1.3	1.082 1.085 .3	1.052 1.056 .4	1.186 1.182 -.3	1.188 1.207 1.6	.455 .463 1.8		
5		.382 .387 1.3	1.104 1.129 2.3	1.171 1.179 .7	1.045 1.030 -1.4	1.149 1.138 -1.0	1.299 1.277 -1.7	1.201 1.184 -1.4	1.283 1.265 -1.4	1.134 1.113 -1.9	1.073 1.033 -3.7	1.160 1.185 2.2	1.108 1.134 2.3	.370 .388 4.9	
6		.953 .958 .5	1.172 1.178 .5	1.057 1.054 -.3	1.129 1.111 -1.6	1.051 1.037 -1.3	1.151 1.127 -2.1	1.165 1.118 -4.0	1.144 1.117 -2.4	1.065 1.038 -2.5	1.162 1.140 -1.9	1.060 1.062 .2	1.165 1.182 1.5	.938 .960 2.3	
7	.321 .323 .6	1.181 1.185 .3	1.201 1.197 -.3	1.090 1.083 -.6	1.277 1.263 -1.1	1.131 1.116 -1.3	1.218 1.202 -1.3	1.078 1.053 -2.3	1.221 1.202 -1.6	1.155 1.127 -2.4	1.299 1.277 -1.7	1.095 1.087 -.7	1.182 1.197 1.3	1.158 1.185 2.3	.315 .323 2.5
8	.352 .360 2.3	1.043 1.047 .4	1.166 1.162 -.3	1.239 1.230 -.7	1.190 1.181 -.8	1.132 1.117 -1.3	1.071 1.052 -1.8	1.259 1.231 -2.2	1.083 1.053 -2.8	1.162 1.118 -3.8	1.203 1.183 -1.7	1.234 1.232 -.2	1.142 1.164 1.9	.996 1.049 5.3	.347 .361 4.0
9	.320 .322 .6	1.181 1.181 .0	1.199 1.194 -.4	1.096 1.085 -1.0	1.299 1.275 -1.8	1.144 1.125 -1.7	1.223 1.201 -1.8	1.076 1.052 -2.2	1.241 1.202 -3.1	1.140 1.116 -2.1	1.278 1.264 -1.1	1.082 1.084 .2	1.178 1.199 1.8	1.148 1.187 3.4	.315 .324 2.9
10		.959 .957 -.2	1.184 1.178 -.5	1.069 1.059 -.9	1.152 1.136 -1.4	1.048 1.036 -1.1	1.133 1.115 -1.6	1.133 1.117 -1.4	1.144 1.125 -1.7	1.048 1.036 -1.1	1.119 1.111 -.7	1.051 1.054 .3	1.160 1.179 1.6	.931 .959 3.0	
11		.388 .386 -.5	1.132 1.130 -.2	1.187 1.181 -.5	1.041 1.029 -1.2	1.120 1.110 -.9	1.271 1.262 -.7	1.188 1.181 -.6	1.283 1.275 -.6	1.145 1.136 -.8	1.035 1.029 -.6	1.174 1.179 .4	1.112 1.129 1.5	.380 .387 1.8	
12			.457 .461 .9	1.203 1.202 -.1	1.182 1.178 -.3	1.055 1.053 -.2	1.080 1.082 .2	1.228 1.230 .2	1.081 1.085 .4	1.053 1.059 .6	1.176 1.181 .4	1.199 1.202 .3	.448 .461 2.9		
13				.462 .461 -.2	1.127 1.128 .1	1.174 1.177 .3	1.188 1.196 .7	1.150 1.161 1.0	1.176 1.193 1.4	1.149 1.178 2.5	1.113 1.129 1.4	.458 .461 .7			
14					.386 .386 .0	.951 .958 .7	1.170 1.185 1.3	1.030 1.047 1.7	1.157 1.181 2.1	.935 .957 2.4	.380 .387 1.8				
15							.316 .323 2.2	.355 .360 1.4	.316 .322 1.9						

Table 4.5.1
Flux Map Summary[1]

Map #	Burnup (EFPD)	Date	Time	Power (%)	D-bank (steps)	Boron (ppm)
325	0.09	01/03/2002	16:30:00	26.9	150	1851
326	0.87	01/05/2002	11:00:00	74.5	176	1550
327	2.15	01/06/2002	23:00:00	98.2	206	1410

Map #	RMS Power[2] (M-P)	max $F_{\Delta h}$	Fraction to Limit, $F_{\Delta h}$ (FLFH)	max F_Q	Fraction to Limit, F_Q (FLFQ)	Axial Offset (%)
325	1.97%	1.672	0.802	2.465	0.511	4.284
326	2.05%	1.597	0.883	2.236	0.692	-0.444
327	1.69%	1.533	0.918	2.145	0.874	2.230

Map #	Thimbles Used	Thimbles Required	Quadrant Power Tilt Ratio			
			NW	NE	SW	SE
325	44	38	1.003	1.004	1.006	0.988
326	43	38	0.999	1.001	1.003	0.996
327	43	38	0.999	1.001	1.003	0.996

1 Flux Map summary data taken from respective INPAX runs [Reference [POWERTRAX]].
2 RMS Power is a figure of merit for how well the core power distribution is predicted.

Table 4.6.1

Reactivity Computer Checkout

Input Parameters to the Reactivity Computer[1]			
Group	β_i	$I * \beta_i$	λ_i
1	0.000213	0.000206	0.0128
2	0.001334	0.001294	0.0317
3	0.001210	0.001174	0.1208
4	0.002610	0.002531	0.3210
5	0.000960	0.000931	1.4025
6	0.000233	0.000226	3.8751
$\Sigma\beta_i$	0.006559	0.006363	--

Prompt Neutron Lifetime $\equiv \ell^* = 17.89$
Importance Factor $\equiv I = 0.97$
Delayed Neutron Fraction $\equiv \Sigma\beta_{eff} = 0.006363$

Positive Insertion Period Check[2]					
Collection #	Δtime (sec)	Period (sec)	Calculated reactivity	Measured reactivity	% Difference
1	53.4	77.1	70.08	69.50	0.83%
2	55.6	80.3	67.99	68.68	-1.00%
3	57.2	82.6	66.55	67.23	-1.02%
4	61.5	88.7	63.03	64.20	-1.83%
5	70.8	102.1	56.46	58.12	-2.87%
average			64.82	65.55	-1.18%

Negative Insertion Period Check[2]					
Collection #	Δtime (sec)	period (sec)	Calculated reactivity	Measured reactivity	% Difference
1	-145.4	209.8	-44.12	-43.94	0.41%
average			-44.12	-43.94	0.41%

1 Reactivity Computer inputs from Reference [SOR], Table 5.1-1b
2 Measured data from Cycle 11 (RFO 10) performance of EST-923, Reference 5.10.

Table 4.6.2
Low Power Physics Test Results Summary

Boron Endpoint (ppm)				
Configuration	Measured[1]	Predicted[1]	Difference	Acceptance
ARO	2090	2105	16	± 50
CBB-in	1924	1934	10	± 50

Control Rod Worths (pcm)						
Bank	Measured[1]	Predicted[1]	Difference \ (pcm	Difference % diff[2]	Acceptance \ (pcm	Acceptance % diff[2]
CBB	1063	1042	21	1.98		± 10%
SBA	952	981	-29	-3.05		± 15%
SBB	866	887	-21	-2.44		± 15%
SBC	292	306	-14	-4.83	± 100	
CBA	509	526	-17	-3.42	± 100	
CBC	916	952	-36	-3.95		± 15%
CBD	879	1001	-122	-13.9		± 15%
Sum of worths	5477	5696	$^{M}/_{P} = 0.962$		$0.9 \le {}^{M}/_{P} \le 1.1$	

HZP Temperature Coefficient (pcm/°F) RCS @ 2070 ppm				
	Measured[1]	Predicted[1]	Difference	Acceptance
ITC	-5.03	-4.94	0.09	Difference ± 2
MTC	-3.49	-3.40	0.09	Measured ≤ +5

Differential Boron Worth (pcm/ppm)				
Configuration	Measured	Predicted	% Difference[2]	Acceptance
CBB going in[3]	-6.40	-6.09	4.8%	± 10%

1 Measured and predicted data obtained from Cycle 11 (RFO 10) performance of EST-923, Reference 5.10

2 % Difference = [(Measured - Predicted) / Measured] * 100.0

3 $DBW_{CBB\ going\ in} = (Worth_{CBB}) / (Boron\ Endpoint_{B\text{-}in} - Boron\ Endpoint_{ARO})$

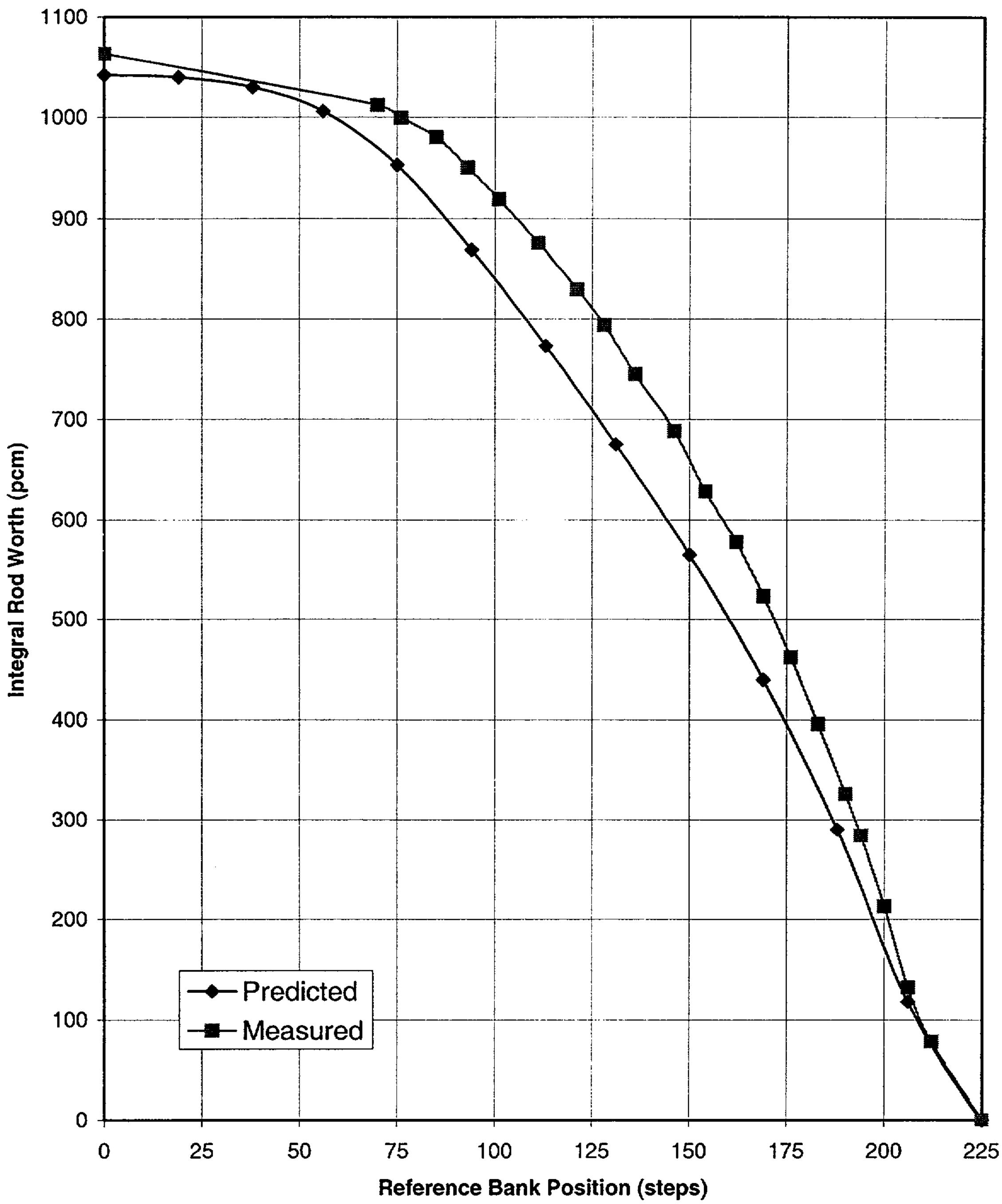
Table 4.7.1

Rod Worth Measurement of the Reference Bank

Initial Step	Final Step	Average Step	Integral Worth (pcm)	Differential Worth (pcm)
70	0	35.0	1063.45	0.73
76	70	73.0	1012.23	2.27
85	76	80.5	999.44	2.11
101	85	93.0	980.61	3.64
111	93	102.0	950.74	3.94
111	101	106.0	919.24	4.31
121	111	116.0	876.22	4.66
128	121	124.5	829.78	4.95
136	128	132.0	794.06	5.84
146	136	141.0	744.98	6.09
154	146	150.0	688.65	6.83
162	154	158.0	628.56	6.82
169	162	165.5	577.71	7.92
176	169	172.5	523.34	8.60
183	176	179.5	462.60	9.63
190	183	186.5	396.24	9.69
194	190	192.0	325.84	11.47
200	194	197.0	283.71	11.09
206	200	203.0	213.19	12.51
212	206	209.0	132.58	9.85
225	212	218.5	78.40	4.17
225	225	225.0	0.00	4.17

Measured data from Cycle 11 (RFO 10) performance of EST-923, Reference 5.10

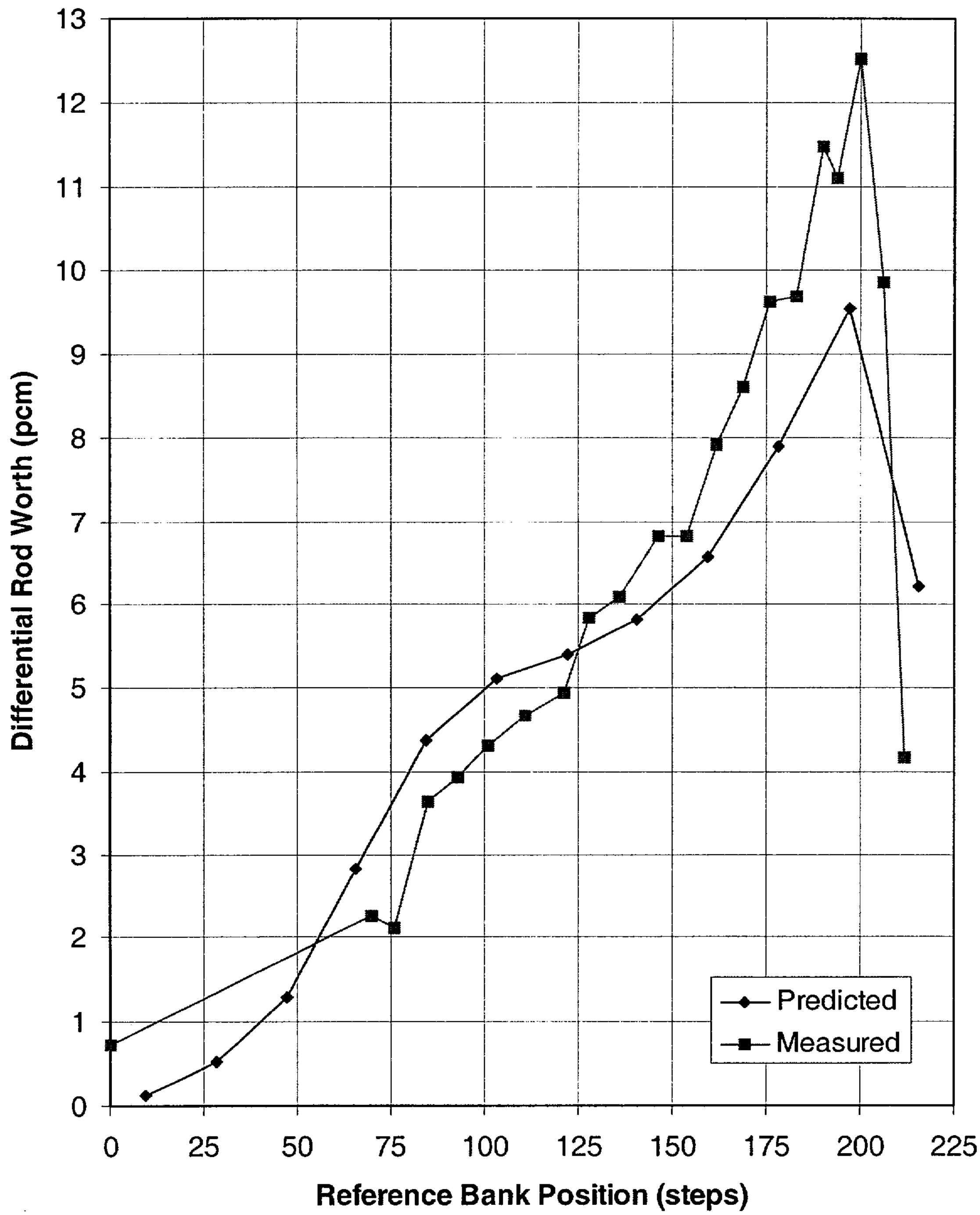
Figure 4.7.1
Integral Worth of the Reference Bank



Measured data from Cycle 11 (RFO 10) performance of EST-923 [Reference 5.10]

Figure 4.7.2

Differential Worth of the Reference Bank



Predicted data taken from SOR [Reference 5.2], Figure 5.3-1
Table contains measured data from Cycle 11 (RFO 10) performance of EST-923 [Reference 5.10].

Table 4.7.3
FSAR Chapter 14 Tests

Test Summary Description	Explanation
Communications System	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Annunciator System	Not impacted by SGR/PUR, system performance is monitored during routine operation, maintenance, and surveillance tests.
Reactor Protection System Engineered Safety Features Actuation Logic	Minimal impact on system from SGR/PUR, system performance discussed in sections 3.1 and 3.2.
Reactor Protection System Engineered Safety Features Actuation Response Time Test	Minimal impact on system from SGR/PUR, system performance discussed in sections 3.1 and 3.2.
Piping Vibration	Minimal impact on system from SGR/PUR, system performance discussed in section 3.5.
Metal Impact Monitoring	Minimal impact on system from SGR/PUR, system performance discussed in section 3.6.
Radiation Monitoring System	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Excore Nuclear Instrumentation (NIS)	Minimal impact on system from SGR/PUR, system performance discussed in section 4.3.
Emergency Diesel	Not impacted by SGR/PUR, system performance is monitored during routine operation and surveillance tests.
Fire Protection System	Not impacted by SGR/PUR, system performance is monitored during routine operation and surveillance tests.
Normal Service Water	Not impacted by SGR/PUR, system performance is monitored during routine operation.

Emergency Service Water	Minimal impact on system from SGR/PUR, system performance discussed in section 3.11.
Compressed and Instrument Air Systems	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Reactor Coolant System Hydrostatic Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.12.
RTD/TC Cross Calibration Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.9.
Pressurizer Relief Tank (PRT) Test	System operation was reviewed and determined to be acceptable by analytical methods.
Safety Injection System Performance Test	System performance discussed in section 3.12.
High-Head Safety Injection System Check Valve Test	Not impacted by SGR/PUR, system performance is monitored during routine startup and surveillance tests.
Safety Injection (SI) Accumulator Test	Not impacted by SGR/PUR, system performance is monitored during routine operation startup and surveillance tests.
Residual Heat Removal System Cold Test	Not impacted by SGR/PUR, system performance is monitored during routine startup and surveillance tests.
Residual Heat Removal System Hot Test	Not impacted by SGR/PUR, system performance is monitored during routine startup and surveillance tests. System operation was reviewed and determined to be acceptable by analytical methods.
Containment Spray System Test	Not impacted by SGR/PUR, system performance is monitored during routine surveillance tests. System operation was reviewed and determined to be acceptable by analytical methods.
Chemical and Volume Control Cold Test	Not impacted by SGR/PUR, system performance is monitored during routine operation and surveillance tests.

Chemical and Volume Control Hot Test	Not impacted by SGR/PUR, system performance is monitored during routine operation and surveillance tests. System operation was reviewed and determined to be acceptable by analytical methods.
Auxiliary Feedwater System Test	Minimal impact on system from SGR/PUR, system performance discussed in section 3.3.
Fuel Handling Equipment System Test	Not impacted by SGR/PUR, system performance is monitored during fuel transfers and surveillance tests.
Fuel Pool Cooling and Cleanup System Test	Not impacted by SGR/PUR, system performance is monitored during normal operation.
Component Cooling Water	Impact on system from SGR/PUR, system performance discussed in section 3.11.
Gaseous Waste Processing System Test	Not impacted by SGR/PUR, system performance is monitored during routine operation and surveillance tests. System operation was reviewed and determined to be acceptable by analytical methods.
Solid Waste Processing Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Liquid Waste Processing System Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Containment Isolation Test	Not impacted by SGR/PUR, system performance is monitored during routine startup and surveillance tests.
Containment Integrated Leak Rate Test and Structural Integrity Test	Minimal impact on system from SGR/PUR, system performance discussed in section 3.4.
Reactor Coolant System Hot Functional Test	Not impacted by SGR/PUR, system performance is monitored during routine operation. System operation was reviewed and determined to be acceptable by analytical methods.
Piping Thermal Expansion and Dynamic Effects Test	Impact on system from SGR/PUR, system performance discussed in section 3.5
Pressurizer Pressure and Level Control Test	System operation was reviewed and determined to be acceptable by analytical methods.

Main Steam System Test	Impact on system from SGR/PUR, system performance discussed in section 3.13.
Feedwater System Test	Impact on system from SGR/PUR, system performance discussed in section 3.13.
Condensate System Test	Impact on system from SGR/PUR, system performance discussed in section 3.13.
Turbine Generator Test	Impact on system from SGR/PUR, system performance discussed in section 3.14.
Circulating Water System Test	System operation was reviewed and determined to be acceptable by performance testing, discussed in section 4.14.
Condenser Vacuum and Condensate Makeup System	System operation was reviewed and determined to be acceptable by performance testing, discussed in section 4.14.
Waste Processing Computer Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Containment Ventilation and Cooling, Primary Shield and Reactor Supports Cooling System Test	Not impacted by SGR/PUR, system performance is monitored during routine startup and surveillance tests.
Plant HVAC Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Engineered Safety Features Integrated Test	System operation was reviewed and determined to be acceptable by testing described in sections 3.1 and 3.2.
Process Computer Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Boron Recycle Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Refueling Water Storage Tank Test	Not impacted by SGR/PUR, system performance is monitored during routine operation. System operation was reviewed and determined to be acceptable by analytical methods.
Primary Makeup Water System Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.

Rod Control System Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Passive Safety Injection System Check Valve Test	Not impacted by SGR/PUR, system performance is monitored during routine startup and surveillance tests. System operation was reviewed and determined to be acceptable by analytical methods.
Containment Recirculation Sump Test	System operation was reviewed and determined to be acceptable by analytical methods.
Containment Vacuum Relief Test	System operation was reviewed and determined to be acceptable by analytical methods.
Combustible Gas Control System In Containment Test	System operation was reviewed and determined to be acceptable by analytical methods.
Gross Failed Fuel Detection System Test	System operation was reviewed and determined to be acceptable by analytical methods.
Essential Services Chilled Water System Test	System operation was reviewed and determined to be acceptable by analytical methods.
Stud Tensioner Hoist Load Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Polar Crane Test Summary	System was revised extensively for SGR and restored to pre-existing configuration prior to startup. System operation was reviewed and determined to be acceptable by analytical methods.
Feedwater Heater Drain, Level and Bypass Control Systems Test	Not impacted by SGR/PUR, system performance is monitored during routine operation. System operation was reviewed and determined to be acceptable by analytical methods.
Seismic Instrumentation Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.

Extraction Steam System Test	Not impacted by SGR/PUR, system performance is monitored during routine operation. System operation was reviewed and determined to be acceptable by analytical methods.
Primary Sampling System Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Secondary Sampling System Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Loss of Instrument Air Test	System operation was reviewed and determined to be acceptable by analytical methods.
Containment Building Hot Penetration Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Simulated Loss of On-Site Power Test	System operation was reviewed and determined to be acceptable by analytical methods.
AC Distribution System Optimum Operating Voltage Test	System operation was reviewed and determined to be acceptable by analytical methods.
Auxiliary Feedwater Turbine Pump Two-Hour Run Test	System operation was reviewed and determined to be acceptable by analytical methods.
Power Ascension Test	The power ascension program described in PLP-632T controlled post SGR/PUR testing.
Moveable Incore Detector Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.4.
Rod Control and Position Indication System Test	Not impacted by SGR/PUR, system performance is monitored during routine startup and surveillance tests.
Rod Drive Mechanism Timing Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.1.
Rod Drop Time Measurement Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.1.

Reactor Coolant System Flow Measurement Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.2.
Reactor Coolant System Flow Coastdown Test	System operation was reviewed and determined to be acceptable by analytical methods.
Calibration of Nuclear Instrumentation Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.3.
Rod Control System Test	System operation was reviewed and determined to be acceptable by analytical methods.
Flux Distribution Measurement Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.4.
Core Performance Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.5.
Power Coefficient Measurement Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.6.
Control Rod Reactivity Worth Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.7.
Boron Reactivity Worth Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.8.
Automatic Rod Control Test	System operation was reviewed and determined to be acceptable by analytical methods.
Steam Generator Moisture Carryover Test	Impact on system from SGR/PUR, system performance discussed in section 4.10.
Load Swing Test	Impact on system from SGR/PUR, system performance discussed in section 4.11.
Large Load Reduction From 75 Percent Power Test	System operation was reviewed and determined to be acceptable by analytical methods.
Turbine Trip From 100 Percent Power Test	System operation was reviewed and determined to be acceptable by analytical methods.
Remote Shutdown Test	System operation was reviewed and determined to be acceptable by analytical methods.

Loss of Offsite Power Test	System operation was reviewed and determined to be acceptable by analytical methods.
Pressurizer Heaters and Spray Valves Capability Test	System operation was reviewed and determined to be acceptable by analytical methods.
Gross Failed Fuel Detection System Test	System operation was reviewed and determined to be acceptable by analytical methods.
Pressurizer Continuous Spray Flow Verification Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.
Reactor Coolant System Leakrate Test	Minimal impact on system from SGR/PUR, system performance discussed in section 4.12.
Main Steam and Feedwater Systems Test	Impact on system from SGR/PUR, system performance discussed in section 4.13.
Shield Survey Test	Not impacted by SGR/PUR, system performance is monitored during routine operation. System operation was reviewed and determined to be acceptable by analytical methods.
Loss of Feedwater Heater(s) Test	System operation was reviewed and determined to be acceptable by analytical methods.
Main Steam Isolation Valve Test	Not impacted by SGR/PUR, system performance is monitored during routine startup and surveillance tests.
Steam Generator Test for Condensation Water Hammer	Water Hammer was a concern for D4 SGs; the feedring design of the Δ75 SGs eliminates this concern. System operation was reviewed and determined to be acceptable by analytical methods.
Steam Turbine-Driven and Motor-Driven Auxiliary Feedwater Pumps Endurance Test	System operation was reviewed and determined to be acceptable by analytical methods.
Resistance Temperature Detector (RTD) Bypass Flow Verification Test	Test is no longer applicable with current RTD configuration.
Secondary Sampling System Test	Not impacted by SGR/PUR, system performance is monitored during routine operation.