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Nuclear Power Plant
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NewYorkPower Authority

Robert J. Barrett
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Cycle 11 Startup Report

References: 1. Indian Point Nuclear Generating Unit No. 3 Technical Specification 6.9.1.1

Dear Sir:

This letter transmits an informational copy of the Indian Point 3 Cycle 11 startup physics test report. While no new fuel designs were used, the Vantage + fuel design, currently being used in the reload feed region of the core, did incorporate some new features compared to previous Vantage + fuel.

This letter contains no new commitments. If you have any questions, please contact Mr. K. Peters.

Very truly yours,

Robert J. Barrett
Site Executive Officer

Attachment: as stated
cc: See next page

IE26

cc: Regional Administrator
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Attachment I to IPN-00-004

Cycle 11 Startup Physics Test Report

(32 pages)

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

Executive Summary / Abstract

During September 10 – October 20, 1999, Indian Point Unit 3 was shutdown for a scheduled refueling outage. At the conclusion of the outage, a series of pre-operational and power ascension tests were performed to verify that reactor core kinetics parameters and protection circuits were consistent with the plant safety analysis. A chronological summary of the test and results are presented below and in the following table:

Test	Results
I. Zero Power	
Core Loading	Satisfactory
RTD and Core Exit Thermocouple Measurements	Satisfactory
Initial Criticality	Satisfactory
Control Rod Worth Measurements	Satisfactory
Critical Boron Endpoint Measurements	Satisfactory
Isothermal Temperature Coefficient Measurement	Satisfactory
II. At Power	
Reactor Thermal Power Calculations / Nuclear Instrumentation Calibrations	Satisfactory
Core Power Distribution Measurements	Satisfactory
Reactor Coolant System Flow Calculation	Satisfactory
Incore Excore Calibration	Satisfactory
Calibration of Overtemperature / Overpower Protection	Satisfactory
Calibration of "High Taverage" Alarm	Satisfactory
Full Power Critical Boron Measurements	Satisfactory

The unit subsequently achieved full power on October 26, 1999.

This report contains detailed descriptions of the cycle 11 core and each of the tests listed above.

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1.0 Introduction

1.1 Plant Description

The Indian Point Unit 3 Nuclear Plant is a four hour loop closed cycle pressurized light water moderated and cooled nuclear reactor operated by the New York Power Authority. The reactor core is designed to produce 3025 megawatts thermal power resulting in a net electrical generating capacity of 965 megawatts of electrical energy.

Westinghouse Electric Corporation designed the Nuclear Steam Supply System.

The plant is located on the east side of the Hudson River, approximately 30 miles north of New York City.

1.2 Test Objectives

This report documents the results of physics tests performed as part of the cycle 11 startup testing program:

The objectives of the physics tests were:

(1) To verify that the operating characteristics of the core are consistent with design predictions
(2) To demonstrate that measured core parameters are consistent with values used in the Safety Analysis
(3) To demonstrate that the core can be operated at licensed thermal power safely and within the limits of the Technical Specifications
(4) To provide data for instrumentation calibration.

1.3 Relevant Design Information

Table 1.1 presents selected design parameters of the Indian Point 3 Nuclear Plant. Figure 1.1 shows the core layout with control rods, mechanical burnable absorbers, sources, and fuel assembly numbers. The Cycle 11 core contains two regions of Westinghouse VANTAGE 5 (V5) fuel (Regions 8-1 and 11-1) and two regions of Vantage + (V+) fuel (Regions 12-1, 12-2, 13-1 and 13-2). The Cycle 11 core has the following unique features described below:

A. The 72 feed assemblies in the Cycle 11 core are of the Vantage + design and incorporates design features of Performance + (P+). The P+ changes to the V+ design include the incorporation of a protective bottom grid located at the bottom end plug and longer fuel rod bottom end plugs to improve debris failure resistance. In addition, a number of dimensional changes have been made to accommodate extended burnup. The fuel assembly bottom nozzle was modified to accommodate use of the protective bottom grid.

B. A new antimony-beryllium secondary source was placed in the core for activation at core location M-03. This new source is of the Westinghouse double encapsulated design and replaces an older single encapsulated type source.

C. Three different types of burnable poisons are being used in the Cycle 11 core:

1) A 20-pin unclad hafnium flux suppression insert is located in the assemblies in the "corners" of the core (8 total) as a means of further reducing neutron fluence on the reactor vessel. The hafnium flux suppressors are a new feature that replaces the Pyrex poison inserts used in previous cores. The flux suppressors offer improved reduction in neutron leakage and are reusable.

2) All seventy-two feed assemblies contain integral fuel burnable absorber (IFBA) rods. These assemblies contain a specific pattern of either 32, 48, or 80 IFBA rods.

3) Wet Annular Burnable Absorber (WABA) inserts are used to provide additional reactivity hold-down in 48 of the 88 feed assemblies. The WABA assemblies contain a specific pattern of either 12, 16, or 20 rods. Figure 1.2 shows the location of all burnable absorbers in the cycle 11 core.

1.4 Sequence of Startup Events

Following core loading, October 6 - 8, 1999, a series of pre-operational test were performed both in the cold shutdown and hot shutdown conditions. Initial cycle 11 criticality was achieved on October 20, 1999 followed by a program of low power physics tests. The unit was synchronized to the grid on October 21, 1999. Full power was achieved on October 26, 1999.

1.5 Summary of Measured and Predicted Core Parameters

Presented in Table 1.2 is a summary of selected results of physics tests and at-power distribution measurements.

Table 1.1
Indian Point Unit 3 Cycle 11
Core Design Parameters

Number of Fuel Assemblies	193
Region 8-1	1
Region 11-1	32
Region 12-1	56
Region 12-2	32
Region 13-1	56
Region 13-2	16
Lattice Configuration	15x15
Number of Fuel Rods Per Assembly	204
Fuel Loading, MTU	87.57
Number of Assemblies Containing RCC Full Length	53
Number of Absorber Rods Per RCC Assembly	20
Number of Control Rod Assembly Guide Thimbles Per Assembly	20
Number of Instrumentation Thimbles Per Assembly	1
Number of Midspan Grids	7
Number of IFM Grids (Vantage + Fuel only)	3
Number of Protective Bottom Grids (Performance + Fuel only)	1
Heat Output, MWth	3025
Percent Heat Generated in Fuel	97.4
Hot Zero Power Coolant Temperature, °F	547.0
Operating Pressure, psia	2250
Maximum Hot Channel Factors (Design)	
Heat Flux Fq(T)	2.42
Nuclear Enthalpy Rise, FΔH (OFA, Vantage 5)	1.59
Nuclear Enthalpy Rise, FΔH (Vantage +, Performance +)	1.635
Average Linear Power, kw/ft Fuel	6.24
Specific Power, kw/kg Uranium	34.54
Initial Enrichments, w/o Uranium 235	
Region 8-1	3.20
Region 11-1	4.00
Region 12-1	4.40
Region 12-2	4.60
Region 13-1	4.30
Region 13-2	4.60

Table 1.2 (Page 1 of 2)
Indian Point Unit 3 Cycle 11
Summary of Zero and At Power Physics Testing Results

I. Critical Boron Concentrations (PPM)

Design Review Criteria (DRC) = ±50 PPM and ±500 PCM
Acceptance Criteria (AC) = within 1000 PCM

	Predicted (P)	Measured (M)	(M-P)	Pass / Fail DRC	Pass / Fail AC
ARO	1689	1700	11	P	P

II Control Bank Worths (PCM)

Design Review Criteria = Individual Bank Worths within 15% or 100 PCM whichever is greater and sum of measured integral Bank Worths is within 8% of sum of predicted integral Bank Worths.
Acceptance Criteria = Total Worth is at least 90% of predicted

Bank	Predicted (P)	Measured (M)	PCT. Diff*	Pass / Fail DRC	Pass / Fail AC
Control A	774.7	790.6	+2.1	P	-
Control B	482.2	506.2	+5.0	P	-
Control C	558.5	585.7	+4.9	P	-
Control D	720.6	723.0	+0.3	P	-
Shutdown A	1027.1	1091.6	+6.3	P	-
Shutdown B	812.9	831.7	+2.3	P	-
Shutdown C	251.2	246.6	+1.8	P	-
Shutdown D	473.7	514.2	+8.5	P	-
Total	5100.9	5289.6	+3.7	P	P

III. Isothermal Temperature Coefficient (PCM / F)

Design Review Criteria = ± 2 PCM / F

	Predicted (P)	Measured (M)	(M-P)	Pass / Fail DRC
ARO	-3.15	-2.795	+ 0.355	P

IV. Inferred Moderator Temperature Coefficient PCM / F) **

Acceptance Criteria = MTC is negative or withdrawal limits imposed

	Predicted (P)	Measured (M)	(M-P)	Pass / Fail AC
ARO	-1.54	-1.11	+ 0.43	P

ARO: All Rods Out

* Percent Difference = 100 (M-P) / P

** Inferred MTC is obtained by subtracting Doppler Coefficient (-1.61 PCM / F) from the Isothermal Temperature Coefficient.

Table 1.2 (Page 2 of 2)
Indian Point Unit 3 Cycle 11
Summary of Zero and At Power Physics Testing Results

I. Power Distribution Measurements

A) Low Power (27.89%)

Tilts	
1.00104	0.96625
1.01271	1.02001

Largest Reaction Rate Integral Deviation – 7.7%

Limiting FQ - 2.014	FQ Limit - 4.620
Highest FDHN (V+) – 1.578	FDHN Limit (V+) - 2.107
Highest FDHN (V5) - 0.775	FDHN Limit (V5) - 2.049

B) Intermediate Power (42.63%)

Tilts	
1.00139	0.97319
1.01215	1.01327

Largest Reaction Rate Integral Deviation – 6.4%

Limiting FQ - 1.913	FQ Limit - 4.568
Highest FDHN(V+) - 1.546	FDHN Limit (V+) - 2.010
Highest FDHN(V5) - 0.797	FDHN Limit(V5) - 1.9549

C) Full Power (99.03%)

Tilts	
1.00305	0.97523
1.01337	1.00835

Largest Reaction Rate Integral Deviation - 5.8%

Limiting FQ - 1.805	FQ Limit - 2.258
Highest FDHN(V+) - 1.489	FDHN Limit(V+) - 1.6398
Highest FDHN(V5) - 0.898	FDHN Limit(V5) - 1.5946

II. Reactor Coolant System Flow Measurement

Measured Flow - 392459.9 GPM   Minimum Required Flow - 375,600

III. Full Power Critical Boron (PPM)

Design Review Criteria (DRC) = within 50 PPM
Acceptance Criteria (AC) = within 1000 PCM (117 PPM)

Burnup				Pass / Fail	
(EFPD)	Predicted (P)	Measured (M)	(M-P)	DRC	AC
7.1	1124.06	1123.21	-0.85*	P	P
50.3	1089.10	1123.19	34.09**	P	P

Note: Design boron letdown curve reduced by 29 ppm per TS 3.10.10 based on the average of reactivity measurements. The design maximum B-10 depletion boron curve was adjusted by only 10 ppm. Cycle 11 core design assumes 20.4 atom% B-10 in RCS, which matches actual B-10 atom % of 20.4. These results differ from previous cycles at IP3 in that the average measured boron exceeds the predicted value. Much of this is attributable to depletion of the isotope B-10 in the RCS boron because the plant has operated essentially at full power since the startup from refueling and very little makeup water is required to be added to the RCS.

* Non-equilibrium Samarium, no B-10 depletion.
** Equilibrium Samarium Conditions, near-maximum B-10 depletion.

Figure 1.1
Indian Point Unit 3 Cycle 11
Core Layout

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15					X23 PD44Z	X32 PD43Z	X44 PD31Z	Y12 PD50Z	X04 PD61Z	X13 PD88Z	X53 PD10Z				
14			X26 HF008	Y44 PD13Z	AA07 PD92Z	AA23 R75	AA63 12W0014	Y21 R73	AA61 12W0022	AA24 R79	AA05 PD80Z	Y49 PD122Z	X15 HF004		
13		X48 HF006	Y85 R105	AA15 SS16	Y25 R82	AA32 16W0022	Y24 R108	AA50 12W0015	Y45 R116	AA39 16W0031	Y32 R95	AA13 PD21Z	Y62 R88	X10 HF002	
12		Y09 PD15Z	AA16 PD90Z	Y54 R123	AA71 20W0036	Y88 PD86Z	AA41 12W0023	Y86 R80	AA33 12W0011	Y72 PD29Z	AA69 20W0037	Y23 R125	AA10 PD23Z	Y42 PD79Z	
11	X14 PD83Z	AA08 PD85Z	Y46 R74	AA72 20W0035	Y84 PD54Z	Y03 PD05Z	Y60 PD34Z	Y30 PD46Z	Y61 R109	Y43 PD95Z	Y64 PD42Z	AA66 20W0039	Y19 R97	AA02 PD47Z	X30 PD19Z
10	X38 PD52Z	AA22 R61	AA54 16W0033	Y69 PD71Z	Y31 PD76Z	Y70 R90	AA38 16W0021	Y40 R128	AA17 16W0028	Y65 R83	Y27 PD73Z	Y67 PD66Z	AA34 16W0026	AA19 R111	X49 PD18Z
9	X03 PD51Z	AA64 12W0016	Y52 R120	AA55 12W0029	Y74 R84	AA46 16W0030	Y48 PD35Z	AA48 12W0013	Y05 PD94Z	AA40 16W0032	Y76 PD28Z	AA18 12W0026	Y55 R65	AA58 12W0028	X39 PD58Z
8	Y08 PD22Z	Y15 R101	AA21 12W0027	Y57 R66	Y07 PD30Z	Y39 R126	AA47 12W0024	U02 R98	AA43 12W0019	Y41 R121	Y02 PD84Z	Y63 R115	AA44 12W0012	Y35 R92	Y53 PD119Z
7	X45 PD74Z	AA59 12W0017	Y37 R112	AA20 12W0018	Y75 PD03Z	AA35 16W0035	Y56 PD36Z	AA42 12W0032	Y18 PD49Z	AA36 16W0034	Y73 R99	AA53 12W0030	Y26 R127	AA60 12W0031	X02 PD81Z
6	X55 PD41Z	AA28 R107	AA27 16W0024	Y79 PD64Z	Y01 PD40Z	Y83 R85	AA25 16W0025	Y28 R118	AA29 16W0023	Y66 R100	Y51 PD62Z	Y81 PD48Z	AA45 16W0036	AA31 R81	X17 PD63Z
5	X35 PD55Z	AA03 PD8Z	Y20 R72	AA67 20W0040	Y78 PD91Z	Y16 PD82Z	Y82 R104	Y36 PD70Z	Y80 PD07Z	Y04 PD12Z	Y59 PD37Z	AA68 20W0033	Y06 R69	AA04 PD96Z	X34 PD16Z
4		Y22 PD93Z	AA11 20Z	Y17 R119	AA65 20W0038	Y77 PD17Z	AA52 12W0025	Y58 R71	AA26 12W0020	Y71 PD38Z	AA70 20W0034	Y38 R124	AA12 09Z	Y13 PD11Z	
3		X25 HF003	Y87 R830	AA09 53Z	Y33 R106	AA49 16W0029	Y47 R117	AA56 12W0021	Y10 R63	AA37 16W0027	Y14 R78	AA14 SS17	Y68 R70	X40 HF005	
2			X51 HF007	Y34 PD78Z	AA01 72Z	AA51 R62	AA57 12W0010	Y11 R122	AA62 12W0009	AA30 R77	AA06 PD60Z	Y50 PD27Z	X36 HF001		
1					X52 PD56Z	X46 PD75Z	X01 PD59Z	Y29 PD06Z	X18 PD39Z	X06 PD77Z	X42 PD04Z				

Figure 1.2
Indian Point Unit 3 Cycle 11
Burnable Absorber Configuration

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R															
P			16FS		32I	80I	12W 48I		12W 48I	80I	32I		16FS		
N		16FS		48I		16W 80I		12W 80I		16W 80I		48I		16FS	
M			48I		20W 80I		12W 80I		12W 80I		20W 80I		6SSA 48I		
L		32I		20W 80I								20W 80I		32I	
K		80I	16W 80I				16W 80I		16W 80I				16W 80I	80I	
J		12W 48I		12W 80I		16W 80I		12W 80I		16W 80I		12W 80I		12W 48I	
H			12W 80I				12W 80I		12W 80I				12W 80I		
G		12W 48I		12W 80I		16W 80I		12W 80I		16W 80I		12W 80I		12W 48I	
F		80I	16W 80I				16W 80I		16W 80I				16W 80I	80I	
E		32I		20W 80I								20W 80I		32I	
D			4SSA 48I		20W 80I		12W 80I		12W 80I		20W 80I		48I		
C		16FS		48I		16W 80I		12W 80I		16W 80I		48I		16FS	
B			16FS		32I	80I	12W 48I		12W 48I	80I	32I		16FS		
A															

0°

270°

TYPE	TOTAL
##W...(NUMBER OF WABA RODLETS)...............	704
##I...(TOTAL NUMBER OF FRESH IFBA RODS)......	4864
##FS..(NUMBER OF FLUX SUPPRESSION RODLETS)...	128
#SSA..(NUMBER OF SECONDARY SOURCE RODLETS)...	10

2.0 Measurement Techniques

2.1 General

The methods for physics test data acquisition can be grouped into four distinct areas: (1) reactivity measurements, (2) measurements of core power distribution, (3) collection of instrumentation data, and (4) thermal power and flow measurements. The purpose of this section is to describe the methods used in each of these areas.

2.2 Reactivity Measurements

Measurements of core reactivity were performed both in subcritical and critical core conditions. In the subcritical mode, measurements were made during initial core loading and the approach to criticality. In the critical mode, measurements were made to determine core kinetics parameters.

2.2.1 Subcritical Measurements

During core loading, the core reactivity was monitored using the response of the two plant source range channels. One of the source range detectors was replaced at the start of the outage due to failure. The detector most likely failed as a result of being weakened by an inadvertent energization at full power earlier in the previous cycle. The new detector functioned satisfactorily. Monitoring was accomplished by determining the normalized inverse count rate ratio (ICRR) for each channel as the core was loaded (see Figure 3.1). During the approach to criticality, ICRR plots using data from the two plant source range channels were used to predict expected criticality. ICRR data were plotted as a function of rod position during rod withdrawal (Figure 3.2), and as a function of measured boron concentration during dilution (Figure 3.3).

2.2.2 Critical Measurements

Small core reactivity changes were determined with the aid of a digital reactivity computer that provided an on-line solution to the point kinetics equations. Reactivity records were maintained on a continuous basis during each test via a strip chart recorder that logged the output from the reactivity computer.

One Nuclear Instrumentation System (NIS) power range channel provided the input signal to the reactivity computer. During zero power measurements, channel N44 was taken out of plant service and used as input to the reactivity computer.

Integral worth of individual rod control cluster assemblies (RCCA) banks were obtained from the reactivity computer's response to the inward movement of the four control banks and four shutdown banks. This measurement method is called Dynamic Rod Worth Measurement (DRWM). The control bank overlap feature was defeated for this test. During the measurement, the reactor was critical by 55 to 70 pcm. The individual banks were inserted and then withdrawn by use of the reactivity computer. The total worth of the control and shutdown banks was measured by the DRWM method.

Isothermal temperature coefficient data was obtained by measuring the reactivity computer response to small temperature changes, a few degrees below design no load temperature. Just critical boron concentration data was obtained from plant chemistry boron analysis of reactor coolant system samples (RCS) under equilibrium conditions. The boron concentration endpoint correction to the measured concentration utilized reactivity computer measurements of the reactivity difference between actual and design core configurations. This data was collected during DRWM testing.

2.3 Power Distributions

The Moveable Detector (M/D) Flux Mapping System was used to collect power distribution data. The power distribution measurements were performed at three different power plateaus in order to verify:

1) Proper core loading
2) Core design calculations
3) Margin in hot channel factors

The M/D system was also used to provide data for excore detector calibration. Data from the M/D system was input to the BEACON computer code to generate detailed three dimensional core power profiles. The BEACON code combines measured flux distributions with design calculated power flux distribution to yield specific fuel rod powers, local burnup, core power tilts, core average axial offset, etc.

2.4 Instrumentation Calibration Data Collection

At each stable power level (statepoint) during the power escalation program (approximately every 10% at and above 50%) measurements were made of RCS loop temperatures (T_{avg} and ΔT), Steam Generator pressure and NIS power range detector current. Temperature and pressure data were obtained from the meters on the control board, from the plant computer, and the individual Control Room instrumentation racks. Core exit thermocouple and RCS RTD data were obtained during isothermal measurements prior to criticality, and a RTD cross-calibration check was performed. Correlation between incore axial power distribution and excore power range detector response were made through simultaneous measurements of core power level, excore detector currents and core power distributions (flux maps).

2.5 Thermal Power and Flow Measurements

Core thermal power was determined by performing a heat balance across each of the steam generators. This measurement required the accurate determination of steam generator pressure, feedwater inlet temperature, and feedwater flow. For each steam generator, steam pressure was taken from the plant computer, feedwater temperature was taken from the resistance temperature detectors (RTD) located in the feedwater headers and feedwater flow was determined from the Leading Edge Flow Meters.

With the plant operating at approximately 94 percent power, a reactor coolant system flow determination was performed. The purpose of this calculation is to verify that RCS flow is at least as great as the flow assumed in the Final Safety Analysis Report and Technical Specification basis. This procedure is performed within 24 hours after power escalation above 90 percent at the beginning of each cycle. The procedure utilizes an energy balance with a secondary thermal power calculation and precision T_{hot} and T_{cold} measurements.

3.0 Test Results

3.1 Core Loading

Core loading was accomplished by adding fuel assemblies to the vessel following a prescribed sequence. The ICRR data obtained from NIS source range channels is presented in Figure 3.1. There were no unexpected changes in core reactivity during the loading of the fuel assemblies.

3.2 Initial Criticality

The approach to criticality began on October 20, 1999 at 0102 hours with the incremental withdrawal of shutdown and control banks. Primary System boron concentration during rod withdrawal was approximately 1782 ppm. Inverse count rate ratio data from two source range channels during rod withdrawal are shown in Figure 3.2. Criticality was achieved with the addition of reactor makeup water. Inverse count rate ratios during boron dilution are shown in Figure 3.3. Throughout the critical approach, count rates from the two source range channels were consistent for monitoring of core reactivity.

3.3 Low Power Physics Tests

3.3.1 Preliminary Measurements

Immediately following criticality, the upper limit of flux level for zero power testing was established as about one decade below nuclear heating. Nuclear heating was determined to begin at 1.8×10^{-6} amps power range. Next a check of the reactivity computer performance was made by measuring four values of reactivity and comparing the value with that inferred from the resultant reactor period from parameters given in the core design report. The results of this test, given in Table 3.1, indicate proper operation of the reactivity computer.

Table 3.1
Reactivity Computer Checkout Results

Period (sec)	Predicted Reactivity (pcm)	Measured Reactivity (pcm)	Difference
135.9	42.4	41.8	-0.6 pcm

3.3.2 Boron Endpoints

The just critical boron concentration was determined from data collected by the digital reactivity computer during DRWM testing. The test results are summarized in Table 1.2 along with design predictions. Measured results were within 11 ppm of design.

3.3.3 Temperature Coefficient

Isothermal temperature coefficient measurements were performed at two core conditions, as summarized in Table 1.2. The inferred all-rods-out, moderator-only temperature coefficient (MTC) was negative. However, since MTC increases with burnup, rod withdrawal limits were developed to insure a negative MTC as required by Technical Specifications. Since Technical Specifications require MTC to be negative or zero when the reactor is critical, control rods and

RCS boron concentration is controlled to maintain a 0 or negative MTC. In order to do this, control rod withdrawal limits (presented as a set of curves) at different RCS temperatures and power levels are developed so that the operators can maintain a negative MTC. The rod withdrawal limits are determined starting at the fully withdrawn position and ending at the control rod insertion limit. The calculation method determines the boron concentration at a particular control rod configuration where MTC is equal to 0. A 10 ppm conservatism factor is included. This effect will be significant for approximately the first 3 months of operation until boron concentration starts decreasing.

3.3.4 RCC Bank Worths

RCC Bank worth measurements were performed on all control banks and shutdown banks in non-overlap mode. The measurements were done using the dynamic rod worth measurement (DRWM) method. Measured and predicted integral worths of these eight banks are summarized in Table 1.2.

3.4 At Power Tests

3.4.1 RCS Flow Determination

On October 26, 1999 RCS flow was measured to be 392,459.9 gallons per minute. The flow assumed in the FSAR at the beginning of DNBR analysis is 375,600 gallons per minute. The actual measurement normalized to full (100% indicated) flow indicated that a margin of approximately 4.2 percent exists in RCS Flow.

3.4.2 Reactor Thermal Power Measurements

In order to provide protection against possible non-conservatism in initial nuclear instrumentation readings, the high flux reactor trip setpoint was reduced from the normal 108% value to approximately 80% prior to initial criticality. During startup, initial reactor thermal power measurements were made between 2% and 5% power, based on loop delta-T power correlation, and the nuclear instrumentation was adjusted accordingly to provide correct power indication and sufficient margin to the P-10 setpoint and intermediate range rod stop and trip setpoints. Various NIS bistables were closely monitored to ensure proper setpoint actuation during power ascension. The initial heat balance was performed at approximately 25-30% power. The calculation was repeated at approximately 10% increments from 50% to 100% power. The high flux trip setpoint was raised back to 108% just before reaching 50% power.

3.4.3 Full Power Critical Boron Measurements

After achieving full power, core reactivity balance measurements were performed approximately every 7 effective full power days (EFPD). The reactivity balance calculation provides an assessment of the difference between predicted and measured full power boron concentrations, taking into account xenon, samarium, Tavg, rod position, and reactor power effects. The initial comparison, which is made prior to reaching equilibrium samarium, showed that the measured boron concentration was 0.85 ppm below the predicted value. As samarium reached equilibrium, the difference leveled off to approximately 34 ppm above the predicted value. These results differ from previous cycles at IP3 in that the average measured boron exceeds the predicted value. Much of this is attributable to depletion of the isotope B-10 in the RCS boron because the plant has operated essentially at full power since the startup from refueling and very little makeup water is required to be added to the RCS. Table 3.2, shows the reactivity balance results through the first fifty EFPD of operation. As required by T.S. 3.10.10, a 29 ppm adjustment factor was applied to the design boron curve. The design maximum B-10 depletion boron curve was adjusted by only 10 ppm. These curves are presented as Figure 3.4.

Table 3.2
Reactivity Balance Summary

EFPD	Measured (PPM)	Predicted (PPM)	Delta (PPM)
7.1	1123.21	1124.06	-0.85
10.0	1124.49	1117.84	6.65
17.0	1110.48	1105.74	4.74
22.3	1110.57	1100.94	9.63
27.4	1115.91	1096.43	19.48
35.4	1119.20	1093.22	25.98
38.3	1117.49	1092.41	25.08
42.5	1120.61	1091.27	29.34
50.3	1123.19	1089.10	34.09

3.5 Movable Detector Flux Maps

3.5.1 Low Power

The initial flux map of cycle 11 was taken at approximately 28 percent power. The purpose of this map was to verify proper core loading. The greatest deviation between predicted and measured average reaction rate integrals was 7.7 percent at core location P-04. Based on a review of this map the core was determined to be properly loaded. A summary of parameters is presented below:

Date	October 22, 1999
Map Number	11FCFM01
Power	27.89%
Rod Position	D/167 steps
Greatest Tilt	2.00%
Greatest FDH (V+)	1.578
FDH Limit (V+)	2.107
Greatest FDH (V5)	0.775
FDH Limit (V5)	2.049
Most Limiting FQ	2.014
FQ Limit	4.620
Core Average Axial Offset	3.261

As can be seen from the above data, it was determined that a small quadrant tilt had been built into the core. This measured tilt was not predicted as the core is designed to achieve a zero tilt condition. According to Westinghouse, incore tilts of up to 4.0% that are built into the core are not a concern in terms of effect on core design parameters and hot channel factors. The tilt condition is monitored during periodic flux mapping over the course of the operating cycle and is expected to gradually balance out as the core is operated.

3.5.2 Intermediate Power

The second flux map of Cycle 11 was taken at approximately 43 percent power with equilibrium Xenon established. The purpose of this map was to verify that core power distribution and peaking factor predictions were consistent with measured values and calibrate the excore power range NIS to reflect core conditions and match the built-in core quadrant tilt. The excore power range NIS calibration data was developed using the Westinghouse supplied SPEDCAL function on the BEACON code. This method eliminates the need to perform quarter core flux maps at different axial offsets to develop calibration data and eliminates the core perturbations associated with this activity. The greatest deviation between predicted and measured average reaction rate integrals was –6.4 percent at core location K-02. Based on a review of this map it was concluded that core power distribution and peaking factor predictions were acceptable. A summary of parameters is presented below:

Date	October 23, 1999
Map Number	11FCFM02
Power	42.63%
Rod Position	D/179 steps
Greatest Tilt	1.33%
Greatest FDH (V+)	1.546
FDH Limit (V+)	2.010
Greatest FDH (V5)	0.797
FDH Limit (V5)	1.9549
Most Limiting FQ	1.913
FQ Limit	4.568
Core Average Axial Offset	2.267%

3.5.3 Full Power

The initial full power flux map of cycle 10 was taken on October 28, 1999. The purpose of this map was to verify that measured full power hot channel factors (FQ, FDH) were within Technical Specification limits, and to develop calibration values for the excore power range NIS to reflect full power core conditions to better match the built-in core quadrant tilt. Based on a review of this map all power distribution parameters were within applicable limits. A summary of parameters is presented below:

Date	October 28, 1999
Map Number	11FCFM04
Power	99.03%
Rod Position	D/224 steps
Greatest Tilt	1.337%
Greatest FDH (V+)	1.489
FDH Limit (V+)	1.6398
Greatest FDH (V5)	0.898
FDH Limit (V5)	1.5946
Most Limiting FQ	1.805
FQ Limit	2.258
Highest Deviation between measured and predicted integrals	5.8%
Core Average Axial Offset	1.937%

Figure 3.1
Indian Point Unit 3 Cycle 11
ICRR During Core Loading, Sheet 1 of 8

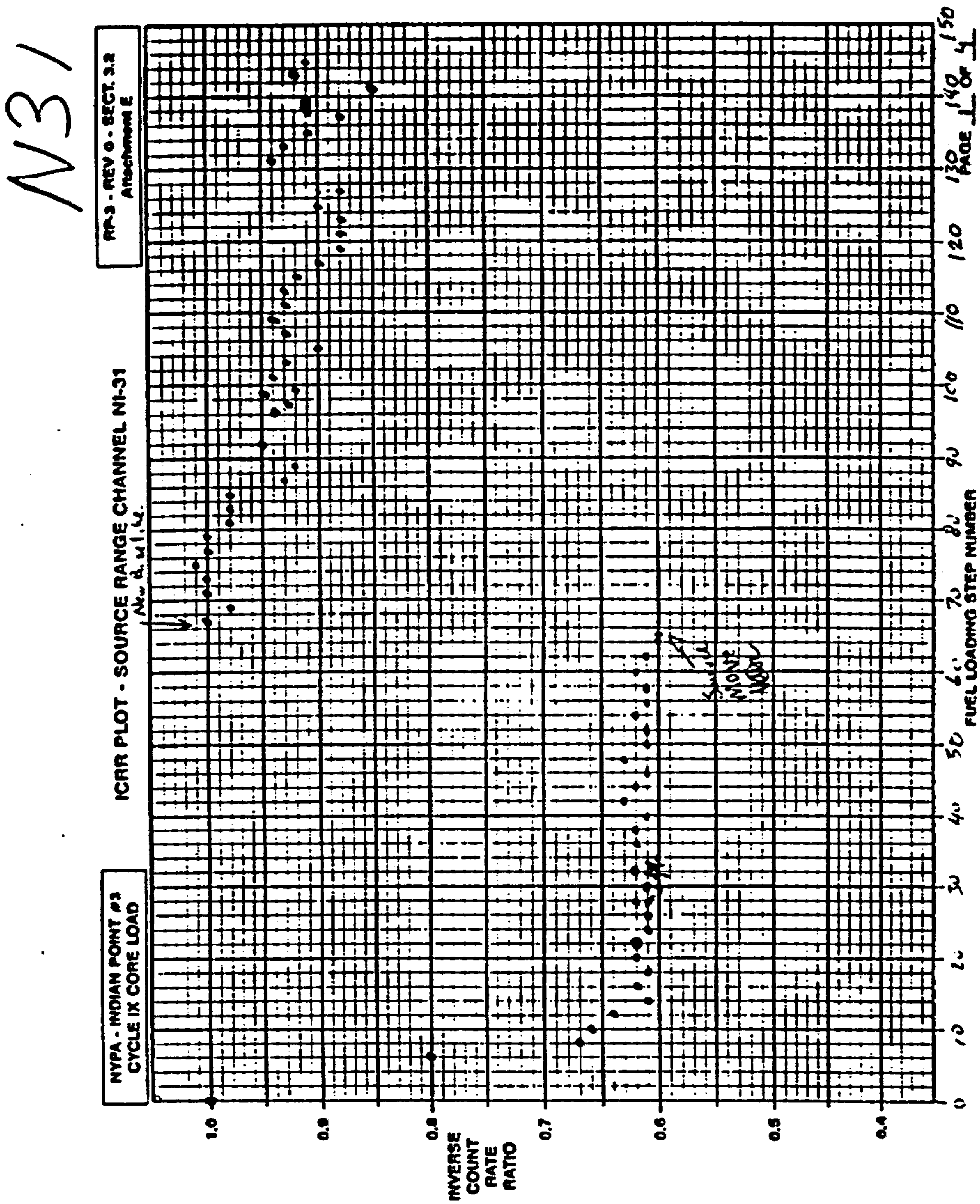


Figure 3.1
Indian Point Unit 3 Cycle 11
ICRR During Core Loading, Sheet 2 of 8

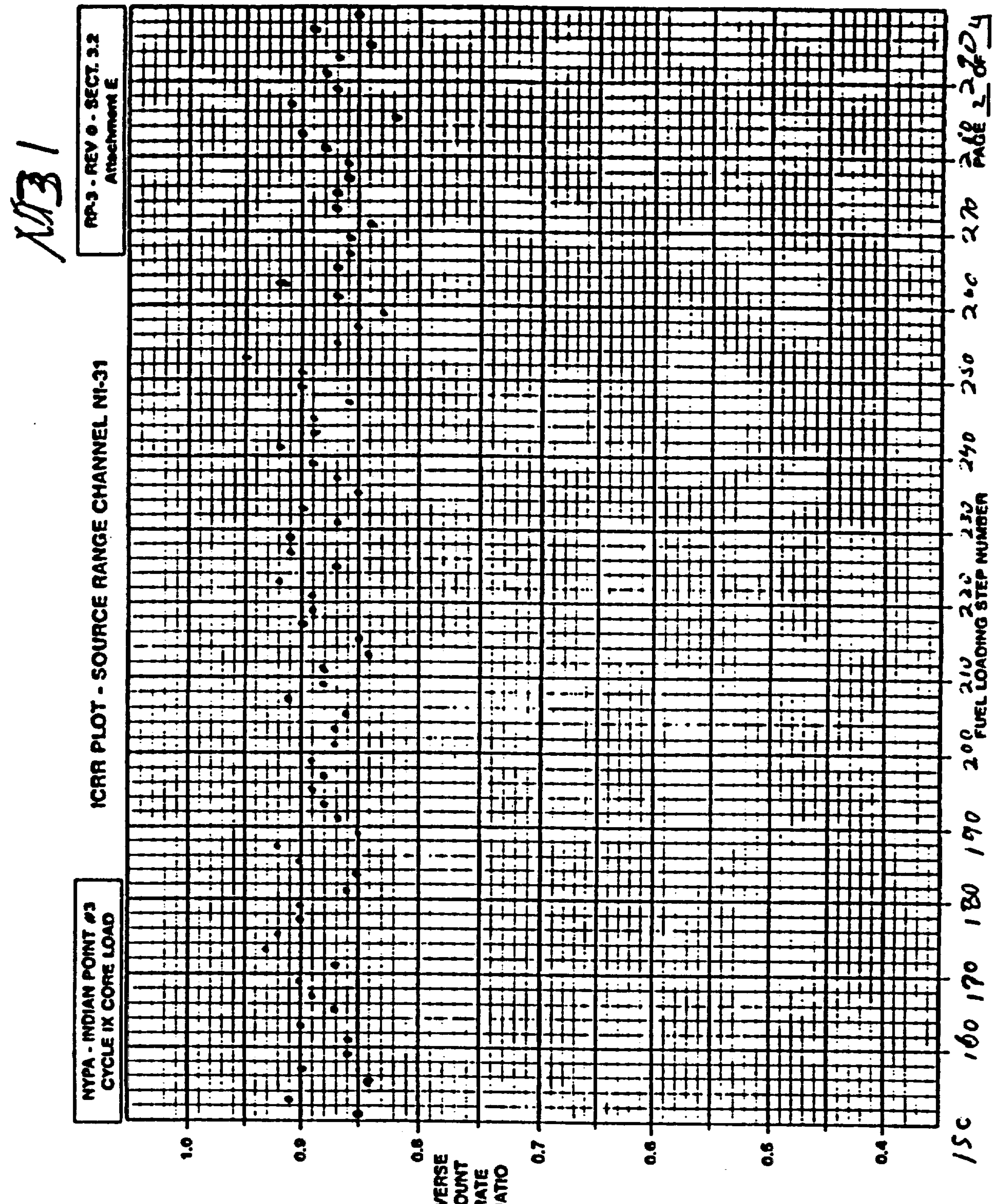
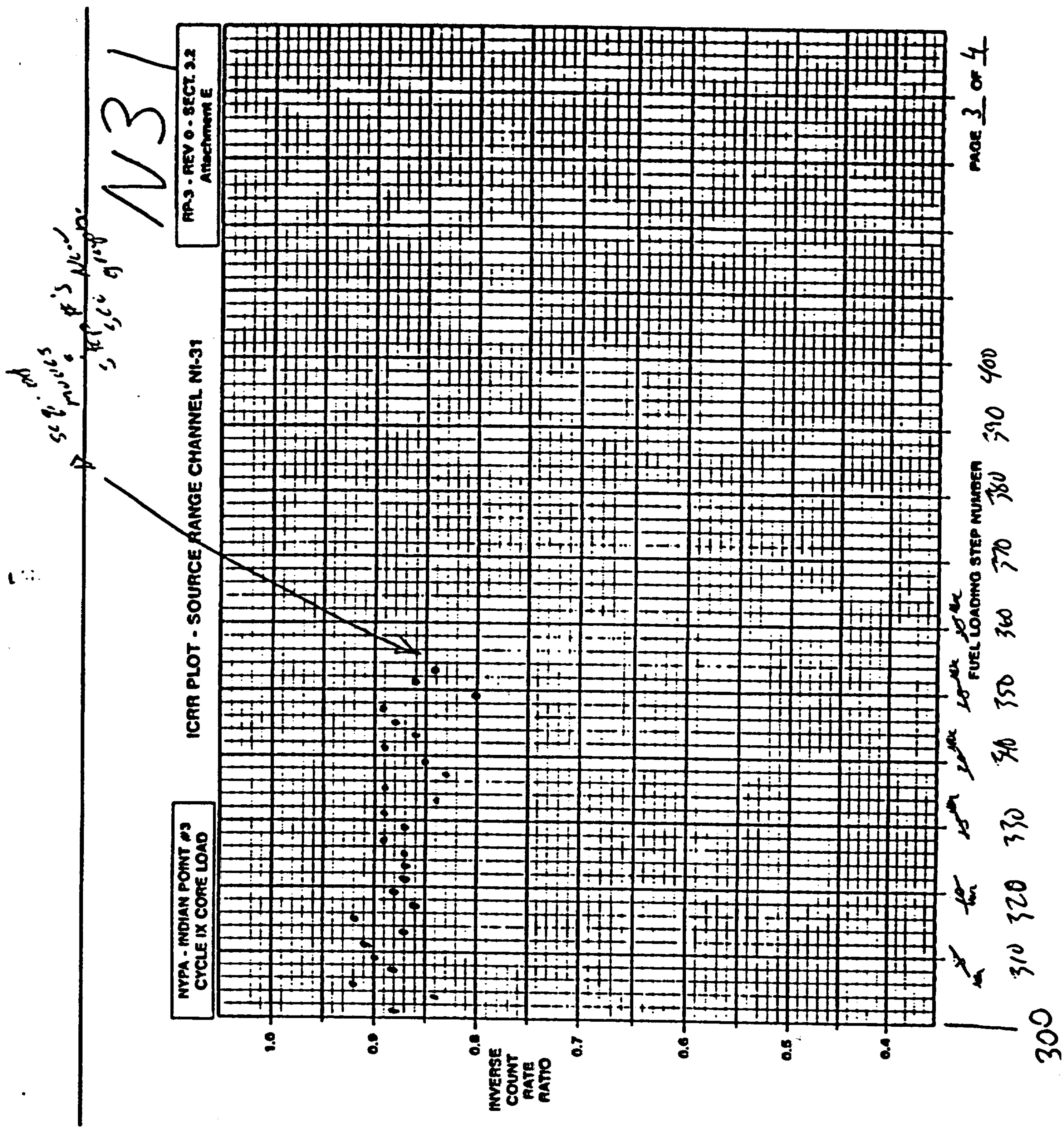


Figure 3.1
Indian Point Unit 3 Cycle 11
ICRR During Core Loading, Sheet 3 of 8



31 NI

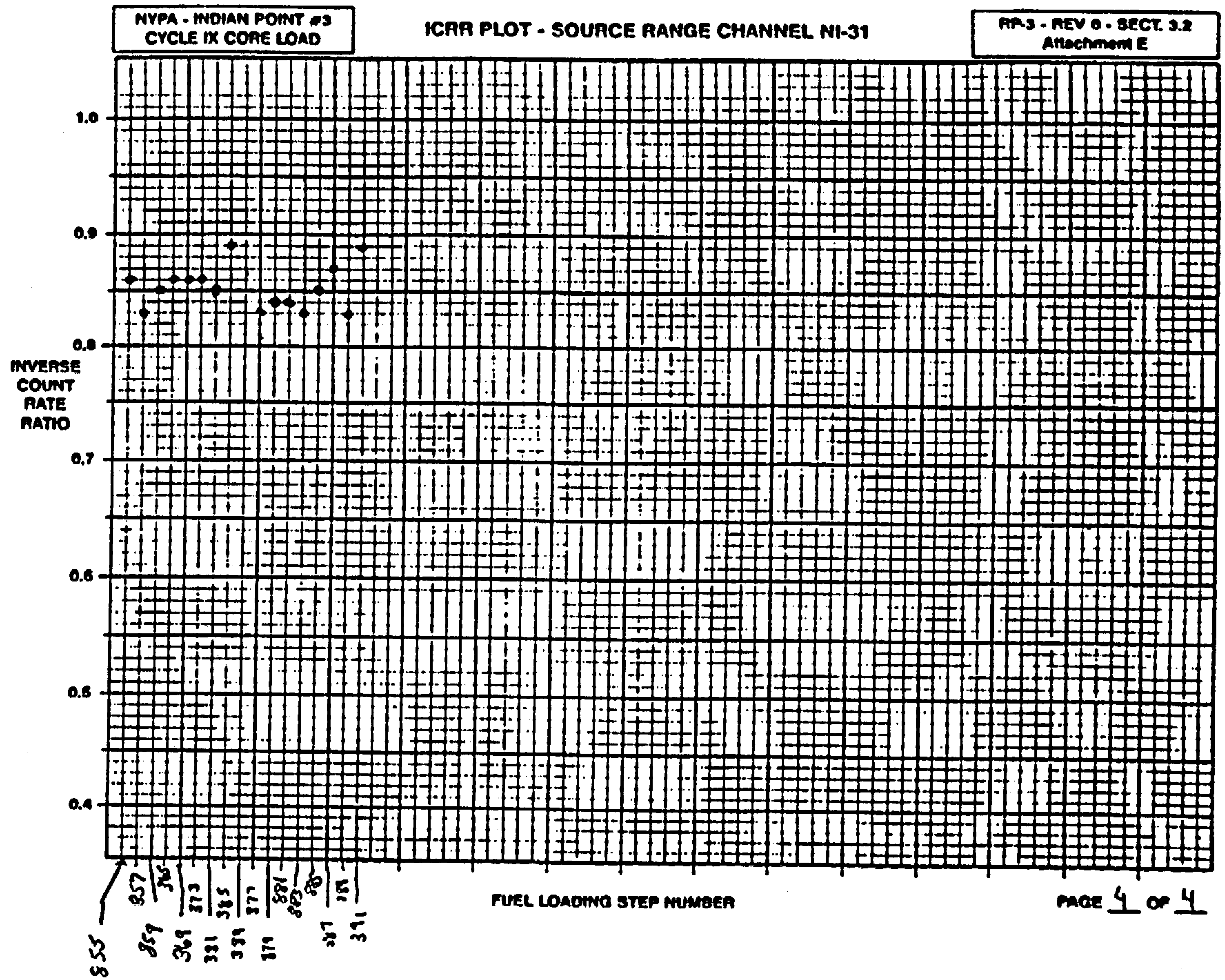
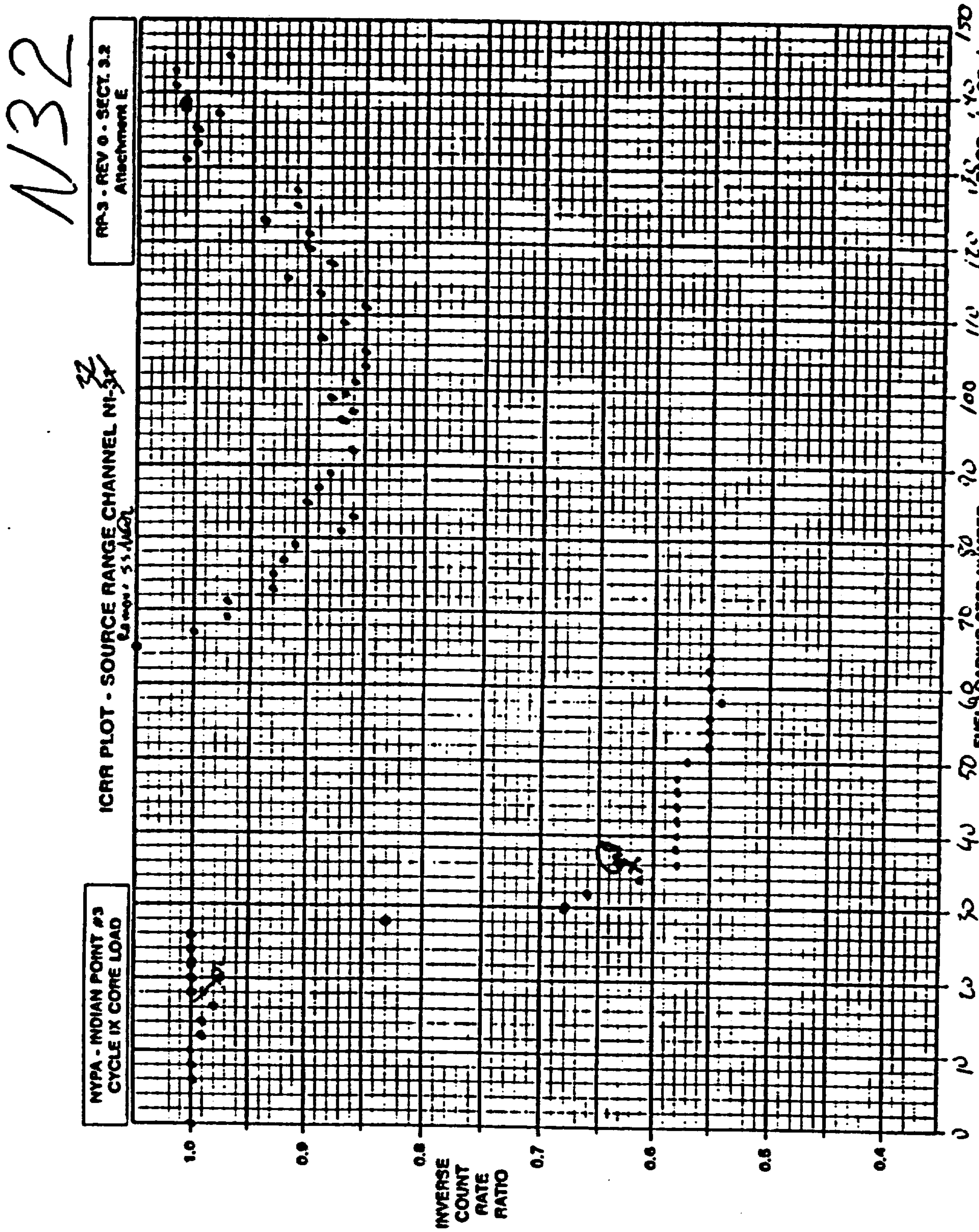


Figure 3.1
Indian Point Unit 3 Cycle 11
ICRR During Core Loading, Sheet 4 of 8

Figure 3.1
Indian Point Unit 3 Cycle 11
ICRR During Core Loading, Sheet 5 of 8



NJ 32

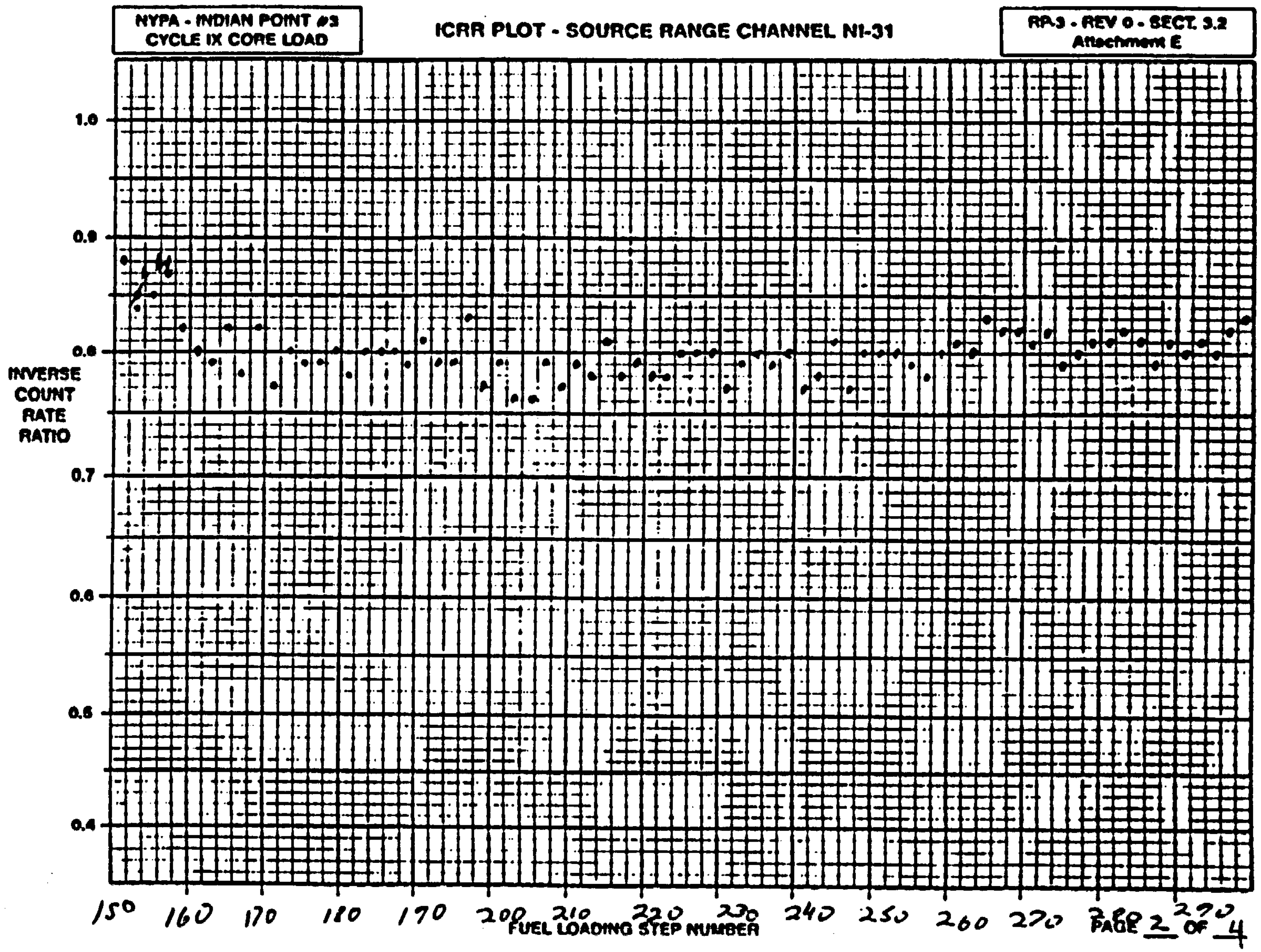


Figure 3.1
Indian Point Unit 3 Cycle 11
ICRR During Core Loading, Sheet 6 of 8

N32

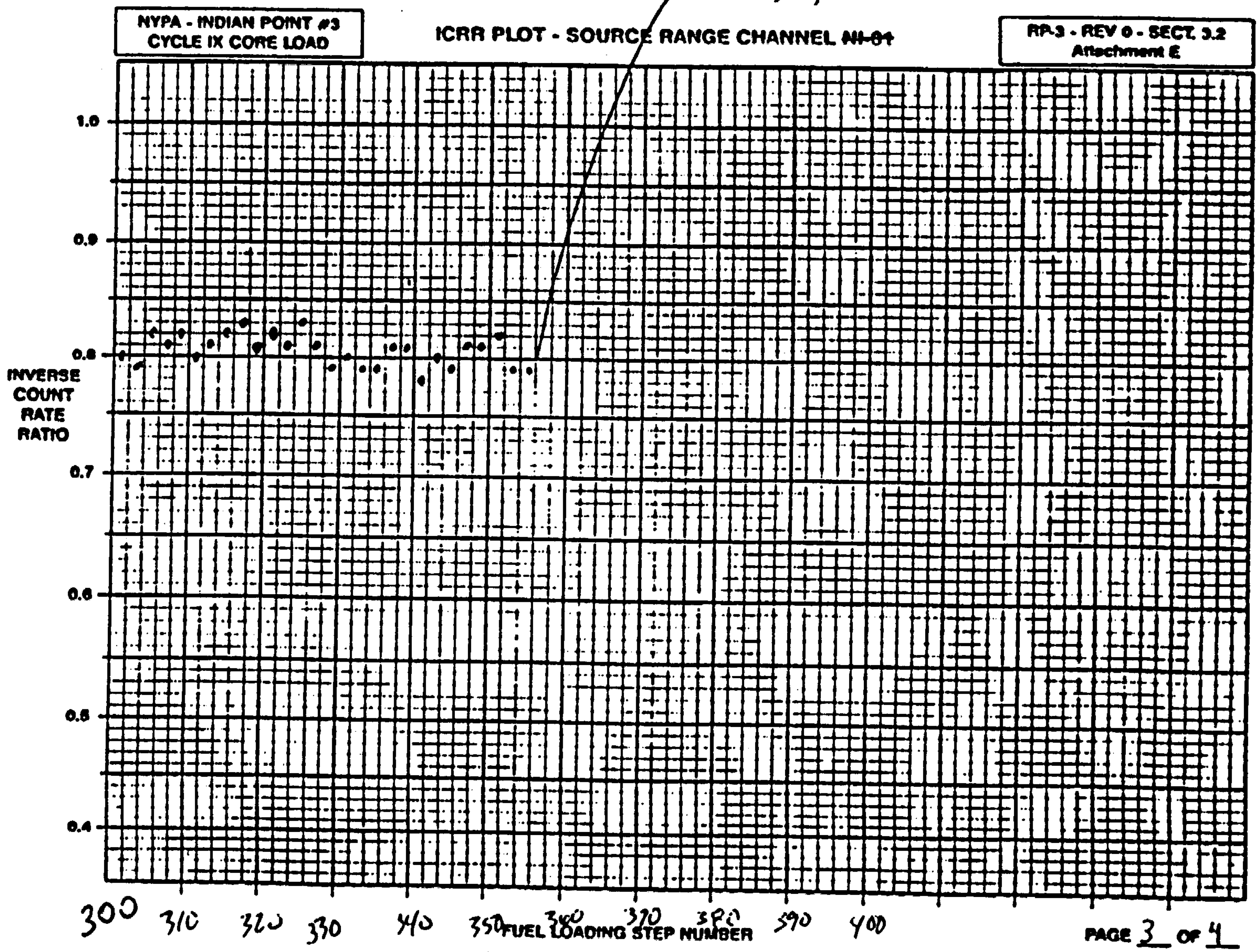
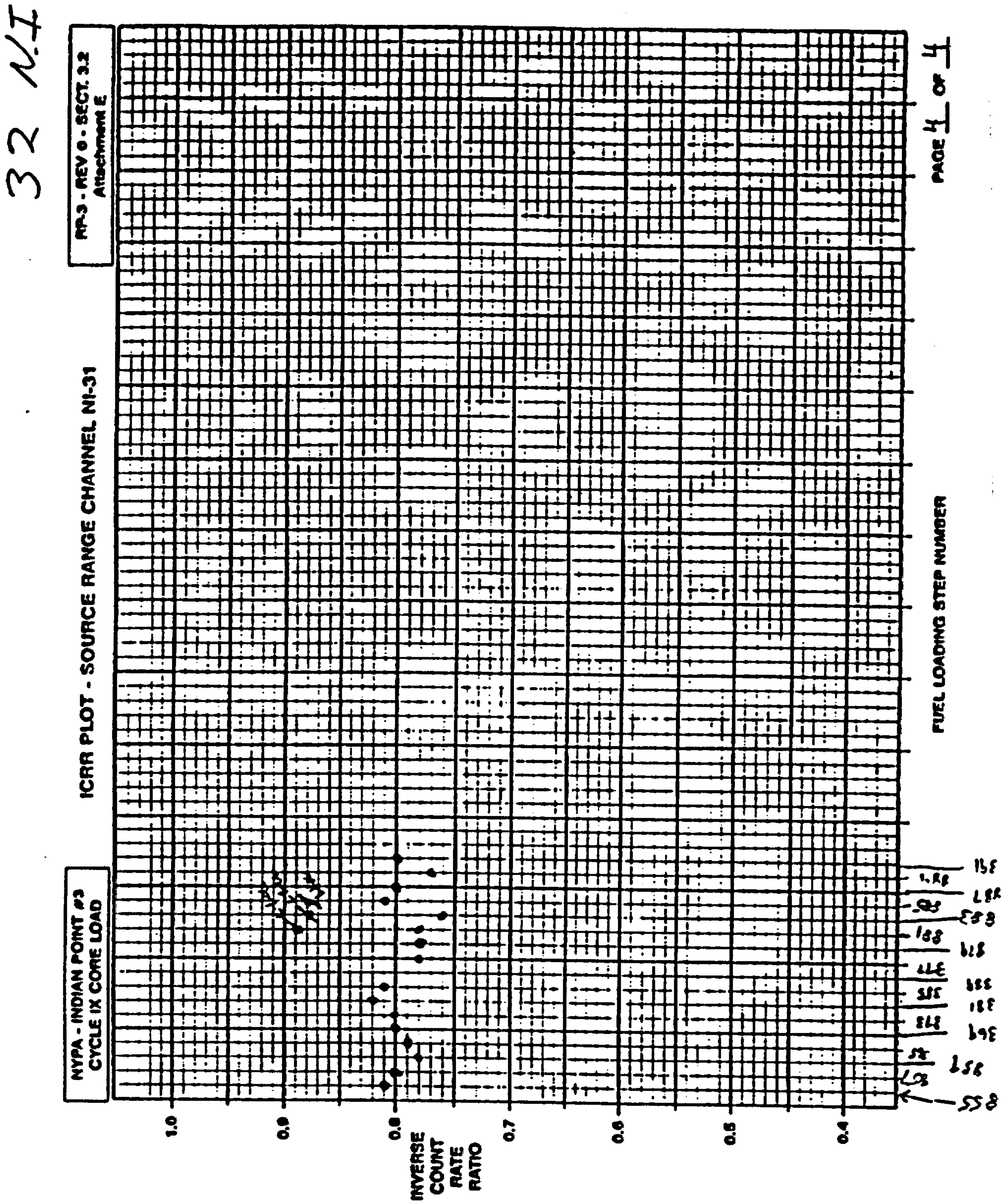
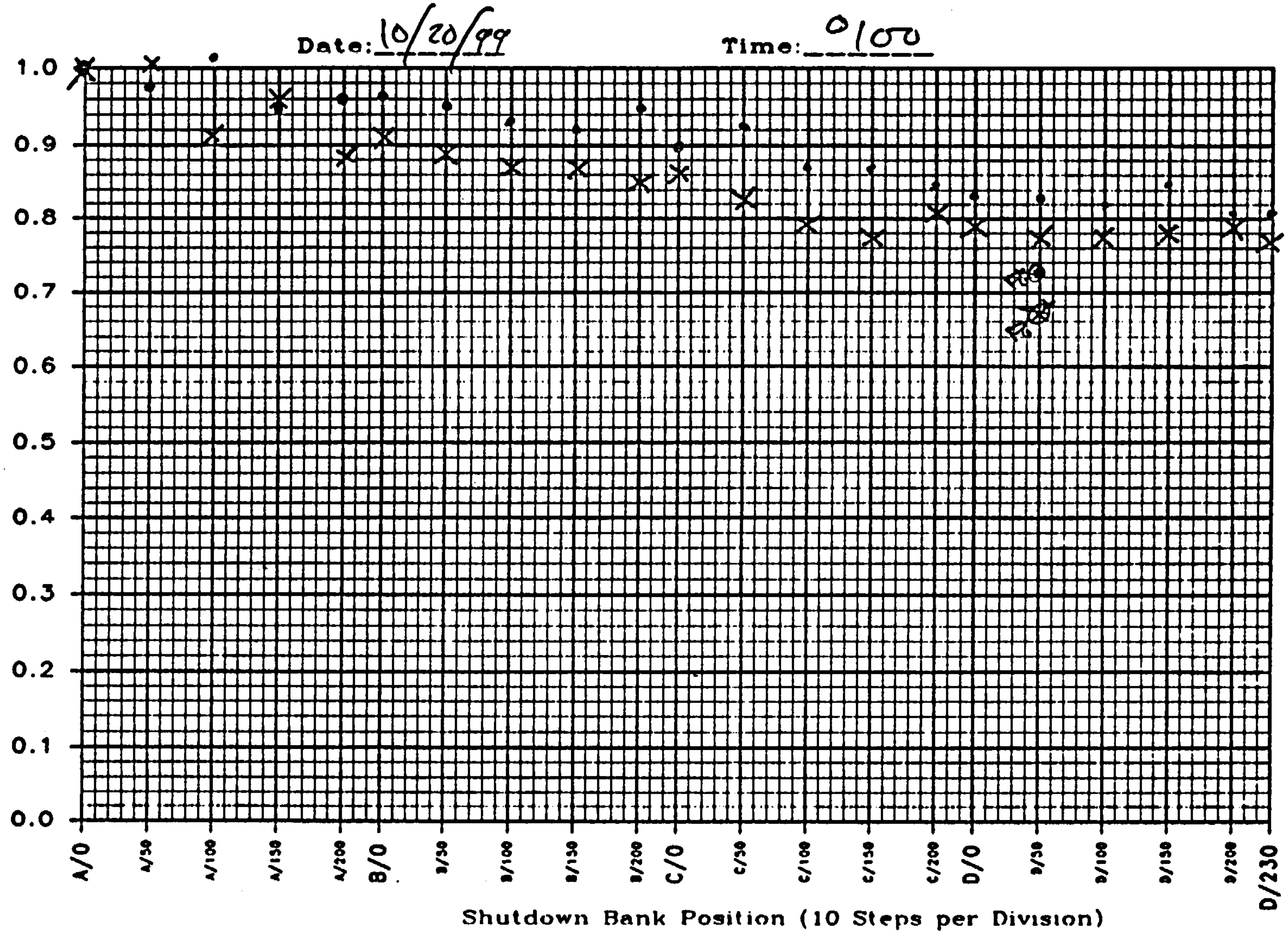


Figure 3.1
Indian Point Unit 3 Cycle 11
ICRR During Core Loading, Sheet 7 of 8

Figure 3.1
Indian Point Unit 3 Cycle 11
ICRR During Core Loading, Sheet 8 of 8





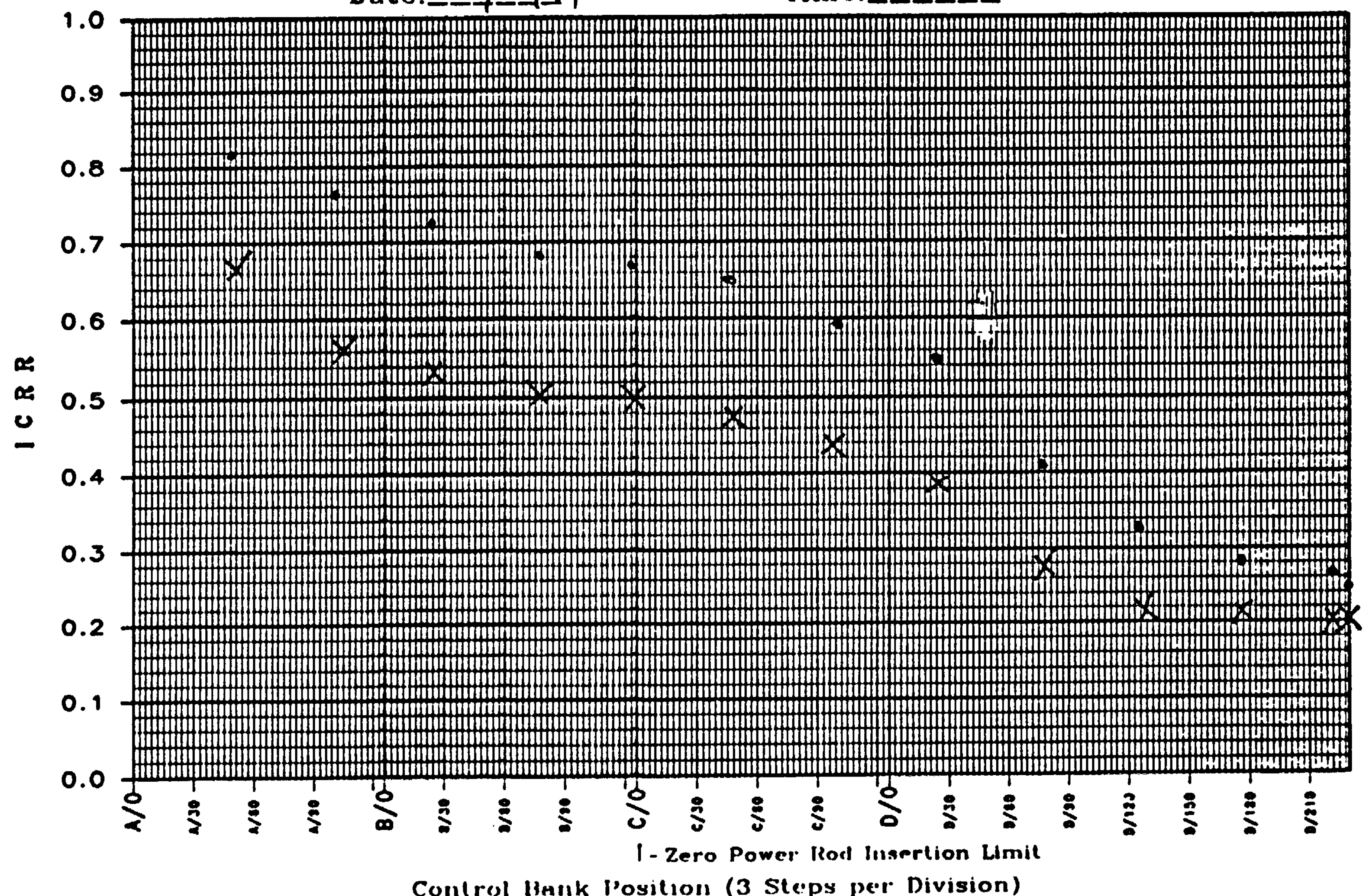
ICRR PLOT FOR SHUTDOWN BANK WITHDRAWAL

• = SR31
X = SR32

Figure 3.2
Indian Point Unit 3 Cycle 11
ICRR vs. Bank Position, Sheet 1 of 2

ICRR PLOT FOR CONTROL BANK WITHDRAWAL

Date: 10/20/99 Time: 0356



• = N31
x = N32

Figure 3.2
Indian Point Unit 3 Cycle 11
ICRR vs. Bank Position, Sheet 2 of 2

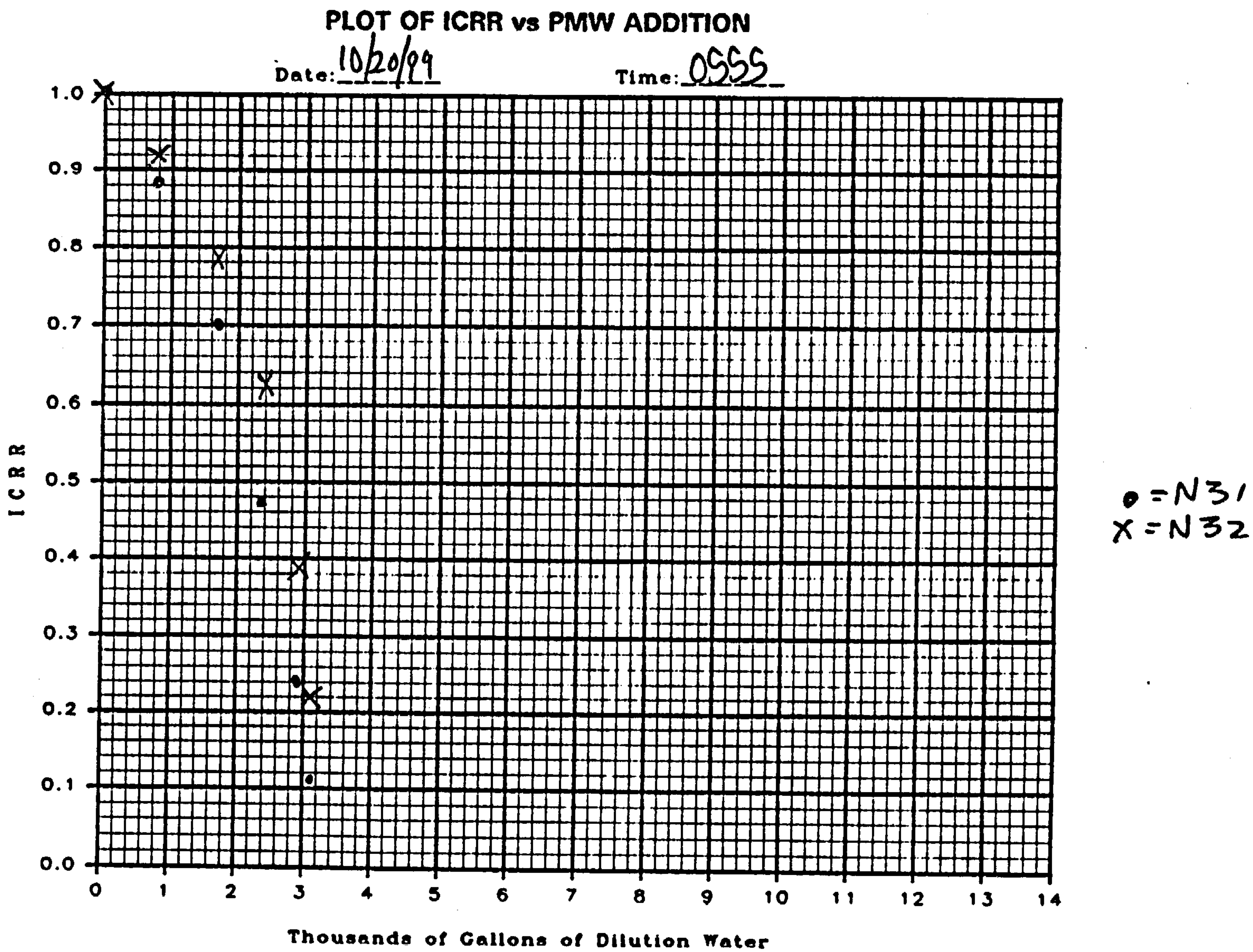
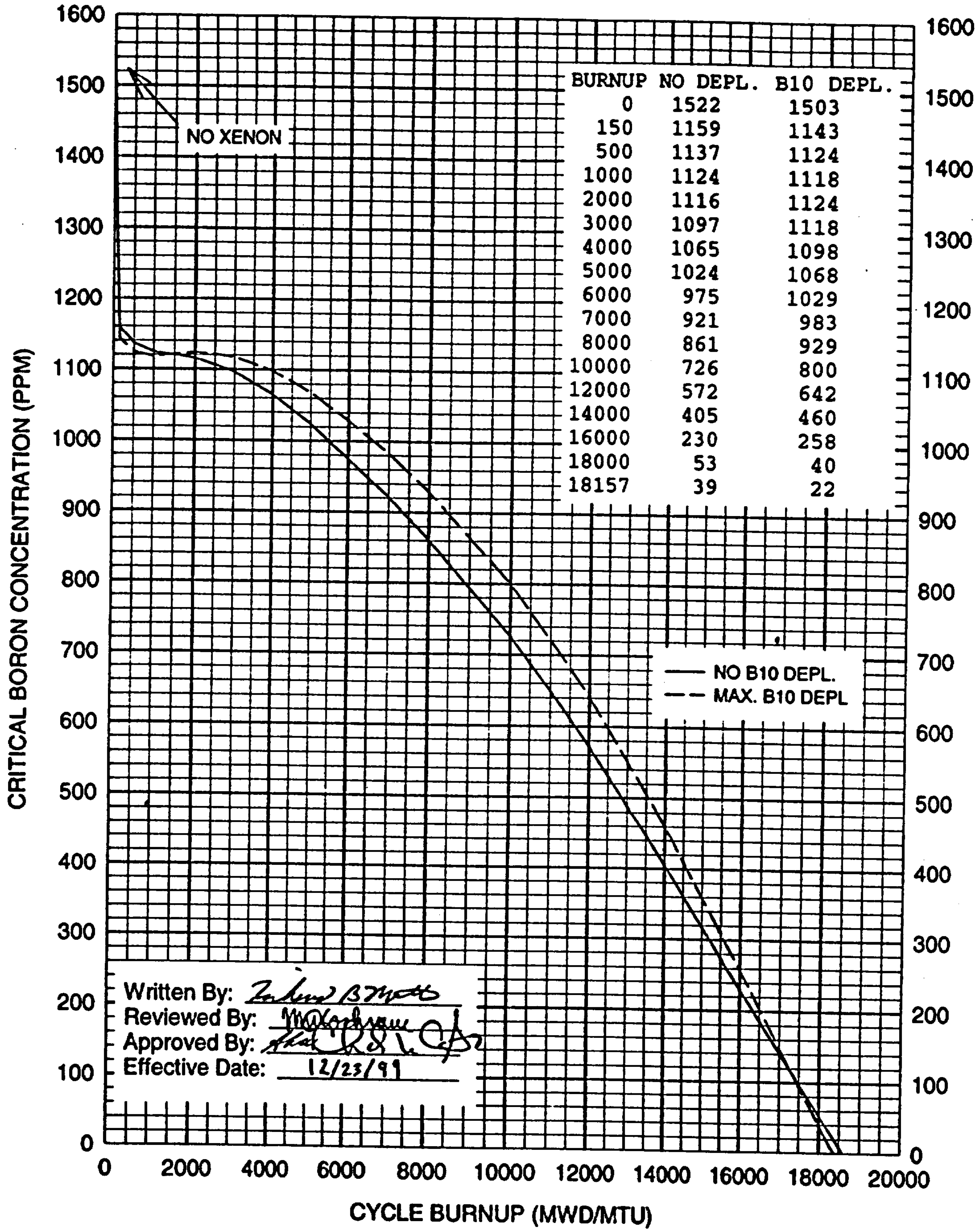


Figure 3.3
Indian Point Unit 3 Cycle 11
ICRR vs. Gal Dilution Water Added

Figure 3.4
Indian Point Unit 3 Cycle 11
Critical Boron Concentration vs. Burnup (Graph RCS-17A)



4.0 Instrument Measurements / Calibrations

4.1 Incore Thermocouple, Wide Range and Narrow Range RTD Measurement

The primary purpose of this test was to verify that the narrow range RTD's were functioning properly. This was accomplished by making comparative measurements of narrow range RTD's at five different temperatures (373, 408, 449, 500, and 541° F) while the reactor coolant system was held in an approximately isothermal condition. Only narrow range RTD's that deviated from the mean by less than 0.5°F are used for reactor protection and control. All narrow range RTD's met the acceptance criteria.

Additionally, this test collected wide range RTD readings and core exit thermocouple readings at the same temperature plateaus.

4.2 Incore - Excore Detector Calibration

As described in sections 3.5.2 and 3.5.3, full core flux maps taken at approximately 43% and 99% power to were used to obtain calibration data for the power range excore instrumentation. The calibration data was developed using the SPEDCAL function of the BEACON code. This methodology requires only the input of a single flux map to develop calibration data and eliminates the need to collect data at varying axial offsets and the associated core perturbations.

4.3 Calibration of OPDT and OTDT Setpoints

Steam generator Tave and Delta-T was measured at approximate power levels of 30, 50, 60, 70, 80, and 90%. Prior to exceeding 90 percent power, enthalpy-corrected extrapolations of full power values was calculated and are presented in Table 4.1. The extrapolated full power values were used to recalibrate the overpower and overtemperature reactor protection setpoint. The extrapolated values were verified once the plant reached full power and were determined to be accurate.

Table 4.1
Extrapolated Full Power Temperatures

Delta T (°F)		Tavg (°F)
Loop 31	54.33	567.51
Loop 32	51.89	566.52
Loop 33	53.63	567.27
Loop 34	53.94	566.04

4.4 Calibration of "High T_{ave}" Alarm

In order to ensure that T_{cold} does not exceed 547.9°F, as specified in the cycle 11 safety analysis, the "High T_{avg}" alarm setpoint was verified to be set conservatively at 571.3°F. This was based on calculations from the extrapolated full power core Delta-T listed in Section 4.3.