Attachment I to IPN-97-167

Cycle 10 Startup Physics Test Report

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

ATTACHMENT I

INDIAN POINT 3 START-UP

SUMMARY REPORT

Executive Summary / Abstract

During May - September 1997, Indian Point Unit 3 was shutdown for a scheduled refueling outage. At the conclusion of the outage, a series of preoperational 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 Measurements	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 T avarage" Alarm	Satisfactory
Calibration of New Power Range Detectors	Satisfactory
Full Power Critical Boron Measurements	Satisfactory

The unit subsequently achieved full power on September 27, 1997.

This report contains detailed descriptions of the cycle 10 core and each of the tests listed above.

Indian Point Unit 3 Cycle 10
Zero Power Physics Testing Results

I. Critical Boron Concentrations (PPM)

Design Review Criteria (DRC) = ±50 PPM and ±500 PCM
Acceptance Criteria (AC) = within 1000 PCM (144 PPM)

	Predicted (P)	Measured (M)	(M-P)	Pass / Fail DRC	Pass / Fail AC
ARO	1588	1575.0	13.0	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	867.1	894.9	+3.2	P	-
Control B	460.7	506.5	+9.9	P	-
Control C	619.9	636.0	+2.6	P	-
Control D	786.7	814.8	+3.6	P	-
Shutdown A	1015.0	1069.2	+5.3	P	-
Shutdown B	824.0	889.6	+8.0	P	-
Shutdown C	304.8	321.5	+5.5	P	-
Shutdown D	418.5	457.8	+9.4	P	-
Total	5296.7	5590.3	+5.5	P	P

III. Isothermal Temperature Coefficient (PCM / F)

Design Review Criteria = ± 2 PCM / F

	Predicted (P)	Measured(M)	(M-P)	Pass / Fail DRC
ARO	-4.16	-3.94	+0.22	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	-2.54	-2.32	+0.22	P

ARO: All Rods Out

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

* Inferred MTC is obtained by subtracting Doppler Coefficient (-1.56 PCM / F) from the Isothermal Temperature Coefficient.

Indian Point Unit 3 Cycle 10
at Power Physics Testing Results

I. Power Distribution Measurements

A) Low Power (29.33%)

Tilts

1.0014	0.9867
0.9940	1.0179

Largest Reaction Rate Integral Deviation - 6.1%

Limiting FQ - 1.8832 FQ Limit - 4.8400

Highest FDHN (V+) -1.4905 FDHN Limit (V+) -1.9816

Highest FDHN (V5) -1.4417 FDHN Limit (V5) -1.9271

B) Intermediate Power (69.45%)

Tilts

1.0049	0.9931
0.9898	1.0122

Largest Reaction Rate Integral Deviation - 5.2%

Limiting FQ 1.7188 FQ Limit 3.2429

Highest FDHN(V+) -1.4571 FDHN Limit (V+) -1.7848

Highest FDHN(V5) -1.4124 FDHN Limit(V5) -1.7357

C) Full Power (99.7%)

Tilts

1.0064	0.9965
0.9901	1.0071

Largest Reaction Rate Integral Deviation - 5.2%

Limiting FQ - 1.8050 FQ Limit -2.4267

Highest FDHN(V+) -1.4234 FDHN Limit(V+) -1.6364

Highest FDHN(V5) -1.3547 FDHN Limit(V5) -1.5913

II. Reactor Coolant System Flow Measurement

Measured Flow - 394857.4 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 (EFPD)	Predicted (P)	Measured (M)	(M-P)	Pass / Fail DRC	Pass / Fail AC
17.6	1018.2	1042.3	-24.1*	P	P
49.4	1081.2	1087.7	-6.5**	P	P

Note: Design boron letdown curve reduced by 11 ppm per TS 3.10.10 based on the average of reactivity measurements. Cycle 10 core design assumes 19.9 atom% B-10 in RCS. Actual B-10 atom % is 20.4. Accounting for this difference, the cycle 10 core matches core design data.

* Non-equilibrium Samarium.
** Equilibrium Samarium Conditions.

Due to difference in Samarium modeling, the deviation between Predicted and measured Critical Boron is greater before Samarium reaches equilibrium.

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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.

The Nuclear Steam Supply System was designed by Westinghouse Electric Corporation.

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 10 startup testing program:

The objectives of the physics test 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, and (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 10 core contains three regions of Westinghouse VANTAGE 5 (V5) fuel (Regions 9-2, 10-2, 11-1, and 11-2) and one region of Vantage + (V+) fuel (Regions 12-1 and 12-2). The Cycle 10 core has the following unique features described below:

A: During fuel examination, in mast sipping identified 8 fuel assemblies which potentially contained failed rods. After ultrasonic testing (UT), 5 assemblies were found to have failed rods. X60, V07 and W33 all had a single failed rod. V6 and V22 had two failed rods each. A total of 5 failed fuel assemblies with 7 failed rods were found. X60 was returned to the core so a natural uranium rod was inserted in place of the failed rod.

B. Control rod drag testing was performed. This procedure was done per NRC requirement due to control rods not fully inserting during scrams at other Westinghouse plants. No excessive drag forces were found.

C. Eddy current testing was performed on the control rods to identif y which should be replaced. Control rods were examined for excessive wear and the 14 with the most wear and cracking were replaced. Results showed that none of the control rods needed to be replaced. This was the first time control rod wear measurements had been performed at IP3.

D. The 88 feed assemblies in the Cycle 10 are Vantage + type fuel assemblies which include Intermediate Flow Mixing Grids (IFM's). The IFM's create more turbulent flow and allow for higher peaking factors. These higher peaking factors are exclusive to V+ fuel. This results in two FDH values one for each V+ fuel and V5 fuel.

E. Three different types of burnable poisons are being used in the Cycle 10 core: 1) A 20-pin pyrex poison insert is located in the assemblies in the "corners" of the core (8 total) as a means of reducing neutron fluence on the reactor vessel. 2) All of the eighty-eight feed assemblies contain integral fuel burnable absorber (IFBA) rods. These assemblies contain a specific pattern of 80 IFBA rods. 3) Wet Annular Burnable Absorber (WABA) inserts are used to provide additional hold-down in 60 of the 88 feed assemblies. Figure 1.2 shows the location of all burnable absorbers in the cycle 10 core.

1.4 Sequence of Startup Events

Following core loading, July 28 - August 1, 1997, a series of pre-operational test were performed both in the cold shutdown and hot shutdown conditions. Initial cycle 10 criticality tests was achieved on September 7, 1997 followed by a program of low power physics tests. The unit was synchronized to the grid on September 12, 1997. Full power was achieved on September 27, 1997.

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

Core Design Parameters

Number of Fuel Assemblies	193
Region 9-2	1
Region 10-2	24
Region 11-1	56
Region 11-2	24
Region 12-1	56
Region 12-2	32
Lattice Configuration	15x15
Number of Fuel Rods Per Assembly	204
Fuel Loading, MTU	88.18
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
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 (Vantage 5)	1.59
Nuclear Enthalpy Rise, FΔH (Vantage +)	1.635
Average Linear Power, kw/ft Fuel	6.24
Specific Power, kw/kg Uranium	34.30
Initial Enrichments, w/o Uranium 235	
Region 9-2	3.80
Region 10-2	4.20
Region 11-1	4.00
Region 11-2	4.40
Region 12-1	4.40
Region 12-2	4.60

Table 1.2
Indian Point Unit 3 Cycle 10
Zero Power Physics Testing Results

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Design Review Criteria (DRC) = ±50 PPM and ±500 PCM
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** Equilibrium Samarium Conditions.

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Figure 1.1

Core Layout

N

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15					W66 PD07Z	X18 PD38Z	X46 PD76Z	X26 PD80Z	X61 PD47Z	X06 PD77Z	W71 PD37Z				
14			W42 20P15X	X42 PD04Z	Y27 PD73Z	Y54 R123	Y64 12W0002	Y46 R116	Y86 12W0003	Y40 R128	Y31 PD76Z	X52 PD58Z	W44 20P16X		
13		W41 20P11X	X01 R73	Y30 SS16	X63 R98	Y77 20W0024	X07 R82	Y12 16W0007	X11 R106	Y71 20W0018	X81 R80	Y35 PD03Z	X03 R72	W69 20P12X	
12		X34 PD16Z	Y21 PD35Z	X05 R101	Y76 20W0023	X71 PD71Z	Y63 20W0012	X37 R100	Y68 20W0014	X89 PD119Z	Y74 20W0021	X56 R74	Y02 PD84Z	X35 PD55Z	
11	W86 PD27Z	Y16 PD82Z	X64 R92	Y82 20W0028	X33 PD50Z	Y06 20W0001	X78 PD28Z	Y50 16W0010	X73 R109	Y20 20W0004	X12 PD90Z	Y80 20W0026	X58 R79	Y04 PD12Z	W68 PD42Z
10	X49 PD18Z	Y39 R126	Y67 20W0013	X72 PD17Z	Y14 20W0017	X50 R122	Y49 16W0014	W62 R97	Y56 16W0020	X09 R70	Y33 20W0002	X66 PD29Z	Y69 20W0016	Y23 R125	X45 PD74Z
9	X40 PD54Z	Y70 12W0005	X21 R61	Y62 20W0011	X76 R95	Y18 16W0011	W53 PD85Z	Y10 16W0006	W67 PD51Z	Y13 16W0009	X75 PD13Z	Y58 20W0008	X16 R77	Y59 12W0001	X38 PD52Z
8	X25 PD34Z	Y52 R120	Y08 16W0004	X54 R104	Y42 16W0017	W50 R78	Y55 16W0019	V44 R830	Y37 16W0016	W54 R84	Y22 16W0012	X22 R81	Y53 16W0001	Y26 R127	X10 PD93Z
7	X17 PD63Z	Y84 12W0008	X08 R75	Y86 20W0030	X74 PD46Z	Y09 16W0005	W49 PD59Z	Y24 16W0013	W65 PD61Z	Y48 16W0018	X80 R83	Y87 20W0031	X24 R111	Y66 12W0004	X48 PD48Z
6	X39 PD58Z	Y17 R119	Y81 20W0027	X67 PD79Z	Y32 20W0003	X41 R90	Y05 16W0003	W48 R71	Y34 16W0015	X47 R115	Y25 20W0015	X68 PD70Z	Y79 20W0025	Y41 R121	X55 PD41Z
5	W43 PD38Z	Y03 PD05Z	X59 R62	Y60 20W0009	X27 PD81Z	Y19 20W0006	X77 R66	Y44 16W0008	X79 PD66Z	Y46 20W0005	X29 PD06Z	Y61 20W0010	X60 R107	Y43 PD95Z	W45 PD91Z
4		X30 PD19Z	Y07 PD30Z	X31 R99	Y73 20W0020	X65 PD94Z	Y85 20W0029	X43 R112	Y57 20W0007	X70 PD78Z	Y75 20W0022	X20 R69	Y11 PD15Z	X14 PD63Z	
3		W52 20P09X	X02 R108	Y15 PD86Z	X57 R85	Y88 20W0032	X28 R63	Y29 16W0002	X19 R88	Y72 20W0019	X62 R65	Y36 SS15	X04 R105	W47 20P10X	
2			W64 20P14X	X53 PD10Z	Y51 PD62Z	Y28 R118	Y83 12W0007	Y47 R117	Y78 12W0006	Y38 R124	Y01 PD40Z	X23 PD44Z	W70 20P13X		
1					W46 PD36Z	X32 PD43Z	X15 PD23Z	X36 PD11Z	X13 PD88Z	X44 PD31Z	W61 PD122Z				

Figure 1.2

Burnable Absorber Configuration

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

0°

270°

TYPE	TOTAL
##P...(NUMBER OF PYREX RODLETS)..............	160
##W...(NUMBER OF WABA RODLETS)...............	1056
##I...(NUMBER OF FRESH IFBA RODS)............	7040
#SSA..(NUMBER OF SECONDARY SOURCE RODLETS)...	8

2.0 Measurements

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. Monitoring was accomplished by determining the normalized inverse count rate ratio (ICRR) for each channel as the core was loaded (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 Small core reactivity changes were determined with the aid of a reactivity computer which 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 which logged the output from the reactivity computer.

The input signal to the reactivity computer was provided by one Nuclear Instrumentation System (NIS) power range channel. 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. 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 were measured.

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. For boron concentration endpoints, corrections to the measured concentration utilized reactivity computer measurements of the reactivity difference between actual and design core configurations.

2.3 Power Distribution

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) design calculations, and 3) margin in hot channel factors. The M/D system was also used at a fourth plateau to provide data for excore detector calibration. Data from the M/D system was input to the INCORE 3D computer code to generate detailed three dimensional core power profiles. The INCORE 3D 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 Instrument 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 meters. 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. Correlations 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 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 September 6, 1997 at 1306 hours with the incremental withdrawal of shutdown and control banks. Primary System boron concentration during rod withdrawal was approximately 1717 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 4.0×10^{-7} 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
227.8	28.1	28.0	-0.2 pcm

3.3.2 Boron Endpoints

The just critical boron concentration was measured for three rod configurations. The test results are summarized in Table 1.2 along with design predictions

3.3.3 Temperature Coefficient

Isothermal temperature coefficient measurements were performed at two core conditions, as summarized in Table 1.2. The all-rods-out, moderator-only temperature coefficient (MTC) was negative. However, since MTC increases with burnup, rod withdrawal limits were required 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 are 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 powers must be 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 insertion limit. The calculation method determines the boron concentration at a particular Control rod configuration where MTC is 0. A 10 ppm conservatism factor is included. This effect will be significant for approximately the first 5 months of operation until boron concentration starts decreaseing.

3.3.4 RCC Bank Worths

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 four banks are summarized in Table 1.2.

3.4 At Power Tests

3.4.1 RCS Flow Determination

On September 26, 1997 RCS flow was measured to be 394,857 gallons per minute. The flow assumed in the FSAR at the beginning of DNBR analysis is 375,600 gallons per minute. The actual measurement indicated that a margin of approximately 4.88 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 85% 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 30% power. The calculation was repeated at approximately 10% increments between 50% and 100% power. The high flux trip setpoint was raised back to 108% after reaching 70% power.

3.4.3 Full Power Critical Boron Measurement

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 24 ppm below the predicted value. As samarium reached equilibrium, the difference leveled off to approximately 6.5 ppm below the predicted value. Table 3.2, shows the reactivity balance results through the first full power month of operation. As required by T.S. 3.10.10, an 11 ppm adjustment factor was applied to the design boron curve.

Table 3.2
Reactivity Balance Summary

EFPD	Measured (PPM)	Predicted (PPM)	Delta (PPM)
17.6	1018.2	1042.3	-24.1
23.00	1029.4	1047.8	-18.4
28.5	1045.3	1054.3	-9.0
33.5	1046.1	1062.0	-15.9
37.4	1059.2	1068.3	-9.1
42.4	1068.9	1076.4	-7.5
44.4	1070.6	1079.6	-9.0
49.4	1081.2	1087.7	-6.5

3.5 Movable Detector Fluxmaps

3.5.1 Low Power

The initial fluxmap of cycle 10 was taken at approximately 29 percent power with equilibrium Xenon. The purpose of this map was to verify proper core loading. The greatest deviation between predicted and measured average reaction rate integrals was 6.1 percent at core location B-13. Based on a review of this map the core was determined to be properly loaded. A summary of parameters is presented below:

Date	September 13, 1997
Map Number	9FCFM1
Power	29.33%
Rod Position	D/165 steps
Greatest Tilt	1.79%
Greatest FDH (V+)	1.4905
FDH Limit (V+)	1.9816
Greatest FDH (V5)	1.4417
FDH Limit (V5)	1.9271
Most Limiting FQ	1.8832
FQ Limit	4.8400
Highest Deviation between measured & predicted Integrals	-6.1%
Core Average Axial Offset	-1.443

3.5.2 Intermediate Power

The second fluxmap of Cycle 10 was taken at approximately 69 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. The greatest deviation between predicted and measured average reaction rate integrals was -5.2 percent at core location R-6. 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	September 22, 1997
Map Number	9FCFM2
Power	69.45%
Rod Position	D/196 steps
Greatest Tilt	1.22%
Greatest FDH (V+)	1.4571
FDH Limit (V+)	1.7848
Greatest FDH (V5)	1.4124
FDH Limit (V5)	1.9271
Most Limiting FQ	1.7188
FQ Limit	3.2429
Highest Deviation between measured and predicted integrals	-5.2%
Core Average Axial Offset	-2.642%

3.5.3 Full Power

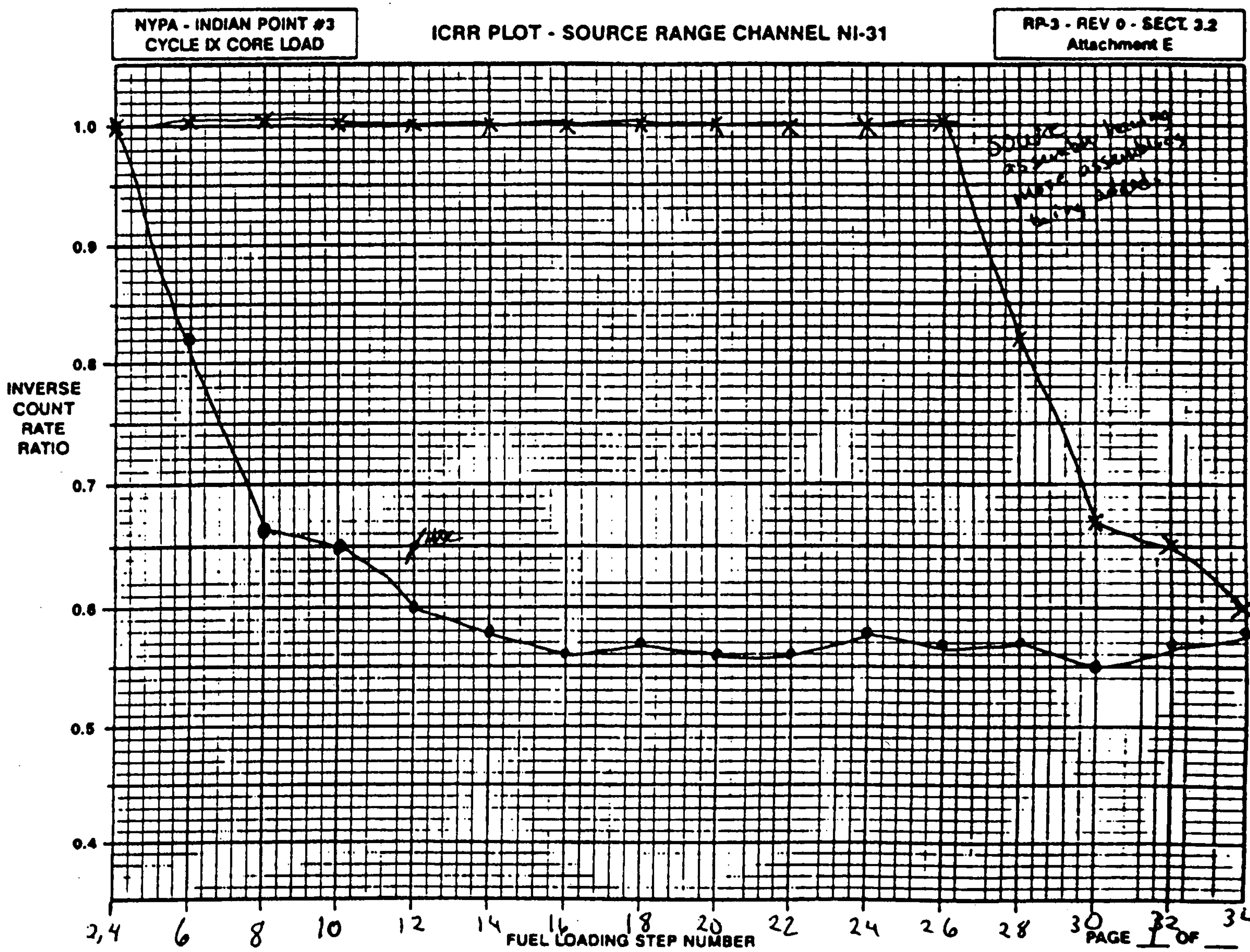
The initial full power fluxmap of cycle 10 was taken on October 1, 1997. The purpose of this map was to verify that measured full power hot channel factors (FQ, FDH) were within Technical Specification limits. Based on a review of this map all power distribution paremeters were within applicable limits. A summary of parameters is presented below:

Date	October 1, 1997
Map Number	9FCFM4
Power	99.72%
Rod Position	D/224 steps
Greatest Tilt	0.71%
Greatest FDH (V+)	1.4234
FDH Limit (V+)	1.6364
Greatest FDH (V5)	1.3547
FDH Limit (V5)	1.5913
Most Limiting FQ	1.8050
FQ Limit	2.4267
Highest Deviation between measured and predicted integrals	-5.2%
Core Average Axial Offset	-2.483%

● = N 31
X = N 32

Attachment E - Inverse Count Rate Ratio Plot (ICRR)

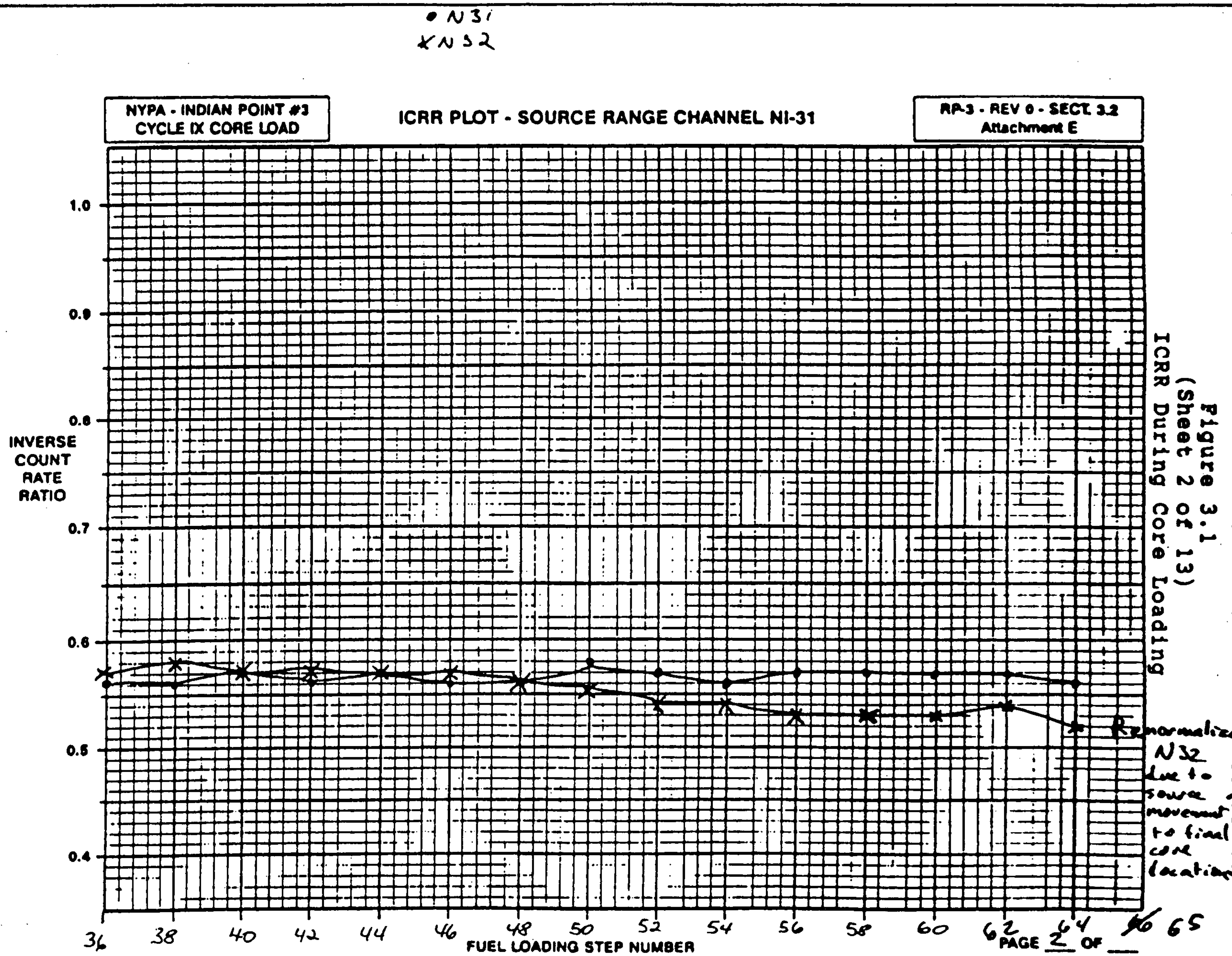
Figure 3.1
(Sheet 1 of 13)
ICRR During Core Loading



1 of

Attachment E - Inverse Count Rate Ratio Plot (ICRR)

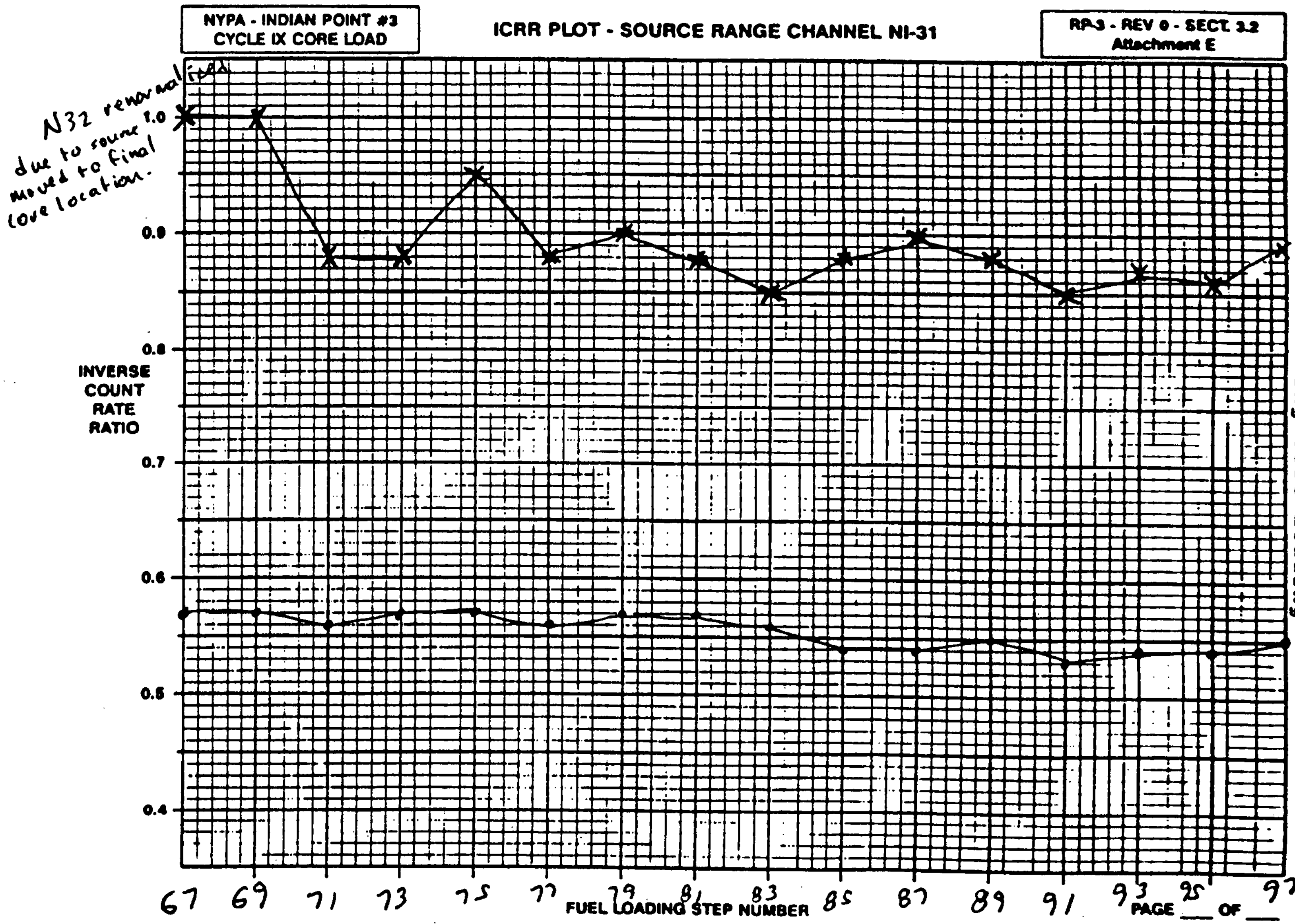
Figure 3.1
(Sheet 2 of 13)
ICRR During Core Loading



Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 3 of 13)
ICRR During Core Loading

• N31
x N32



3 of

Attachment E - Inverse Count Rate Ratio Plot (ICRR)

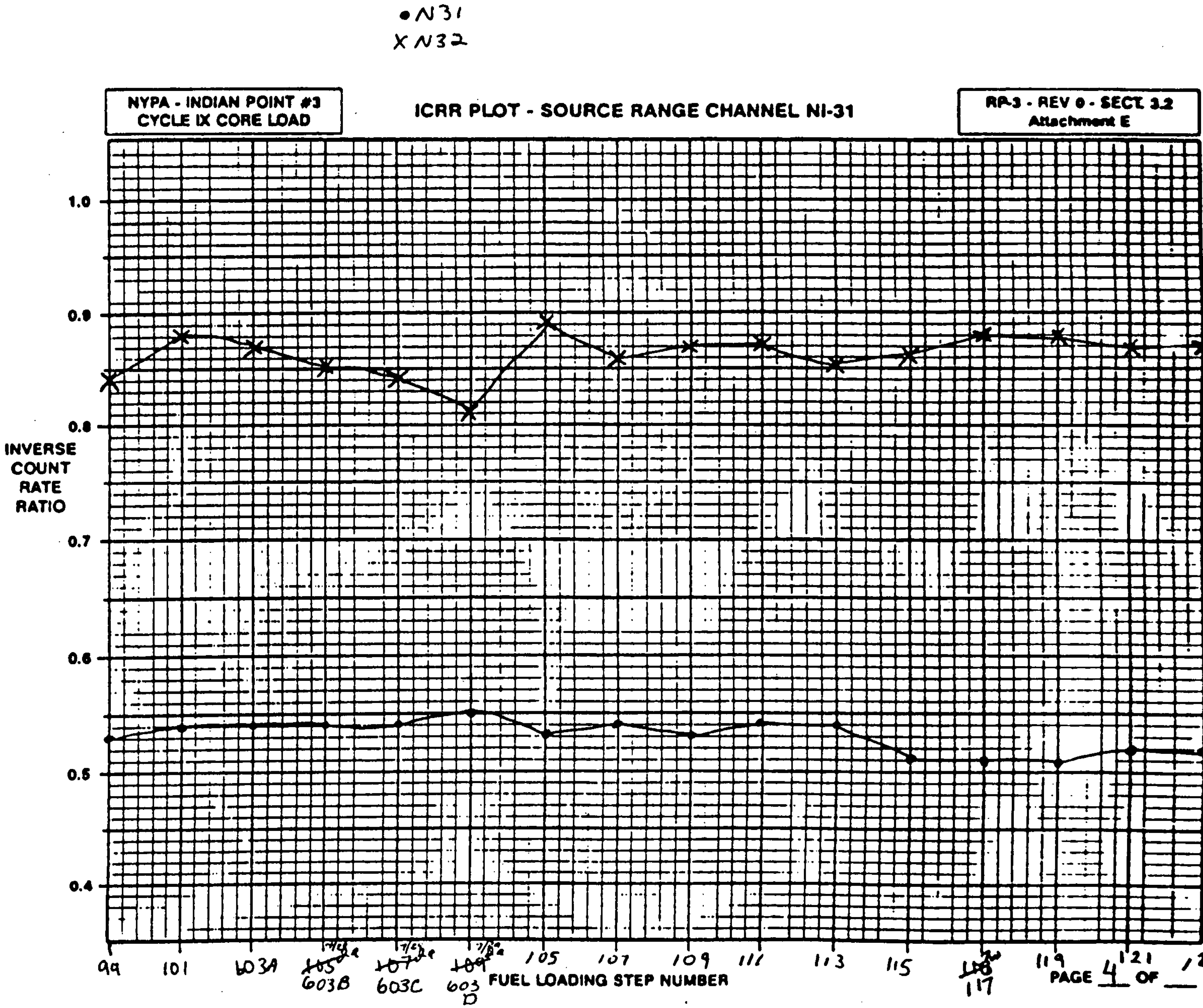
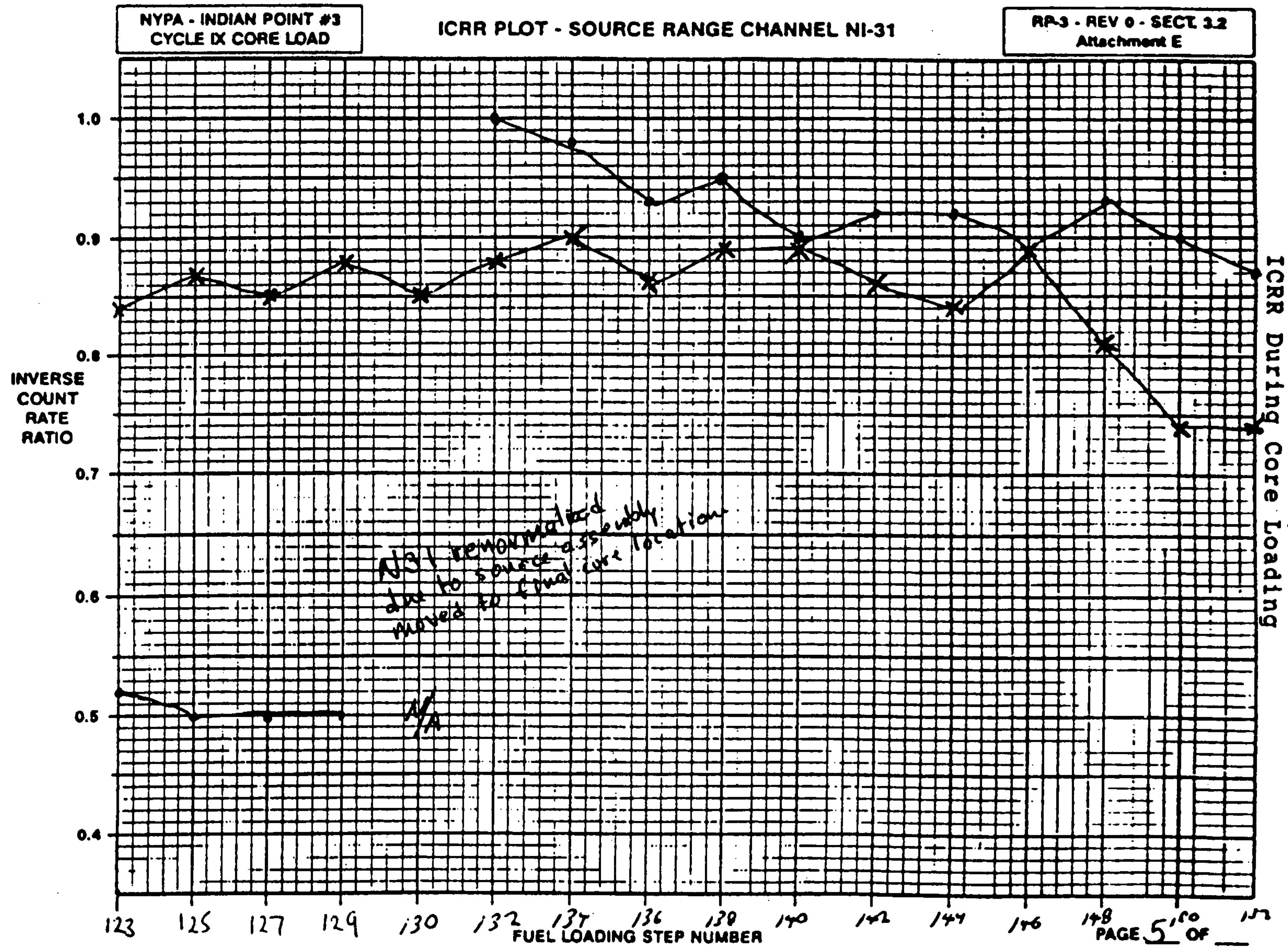


Figure 3.1
(Sheet 4 of 13)
ICRR During Core Loading

Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 5 of 13)
ICRR During Core Loading

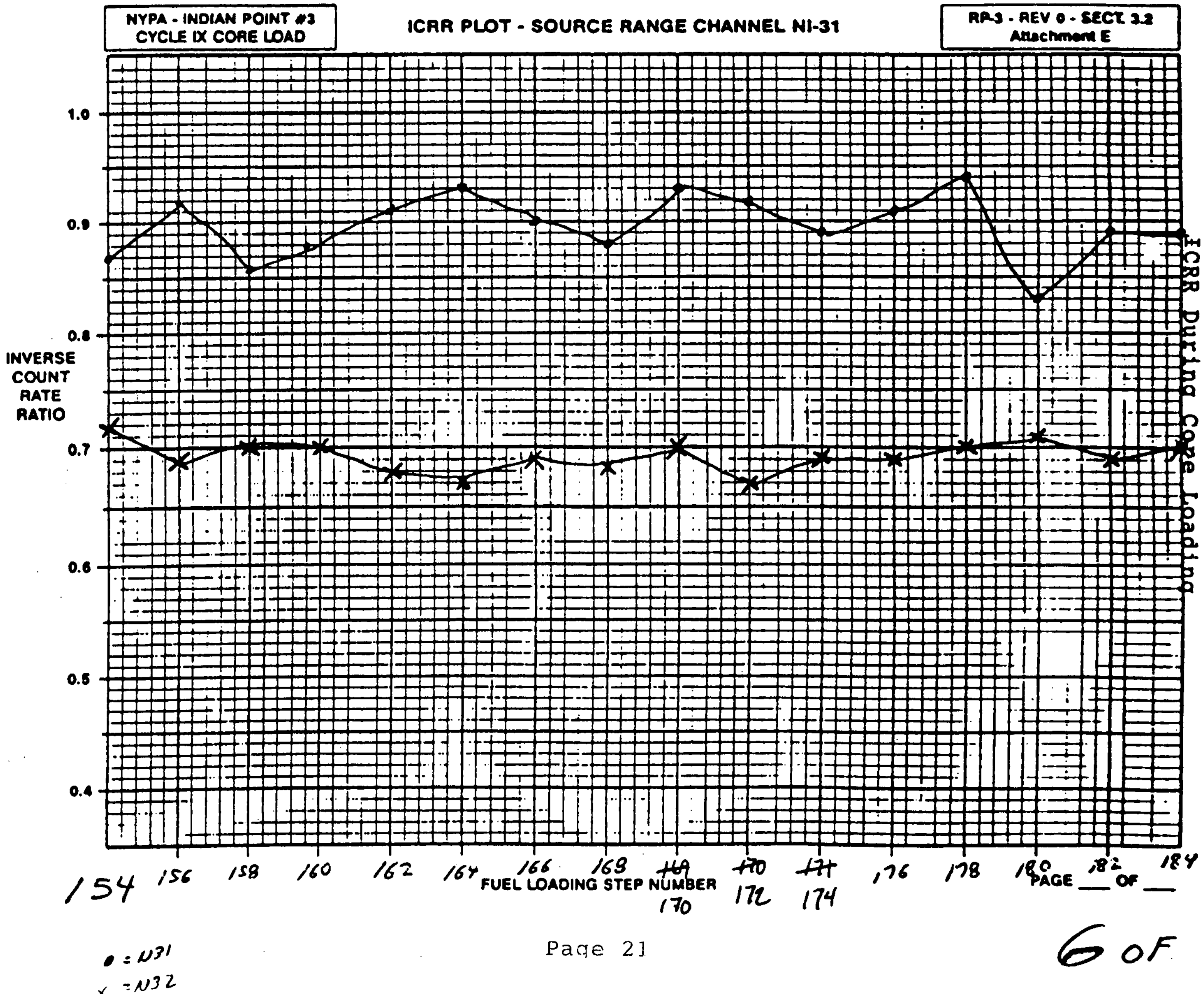


• = N31
X = N32

INVERSE COUNT RATE RATIO MONITORING DURING REFUELING

Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 6 of 13)
ICRR During Core Loading



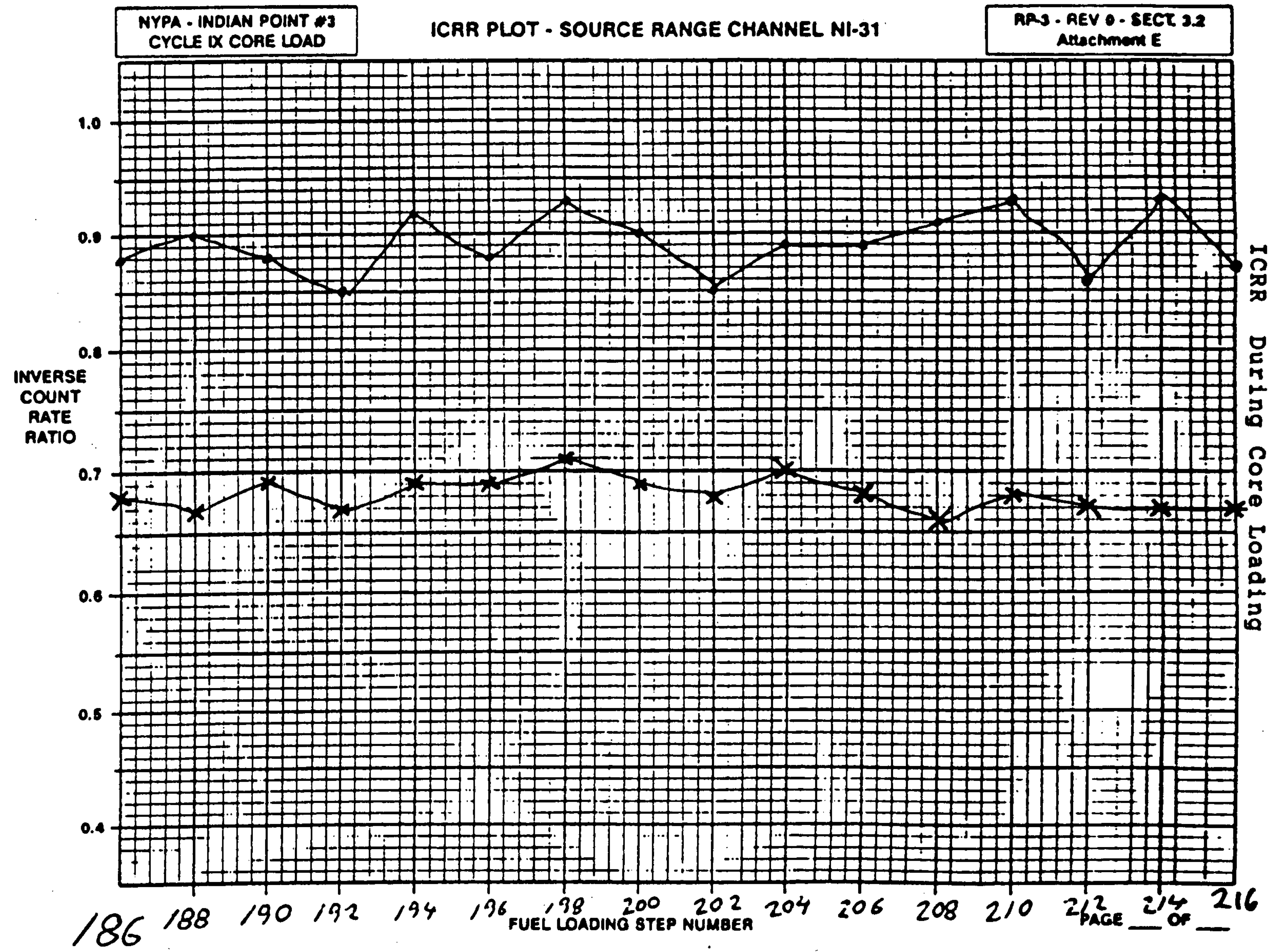
6 OF

• N31
x N32

INVERSE COUNT RATE RATIO MONITORING DURING REFUELING

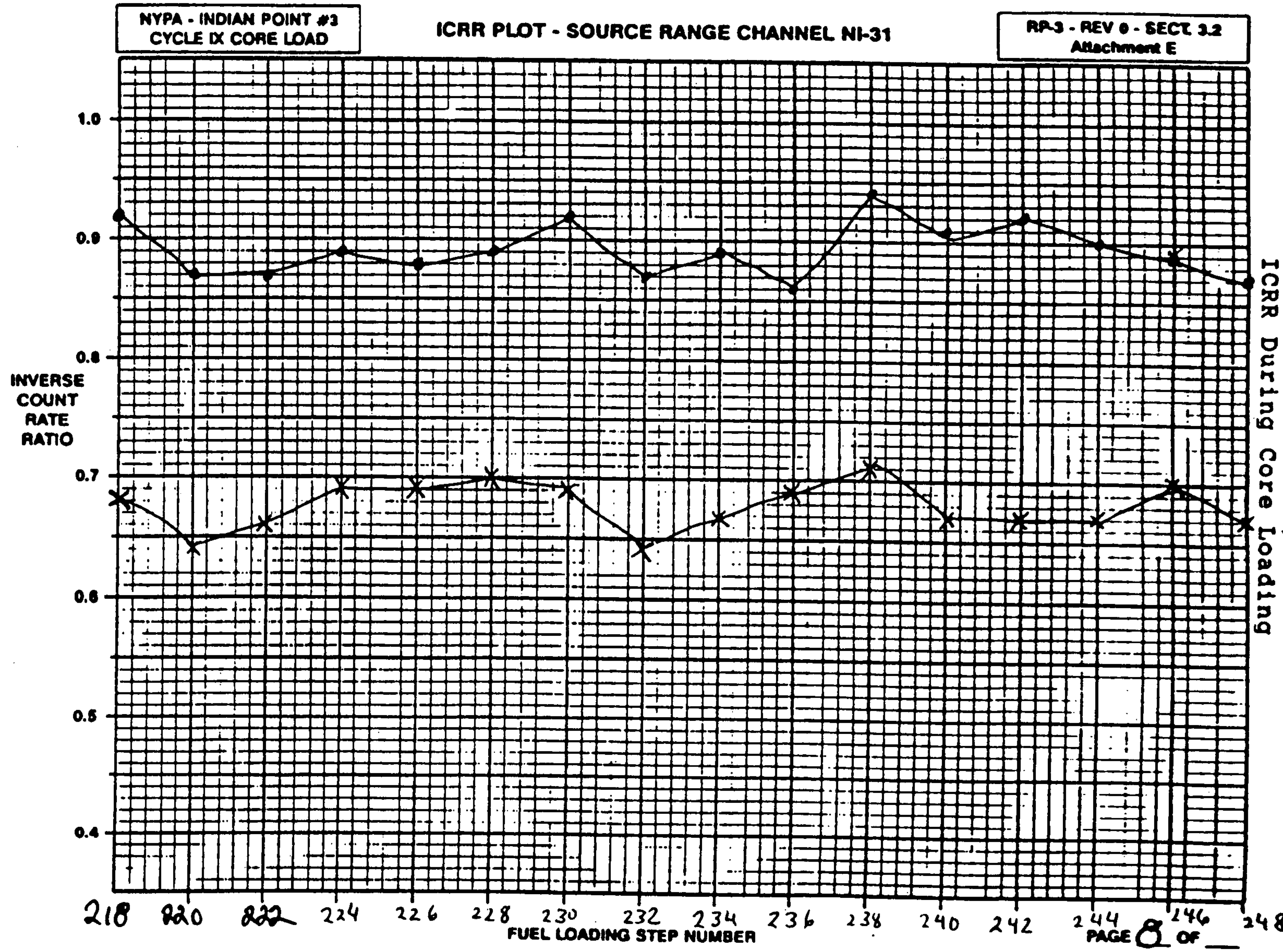
Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 7 of 13)
ICRR During Core Loading



7 OF

• N31
× N32

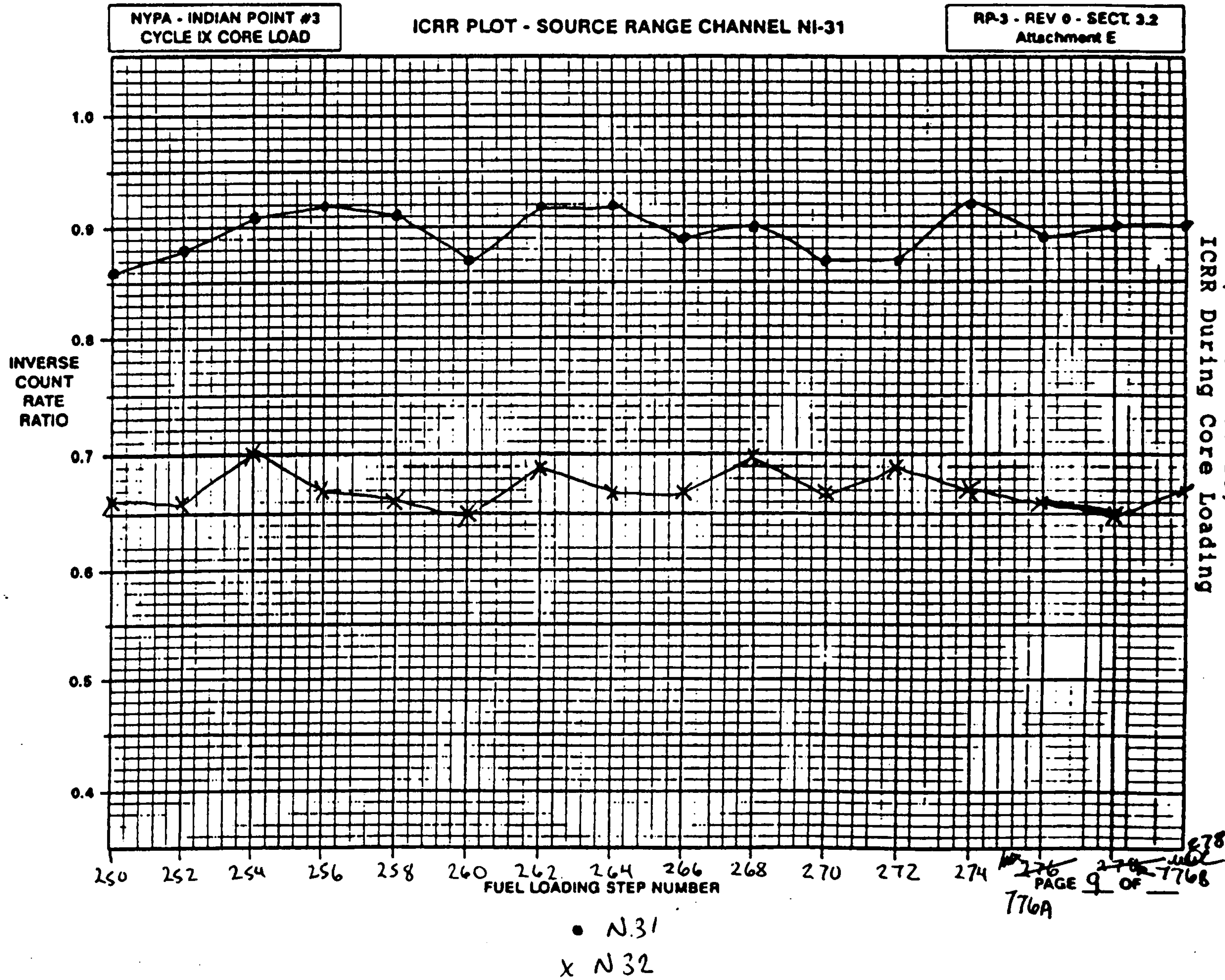


Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 8 of 13)
ICRR During Core Loading

Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 9 of 13)
ICRR During Core Loading



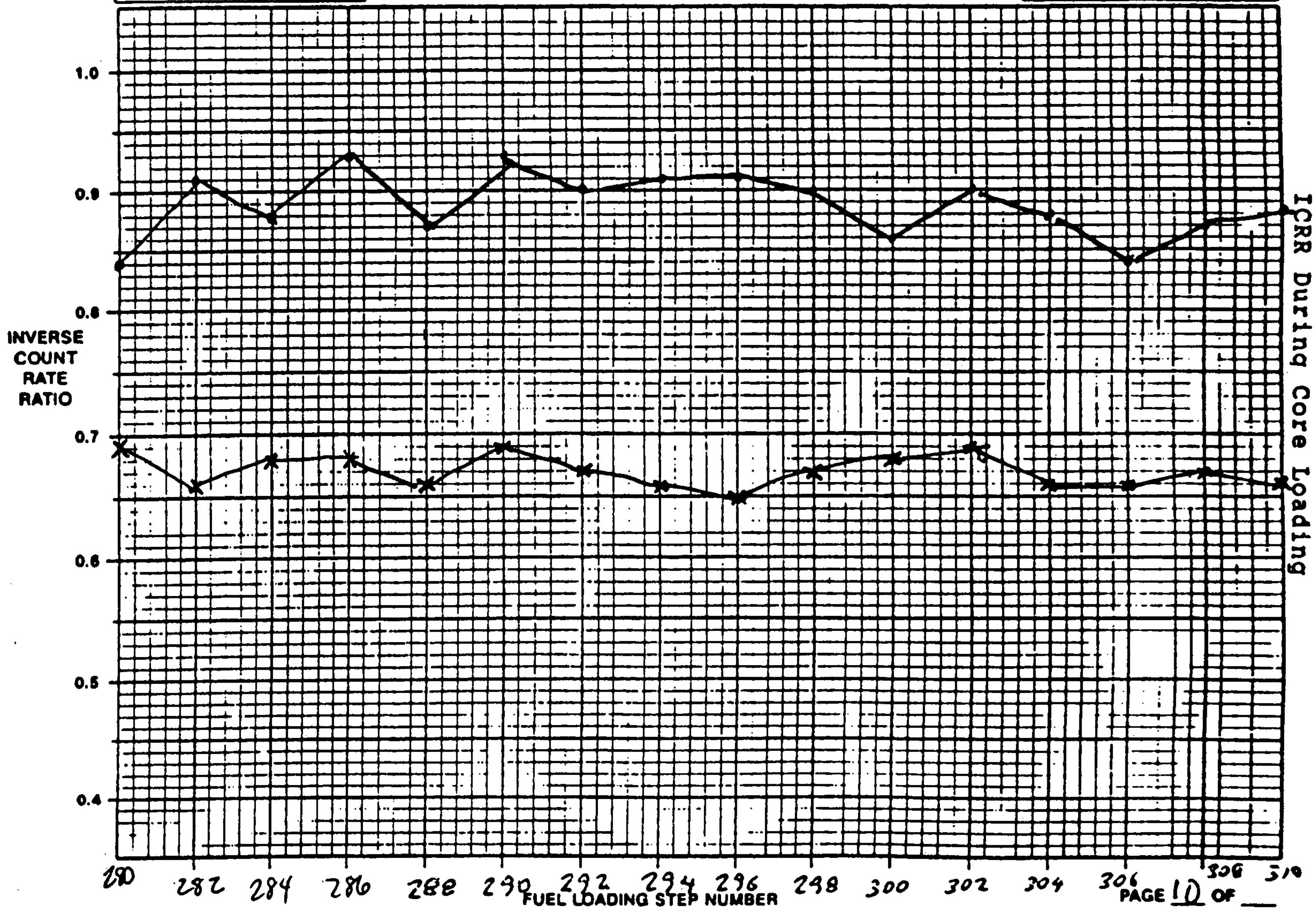
Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 10 of 13)
ICRR During Core Loading

NYPA - INDIAN POINT #3
CYCLE IX CORE LOAD

ICRR PLOT - SOURCE RANGE CHANNEL NI-31

RP-3 - REV 0 - SECT. 3.2
Attachment E



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• N31
x N32

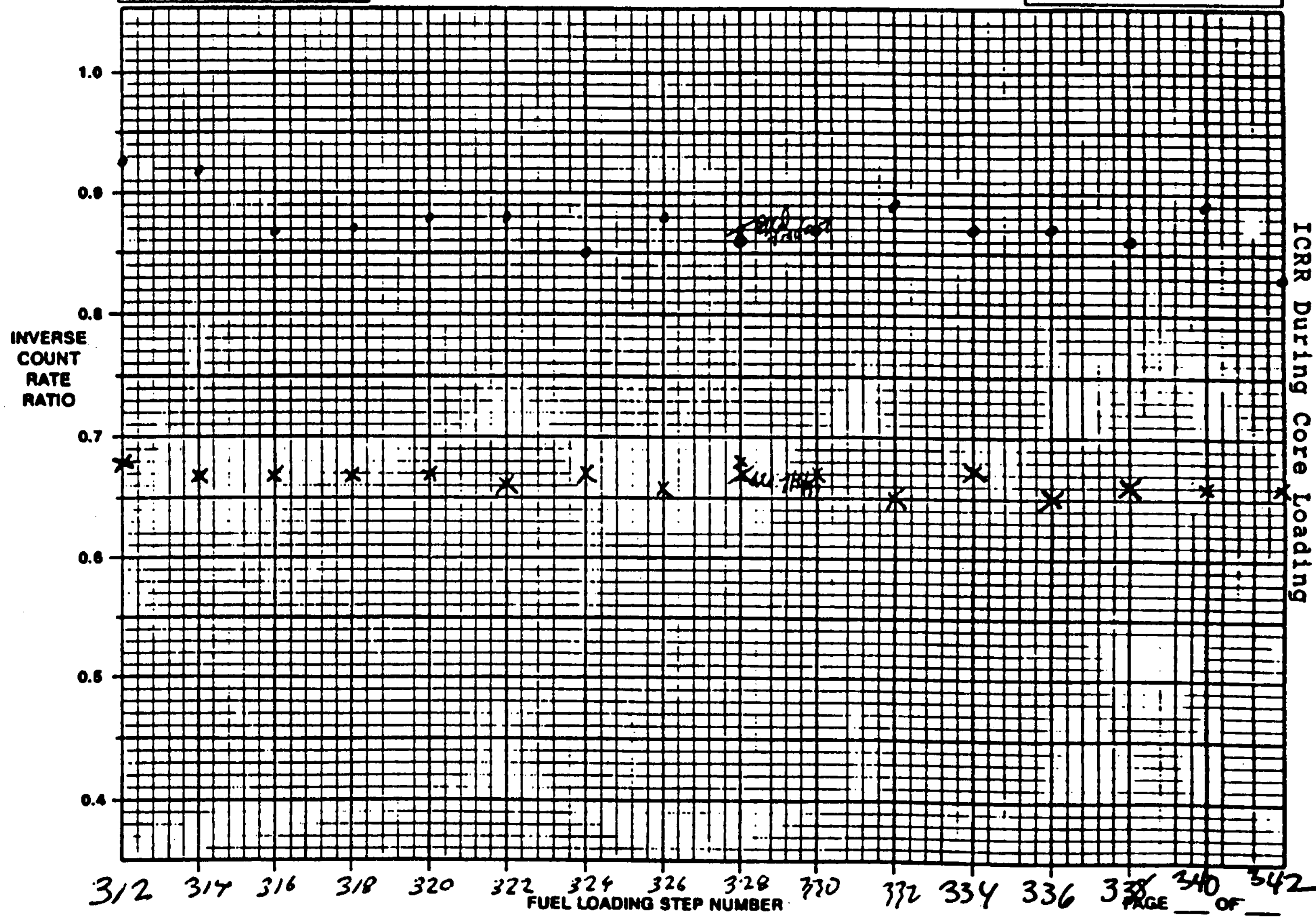
Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 11 of 13)
ICRR During Core Loading

NYPA - INDIAN POINT #3
CYCLE IX CORE LOAD

ICRR PLOT - SOURCE RANGE CHANNEL NI-31

RP-3 - REV 0 - SECT. 3.2
Attachment E



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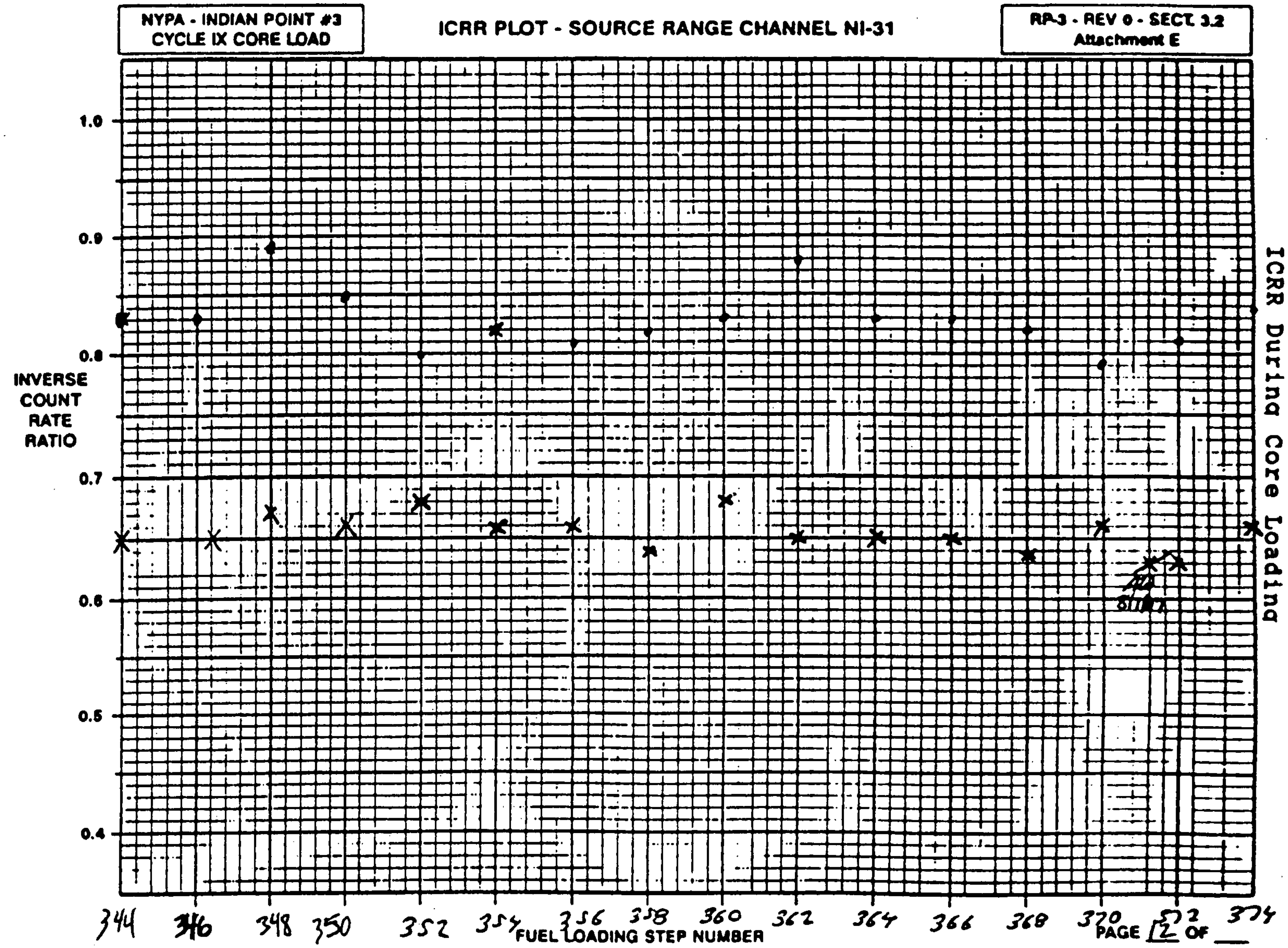
• = N31
x = N32

Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 12 of 13)
ICRR During Core Loading



• N31
x N32

Attachment E - Inverse Count Rate Ratio Plot (ICRR)

Figure 3.1
(Sheet 13 of 13)
ICRR During Core Loading

NYPA - INDIAN POINT #3
CYCLE IX CORE LOAD

ICRR PLOT - SOURCE RANGE CHANNEL NI-31

RP-3 - REV 0 - SECT. 3.2
Attachment E

PAGE ___ OF ___

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ATTACHMENT 7
ICRR PLOT FOR SHUTDOWN BANK WITHDRAWAL

Page 1 of 1

X = 31
O = 32

Date: 9/6/97 Time: 1306

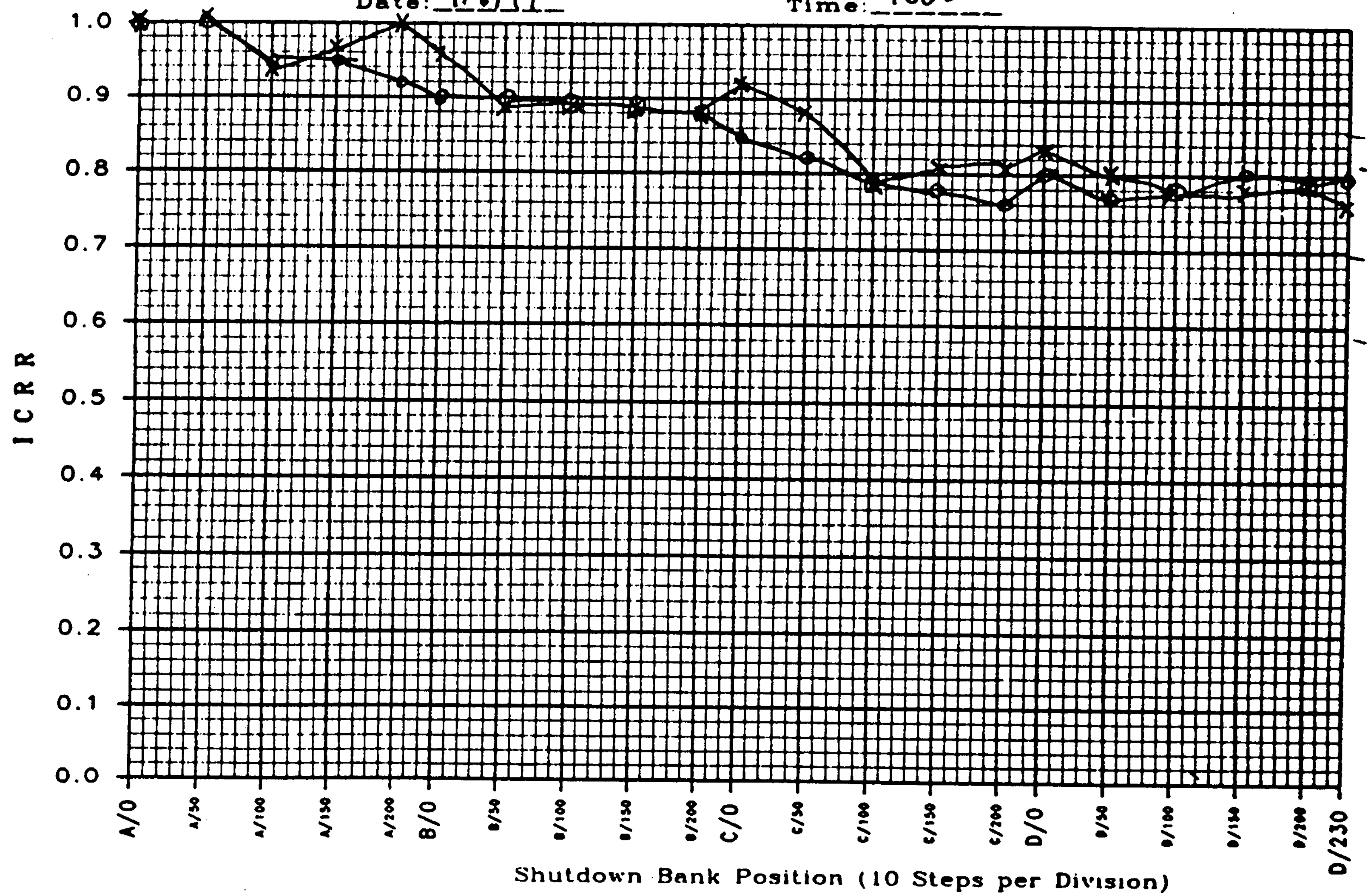
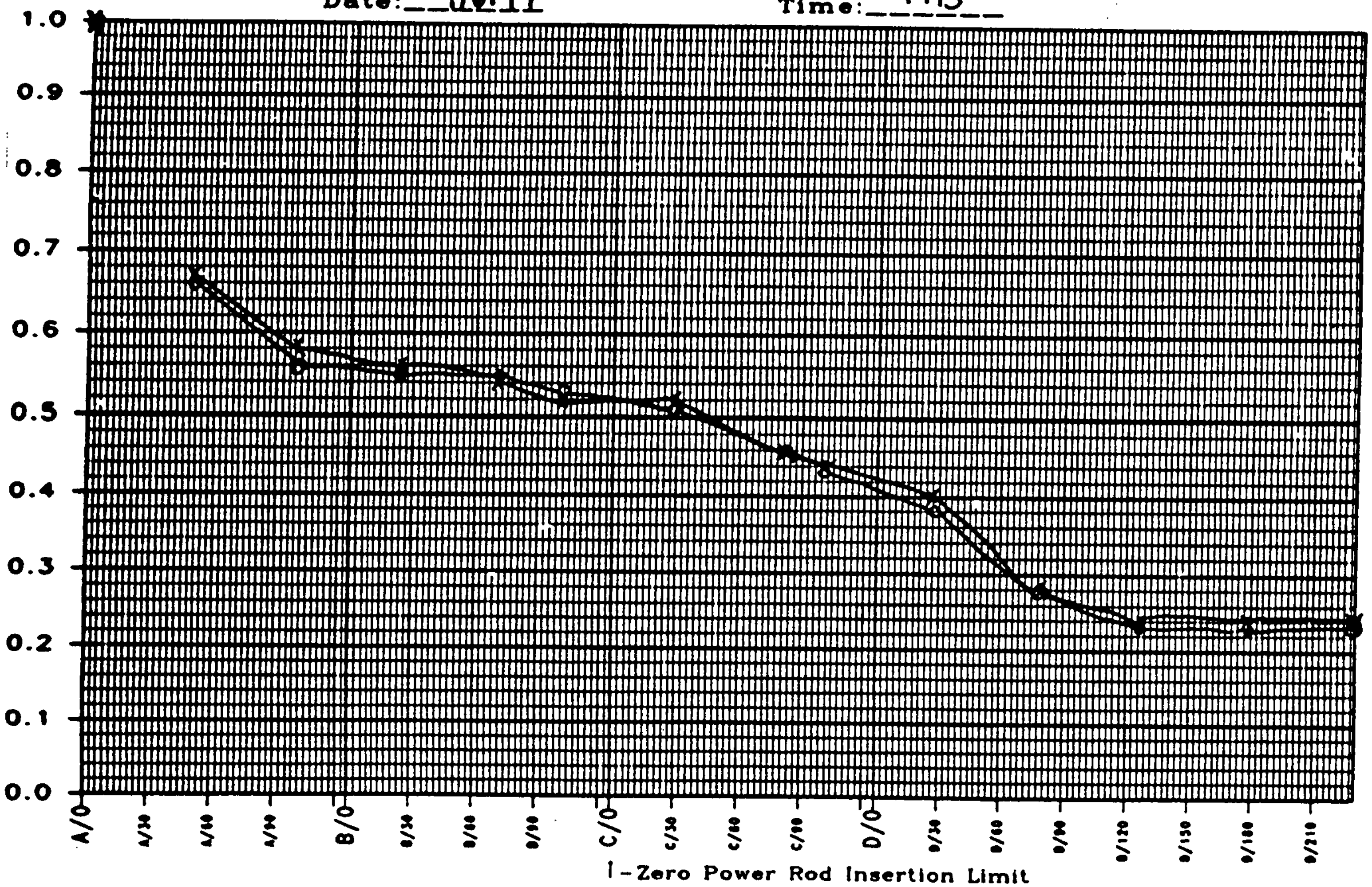


Figure 3.2
(Sheet 1 of 2)
ICRR vs. Bank Position

X = N31
O = N32

ATTACHMENT 9
ICRR PLOT FOR CONTROL BANK WITHDRAWAL

Date: 9/6/97 Time: 1715



Control Bank Position (3 Steps per Division)

Figure 3.2
(Sheet 2 of 2)
ICRR vs. Bank Position

N31 - •
N32 - X

ATTACHMENT 12
PLOT OF ICRR vs PMW ADDITION

Date: 9/6/97 Time: 2345

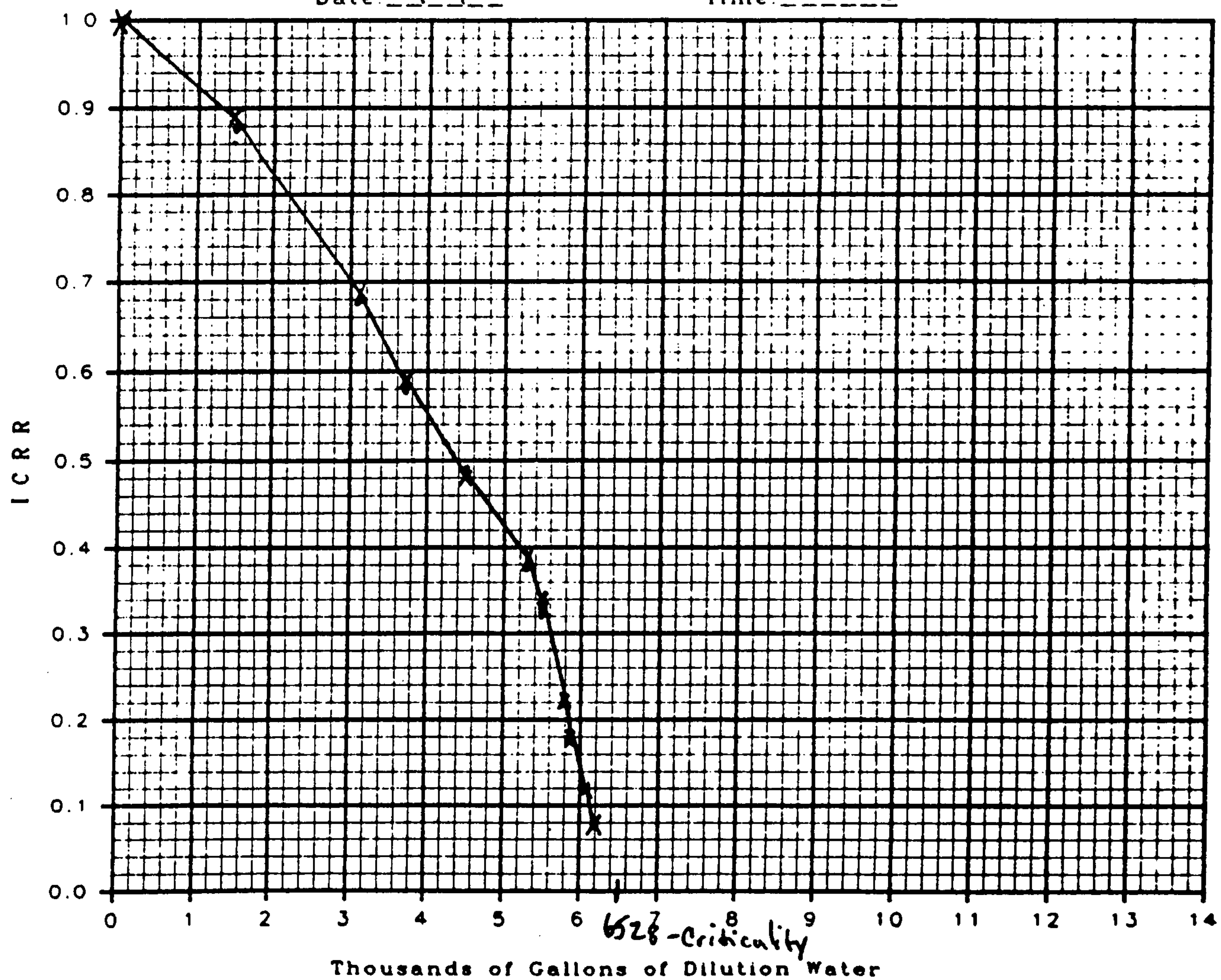


Figure 3.3
ICRR vs. Gallons Dilution Water

4.0 Instrument Measurements Calibrations

4.1 Incore Thermocouple, Wide Range RTD 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 (374, 402, 452, 503, and 549° 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

One full-core map and 6 quarter core maps were taken at approximately 89% power to obtain calibration data for the excore instrumentation. These maps covered a range in axial offset from -8.57% to +0.77 generated by insertion of control bank D. INCORE 3D analysis provided a measured value of the excore calibration.

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 ninety percent power, an extrapolation of full power values was calculated. The extrapolated full power values were used to recalibrate the overpower and overtemperature reactor protection setpoint.

Extrapolated Full Power Temperatures

Delta T (°F)		Tavg (°F)
Loop 31	53.2	567.0
Loop 32	52.1	566.2
Loop 33	52.4	566.8
Loop 34	52.8	565.8

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 10 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.