Crystal River Unit 3 Nuclear Generating Plant

CYCLE 2

STARTUP REPORT

OCTOBER 1979

Florida Power
CORPORATION

POOR ORIGINAL

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FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

LICENSE NO. DPR-72

CYCLE 2

STARTUP REPORT

APPROVED BY: Nuclear Plant Manager

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Date 10/17/79

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

On April 23, 1979, Crystal River Unit 3 was shutdown for scheduled refueling and maintenance at an EOC-1 exposure of 440.1 EFPD. Fifty-six Batch 1 assemblies were removed from the core, Batch 2 and Batch 3 assemblies were shuffled in the inner two-thirds of the core, and fifty-six fresh Batch 4 assemblies were loaded on the core periphery. The new Cycle 2 core loading pattern is shown in Figure 1.1-1.

This report, prepared and submitted in accordance with Technical Specification 6.9.1, describes the precriticality, zero power, and power testing performed and the results obtained.

1.2 SUMMARY

Crystal River Unit 3 was brought to Hot Standby condition. Control Rod Trip Testing, Reactor Coolant Flow and Flow Coastdown Testing, and Source Range Testing were conducted. The test methods used and results achieved are reported in Section 2.0.

Criticality was achieved on July 29, 1979, at 1743 and Zero Power Physics Testing was performed. Results for Zero Power Physics Tests are reported in Section 3.0.

Zero Power Physics Testing was followed by Power Escalation Testing. The generator was synchronized on July 31, 1979 at 2251 and Power Testing proceeded normally. The test method used and results achieved are reported in Section 4.0.

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Figure 1.1-1 Core Loading Diagram for Crystal River 3, Cycle 2

XX Cycle 1 location
Y Batch Number

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						4	4	4	4	4					
B				4	4	4	B7 2	H5 2	B9 2	4	4	4			
C			4	4	F7 2	C6 2	M2 3	C8 2	M14 3	C10 2	F9 2	4	4		
D		4	4	H5 2	L2 3	N2 3	N3 3	R8 3	N13 3	N14 3	L14 3	F8 2	[illegible]	4	
E		4	G6 2	B10 3	O13 3	E6 2	L1 3	D9 2	L15 3	E10 2	O3 3	B6 3	[illegible] 2	4	
F	4	4	F3 2	B12 3	F5 2	N11 2	K1 3	P8 3	K15 3	N4 2	F11 2	B4 3	F13 2	4	4
G	4	G2 2	B11 3	C12 3	A10 3	A9 3	D7 2	G8 2	G12 2	A7 3	A6 3	C4 3	B5 3	G14 2	4
H	4	M12 2	B3 2	H15 3	G4 2	K14 3	H7 2	H8 2	H9 2	H2 3	K12 2	H1 3	B13 2	E4 2	4
K	4	K2 2	P11 3	O12 2	R10 3	R9 3	K4 2	K8 2	H9 2	R7 3	R6 3	O4 3	P5 3	K14 2	4
L	4	4	L3 2	P12 3	L5 2	R12 2	G1 3	B8 3	G15 3	D5 2	L11 2	P4 3	L13 2	4	4
M		4	E6 2	P10 3	C13 3	N6 2	F1 3	H7 2	F15 3	K10 2	C3 3	P6 3	K10 2	4	
N		4	4	N8 2	F2 3	D2 3	D3 3	A8 3	D13 3	D14 3	F14 3	R11 2	4	4	
O			4	4	L7 2	O6 2	E2 3	O8 2	E14 3	O10 2	L9 2	4	4		
P				4	4	4	P7 2	D11 2	P9 2	4	4	4			
R						4	4	4	4	4					

2.0 PRECRITICALITY TESTING

2.1 REACTOR COOLANT FLOW AND FLOW COASTDOWN TEST

Prior to reactor criticality, the reactor coolant pump flow and flow coastdown test was performed to ensure the functional capabilities of the reactor coolant system and the reactor coolant pumps during steady state flow and flow coastdown conditions. This test was conducted at hot conditions of 532°F and 2155 psig with the core installed.

Normally, this test is limited to measuring the normal flow rate with four reactor coolant pumps operating and the flow coastdown and time delay characteristics for the loss of four reactor coolant pumps from the normal operating condition. The Cycle 2 testing included the above as well as the flow coastdown and time delay characteristics for the loss of one reactor coolant pump from normal operating conditions. Also included in the flow coastdown testing was the measurement of the signal delay times due to the snubbers present in the flow monitors. Measurements were taken for both the one-pump and the four-pump coastdowns.

2.1.1 PURPOSE

The purpose of the reactor coolant flow and flow coastdown test are listed below.

A. To measure the reactor coolant flow at 532°F and 2155 psig with four reactor coolant pumps operating and the core installed.

B. To compare reactor coolant flow with design requirements and to verify adequate core flow.

C. To measure reactor coolant flow coastdown characteristics with four reactor coolant pumps operating when all four pumps are tripped.

D. To measure reactor coolant flow coastdown characteristics with four reactor coolant pumps operating when only the pump with the highest flow rate is tripped.

E. To verify that the delay time associated with the snubbers in the flow monitors is within acceptable limit

Four acceptance criteria are specified for the reactor coolant flow and flow coastdown test as listed below:

1. Steady state total reactor coolant system flow with the core installed, when corrected to 532°F and 2155 psig shall be within the limits specified in Table 2.1.1.

2. With all four reactor coolant pumps in operation, steady state loop flows shall be within 2% of each other.

3. With the reactor core installed, the delay in the reactor coolant flow signal must be less than or equal to one second during the coastdown interval prior to reaching the applicable flux-flow trip ratio (92.7% of normal flow for one-pump coastdown, 95.1% of normal flow for four-pump coastdown).

4. The reactor coolant system flow must be greater than or equal to the flow versus time relationships in Figures 2.1-1 or 2.1-2, for one-pump coastdown or four-pump coastdown, respectively.

2.1.2 TEST METHOD

Reactor coolant steady state flows were determined by means of the temperature compensated loop flow transmitter voltages input to a Babcock & Wilcox supplied reactimeter and converted to loop flows in millions of pounds per hour (MPPH).

With four reactor coolant pumps operating, ten sets of steady state coolant flow data were taken on the reactimeter and the values averaged. These results were then corrected to reference conditions of 532°F and 2155 psig using the equation below:

$$FR = FM\ (Vm/Vr)$$

Where: Fr = Flow Rate at Reference Conditions, MPPH
Fm = Flow Rate at Measured Conditions, MPPH
Vr = Specific Volume at Reference Conditions, $Ft^3/lb.$
Vm = Specific Volume at Measured Conditions, $Ft^3/lb.$

The measured flow rate corrected to the reference conditions was then compared to the respective acceptance criteria.

Reactor coolant flow coastdown versus time was also determined from the temperature compensated loop flow transmitters. Steady state data was obtained with four reactor coolant pumps in operation. Subsequently, all of the pumps were tripped and data was recorded during the ensuing reactor coolant flow transient. The measured reactor coolant flow at various times during the coastdown transient was then plotted versus time and compared to the acceptance criteria. This process was repeated for the reactor coolant pump with the highest flow rate being tripped.

The snubbered reactor coolant flow delay time was determined for both the one-pump and the four-pump coastdown by leaving one of the channels unsnubbered and measuring the delay time between the snubbered and unsnubbered signals at the flux-flow trip setpoint. The delay times obtained were then compared to the acceptance criterion ($\leq$1 second).

2.1.3 EVALUATION OF TEST RESULTS

Table 2.1-1 gives the minimum and maximum allowable flow rate along with the measured flow rate for four reactor coolant pump operation. It can be seen that the measured flow rate is well within the acceptance criteria, and that the flow imbalance with four reactor pumps was also well within the acceptance criteria of 2 percent.

Table 2.1-2 gives the times at which the snubbered flow transmitters registered 92.7% of the initial flow rate after the reactor coolant pump was tripped. The snubbered flow signals were within 1.0 second of the unsnubbered signal, thereby satisfying the acceptance criterion. Table 2.1-3 gives the times at which the snubbered flow transmitters registered 95.1% of the initial flow rate after all four reactor coolant pumps were tripped. All of the snubbered flow signals were within the 1.0 second acceptance criterion.

From the one- and four-pump coastdown data, those channels which registered the most rapid decrease of flow with time were used to create plots of flow versus time which were then compared to acceptance criteria curves. By utilizing those channels with the steepest "flow versus time" slope, the most conservative data was used. Figures 2.1-1 and 2.1-2 show the results of the flow coastdown test and comparisons with the acceptance criteria curves. All acceptance criteria were met.

2.2 CONTROL ROD DRIVE DROP TIME TEST

2.2.1 PURPOSE

The purpose of the Control Rod Drive Drop Time test was to verify the functional trip capability of the Control Rod Drive System. This was done by verifying the following acceptance criterion:

> To verify, for each control rod assembly, that the total elapsed drop time from the initiation of the trip signal until the control rod assembly was three-fourths inserted, was less than or equal to 1.66 seconds with $T_{avg} \geq 525°F$ and all reactor coolant pumps operating.

2.2.2 TEST METHOD

The Control Rod Drive Drop Time Test was performed using strip chart recorders to time the rod drops. Each control rod group was pulled to 100% withdrawn and then dropped into the core using the manual trip pushbutton. A zero time signal was furnished to the test recorders for each control rod assembly from a contact on the manual trip switch. A second signal to indicate three-fourths insertion was furnished to the recorders by a reed switch located on the position indicator tube of each control rod drive. The test was conducted at the following conditions:

Flow	Temperature	Mode
Four Pumps	>525°F	Hot Standby

2.2.3 EVALUATION OF TEST RESULTS

The results of the rod drop times are presented in Tables 2.2-1 and 2.2-2. The rods with the shortest drop time were CR5-7 and CR7-1, each with a rod drop time of 1.173 seconds. The rod with the longest drop time was CR1-3 with a rod drop time of 1.240 seconds. All of the control rods met the acceptance criterion.

2.3 CHEMICAL AND RADIOCHEMICAL TESTS

Chemical and Radiochemical testing was not performed during this startup. These tests were conducted at initial startup and no plant modifications have been made which would invalidate the results of those tests.

2.4 PRESSURIZER EFFECTIVENESS

No pressurizer effectiveness testing was done during this startup since no modifications had been made to the pressurizer and results of the testing done at initial startup are still valid.

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2.5 CALIBRATION AND NEUTRON RESPONSE OF SOURCE RANGE MONITORING

Precritical testing was completed following response checkout of the neutron source range detectors. The source range instrumentation was verified to provide a count rate of more than 2 counts/sec. A statistical test was performed to determine that the detector response data followed a normal distribution.

2.6 IN-SERVICE LOOSE PARTS AND VIBRATION MONITORING SYSTEM

No loose parts and vibration monitoring system testing was done during this startup since no modification had been made or circumstances encountered which would invalidate the test results from the mid-Cycle 1 startup testing.

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Table 2.1-1

Comparison of Measured Reactor Coolant Flow Rate To The
Acceptance Criteria At Reference Conditions Of 532°F and 2155 psig

Pump Combination	Minimum Acceptable Flow Rate (gpm)	Maximum Acceptable Flow Rate (gpm)	Measured Flow Rate (gpm)	Maximum Acceptable Loop Flow Difference (%)	Measured RC Loop Flow Difference (%)
Four Pumps	374,880	411,800	399,631	2.0	0.27

DELAY TIME OF THE SNUBBERED FLOW
SIGNALS FOR THE ONE REACTOR COOLANT PUMP COASTDOWN

	Snubbered Flow Signal A	Snubbered Flow Signal B
Time to Reach 92.7% of Initial Flow Rate (sec.)	1.95	2.35
Time Difference Between Snubbered and Unsnubbered Flow Signals at 92.7% Flow Rate (sec.)	0.55	0.95
Time Difference $\leq$ 1.0 sec.	Yes	Yes

Table 2.1-2

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Table 2.1-3

DELAY TIME OF THE SNUBBERED FLOW
SIGNALS FOR THE FOUR REACTOR COOLANT PUMP COASTDOWN

	Coolant Loop	Snubbered Flow Signal A	Snubbered Flow Signal B	Snubbered Flow Signal C
Time to Reach 95.1% of Initial Flow Rate (sec.)	A	0.87	1.65	1.02
Time Difference Between Snubbered and Unsnubbered Flow Signals at 95.1% Flow Rate (sec.)	A	0.17	1.00	0.35
Time Difference $\leq$ 1.0 sec.	A	Yes	Yes	Yes
Time to Reach 95.1% of Initial Flow Rate (sec.)	B	0.90	1.07	-- *
Time Difference Between Snubbered and Unsnubbered Flow Signals at 95.1% Flow Rate (sec.)	B	0.47	0.62	-- *
Time Difference $\leq$ 1.0 sec.	B	Yes	Yes	-- *

* Bad Signal

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CONTROL ROD DRIVE DROP TIME
TEST RESULTS

Control Rod	Core Position	Drop Time (Sec.)
CR1-1	C-11	1.220
CR1-2	E-13	1.220
CR1-3	M-13	1.240
CR1-4	O-11	1.217
CR1-5	O-5	1.214
CR1-6	M-3	1.217
CR1-7	E-3	1.237
CR1-8	C-5	1.225
CR2-1	E-9	1.207
CR2-2	G-11	1.203
CR2-3	K-11	1.200
CR2-4	M-9	1.215
CR2-5	M-7	1.205
CR2-6	K-5	1.195
CR2-7	G-5	1.197
CR2-8	E-7	1.193
CR3-1	D-8	1.200
CR3-2	G-9	1.198
CR3-3	E-11	1.196
CR3-4	H-12	1.187
CR3-5	K-9	1.207
CR3-6	M-11	1.198
CR3-7	N-8	1.197
CR3-8	K-7	1.190
CR3-9	M-5	1.190
CR3-10	H-4	1.200
CR3-11	G-7	1.193
CR3-12	E-5	1.200
CR4-1	H-8	1.204
CR4-2	B-10	1.185
CR4-3	F-14	1.194
CR4-4	L-14	1.205
CR4-5	P-10	1.200
CR4-6	P-6	1.202
CR4-7	L-2	1.199
CR4-8	F-2	1.204
CR4-9	B-6	

Table 2.2-1

CONTROL ROD DRIVE DROP TIME
TEST RESULTS

Control Rod	Core Position	Drop Time (Sec.)
CR5-1	C-9	1.185
CR5-2	G-13	1.177
CR5-3	K-13	1.192
CR5-4	O-9	1.178
CR5-5	O-7	1.177
CR5-6	K-3	1.177
CR5-7	G-3	1.173
CR5-8	C-7	1.185
CR6-1	F-8	1.203
CR6-2	D-12	1.200
CR6-3	H-10	1.200
CR6-4	N-12	1.195
CR6-5	L-8	1.195
CR6-6	N-4	1.187
CR6-7	H-6	1.200
CR6-8	D-4	1.205
CR7-1	B-8	1.173
CR7-2	F-10	1.185
CR7-3	H-14	1.192
CR7-4	L-10	1.185
CR7-5	P-8	1.185
CR7-6	L-61	1.182
CR7-7	H-2	1.180
CR7-8	F-6	1.181

Table 2.2-1 (Cont'd)

GROUP AVERAGE
CONTROL ROD DRIVE DROP TIME

Group	Average Drop Time (Sec.)
1	1.224
2	1.202
3	1.196
4	1.198
5	1.198
6	1.181
7	1.183
1-7	1.197

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Table 2.2-2

ONE PUMP COASTDOWN
FLOW VS. TIME

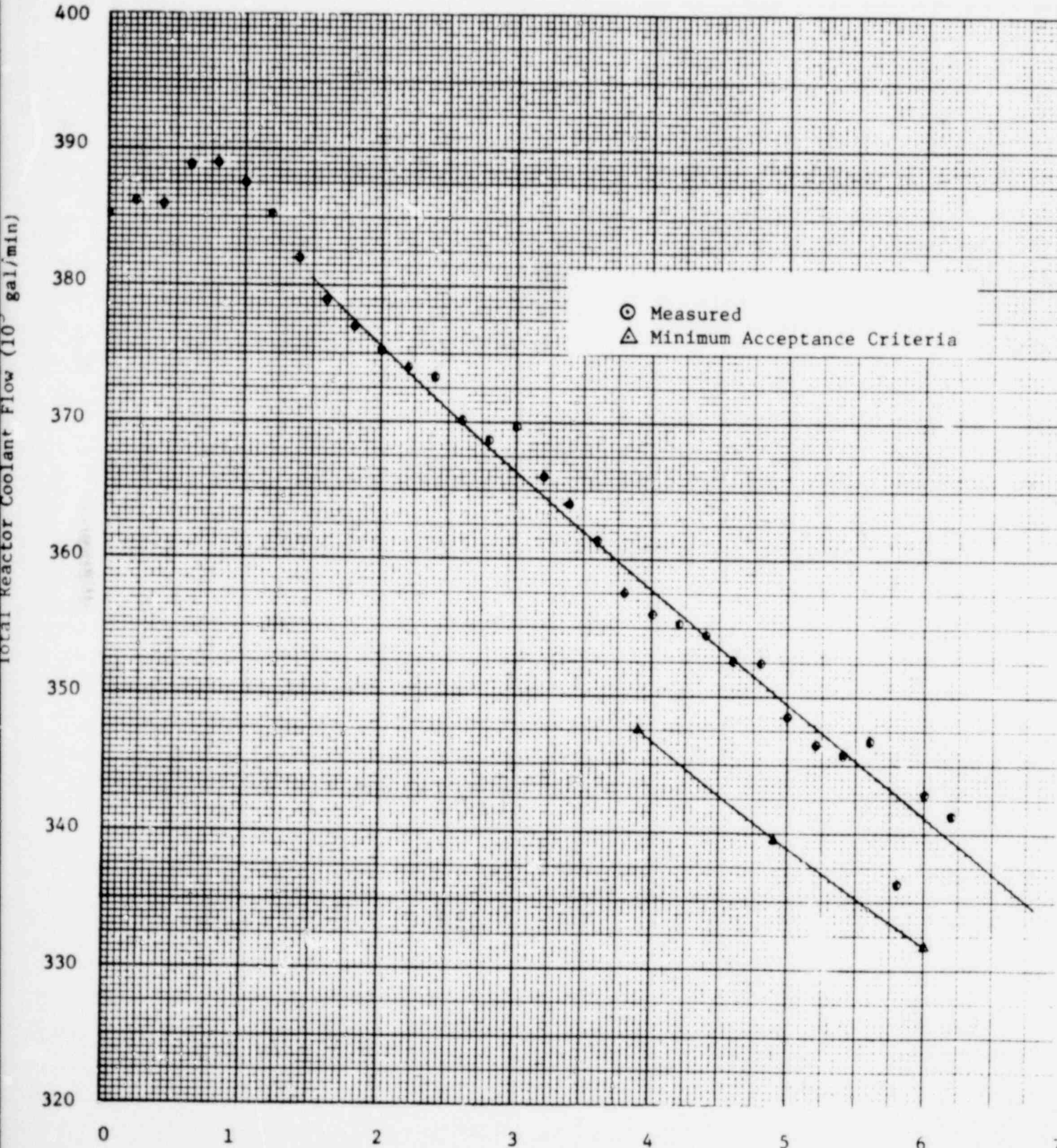

Figure 2.1-1

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FOUR PUMP COASTDOWN
FLOW VS. TIME

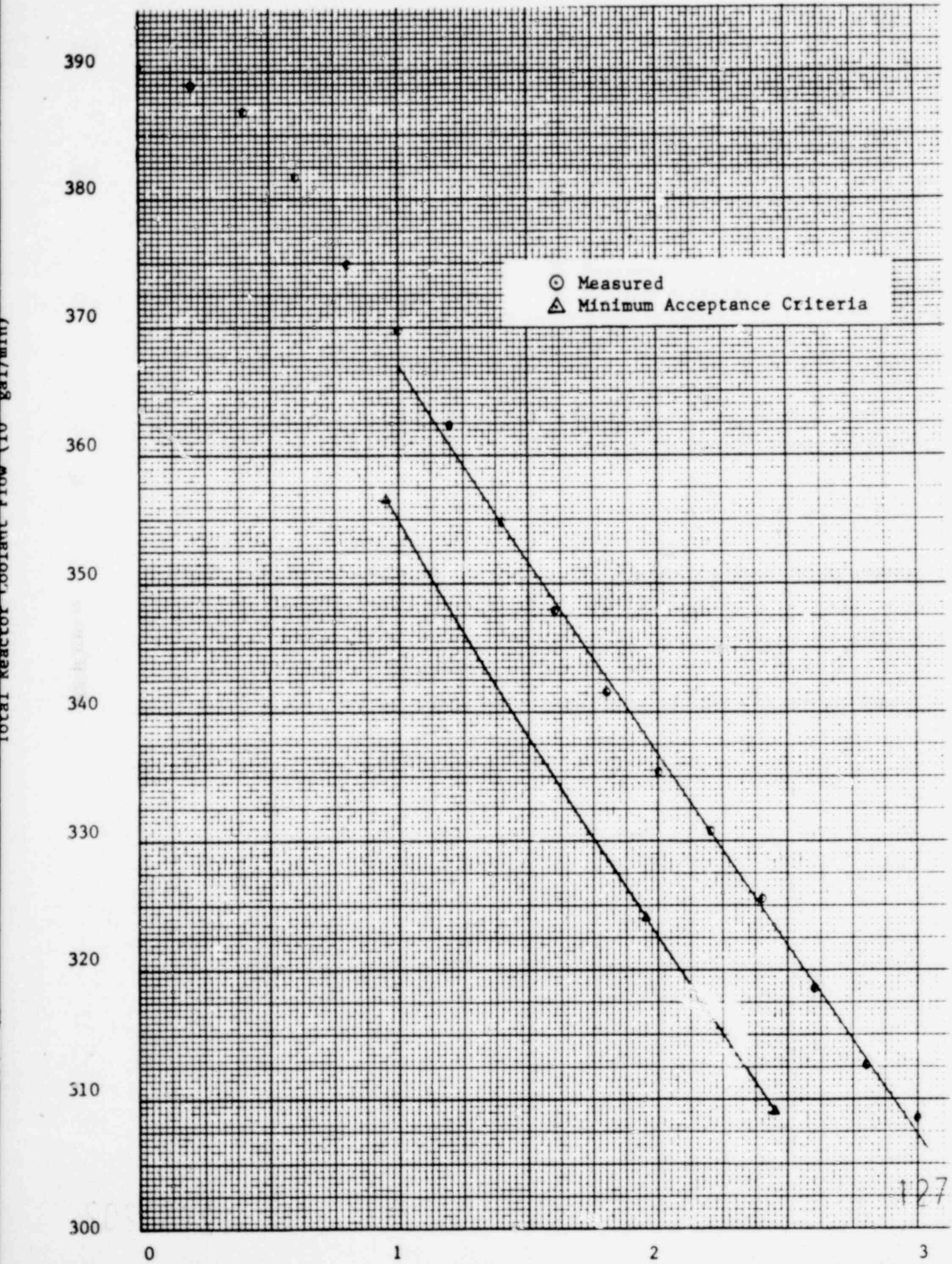

Figure 2.1-2

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3.0 ZERO POWER PHYSICS TESTS

The Zero Power Physics Tests were performed to verify the nuclear design parameters used in the safety analysis, the Technical Specification limits, and the operational parameters. Acceptance criteria for the tests are given in Table 3.0-1. A summary of measured and predicted results obtained during zero power physics testing is given in Table 3.0-2.

3.1 INITIAL CRITICALITY

Initial criticality was achieved on July 29, 1979 at 1743 hours at reactor coolant conditions of 532°F, 2155 psig, and 1241 ppmB. Control rod groups 1 through 4 had previously been withdrawn as part of the heatup procedure with reactor coolant boron concentration at 1483 ppmB. The approach to critical began by withdrawing control rod group 8 to 37.5% withdrawn, groups 5 and 6 to 100% and group 7 to 90% withdrawn. The reactor coolant system was then deborated until criticality was achieved.

Throughout the approach to criticality, inverse multiplication curves versus time, reactor coolant system boron concentration, and quantity of demineralized water added were maintained for the source range detectors by two independent persons. At the end of each control rod group withdrawal and every 30 minutes during the deboration cycle, the count rate was taken from each source range detector by way of the scaler-counters. The ratio of the initial average count rate to the count rate at the end of each reactivity addition was plotted and the criticality point determined by extrapolation.

3.2 NUCLEAR INSTRUMENTATION OVERLAP

Technical Specifications state that prior to operation in the intermediate nuclear instrumentation range, at least a one decade overlap between the source range and intermediate range must be observed. This means that before the source range count rate equals 10^5 cps the intermediate range must be on the scale. If the one decade is not observed, the approach to the intermediate range cannot be continued until the situation has been corrected.

To satisfy the above overlap requirements after initial criticality was reached, core power was slowly increased until the intermediate range channels came on scale. Detector signal response was thus recorded for both the intermediate and source range channel. This was repeated for another decade to assist in determining the number of decades of overlap.

The results of the Nuclear Instrumentation overlap at 532°F and 2155 psig have been tabulated in Table 3.2-1. Examination of Table 3.2-1 shows that the overlap between the source and intermediate range is constant at an average value of 2.10 decades which is well above the minimum of one decade overlap stated in Technical Specifications.

3.3 SENSIBLE HEAT DETERMINATION

Determination of the intermediate range current level at which the production of sensible nuclear heat occurs is important to the Zero Power Physics Test program in that it establishes the upper level limit. Thus, by restricting reactor power

operation to a level reduced by a factor of three below the sensible heat level, the effects of temperature feedback are eliminated in the measurement of physics parameters.

The test method was to increase the intermediate range current level in one third decade increments until the detection of sensible heat. Since turbine bypass valves and pressurizer levels were maintained constant, heat production in the core was observed by change in the core outlet temperature and makeup tank level.

The point at which there was a definite heatup rate was 1.70 X 10^{-7} amps. Therefore, during the test 1.70 X^{-7} amps on channel NI-3 was defined as the "sensible heat" point. From this the upper zero power physics test current limit was established at 5.00 X 10^{-8} amps.

After setting the upper zero power physics test current limit, temperature feedback measurements were performed to verify that the above limit was adequate. This was done by adding +20 pcm at a neutron flux signal of 5.0 X 10^{-8} and verifying that the reactivity remained constant. The results indicated that no temperature feedback effects were present at the upper zero power physics test current limit.

An additional part of the measurement was to ensure that all power range channels were indicating between 0.8 to 2.0% full power at the point of sensible heat. This was done to ensure that the overpower trip protection was adequate. At the point of sensible heat, the indication on NI-5, NI-6, NI-7 and NI-8 were 2.1, 1.9, 1.9 and 1.1% full power. Thus, all power range channels except NI-5 were within their recommended tolerances. Since all channels were set near their maximum sensitivity, no calibration was performed.

3.4 REACTIVITY CALCULATIONS

Reactimeter is the name given to the Babcock and Wilcox reactivity computer which solves the one-dimensional, inverse kinetics equation with six delayed neutron groups for core net reactivity based upon periodic samples of neutron flux. In addition to reactivity and neutron flux, the reactimeter can also record 23 other analog and digital signals from the plant.

After initial criticality and prior to the first physics measurements, an on-line functional check of the reactimeter was performed to verify its readiness for use in the test program. After steady state conditions with a constant neutron flux were established, a small amount of negative reactivity was inserted in the core by inserting control rod group 7. Stop watches were used to measure the doubling time of the neutron flux and the reactivity inserted was determined from period-reactivity curves. The measurement was repeated for several values of reactivity inserted by rod group 7, from $\pm$0.032 to $\pm$0.099% $\Delta k/k$. The reactivities determined from doubling time measurements were then compared with the reactivities calculated by the reactimeter.

The results of the reactimeter verification measurements are summarized in Table 3.4-1. In each case the reactivity calculated by the reactimeter was within the acceptance criteria limit of $\pm$5% of the reactivity determined from doubling times.

3.4.1 "ALL RODS OUT" CRITICAL BORON CONCENTRATION

The "all rods out" critical boron concentration was measured at one isothermal temperature test plateau of 532°F. The measurements actually were made with rod group 7 partially inserted, but the measured boron concentration was adjusted to the all rods out condition using the results of rod worth measurement to determine the reactivity worth, in terms of ppm boron, of the inserted control rods.

The results are tabulated in Table 3.4-2. These results show that the measured critical boron concentration was within the acceptance criterion of 1260 ±100 ppmB. The results were also within the review criterion of 1260 ±50 ppmB.

3.5 CONTROL ROD GROUP WORTHS

The layout of the core according to the standard alpha-numeric mesh showing the initial location of the control rod groups and the location of the 52 incore detector strings is given in Figure 3.5-1. The number of rods in each group and the reactivity control function of each group is listed below.

Rod Group No.	No. of Rods	Control Function
1	8	Safety
2	8	Safety
3	12	Safety
4	9	Safety
5	8	Power Doppler
6	8	Power Doppler
7	8	Transient
8	8	Axial Power Shaping
	69	

Predictions of control rod group worths for groups 1-4, 5, 6, and 7 at Hot Zero Power and Hot Full Power conditions were made in the Physics Test Manual. Measurements of control rod group worths for groups 5, 6, and 7 were made during Zero Power Physics Tests using the boron swap method. This method consisted of setting up a deboration rate and compensating for the change in reactivity by small step changes in rod group positions. The calculation of reactivity on a continuous basis is made by the reactimeter. The output of reactivity in terms of percent milli ρ (PCM) is recorded on a strip chart and also on a magnetic tape by the reactimeter.

The results of both the predicted rod group worths and the measured group worths are tabulated in Table 3.5-1 for a moderator temperature and pressure of 532°F and 2155 Psig respectively. Comparison between measured and predicted rod worth shows groups 5-7 agreed within 5.7 percent. Predicted and estimated rod worths at 579°F are also tabulated:

Integ[illegible] and differential measured rod worth curves for control rod groups 5, 6 and 7 at 532°F and 2155 Psig are plotted for group 8 at 37.5% wd in Figures 3.5-2 to 3.5-7.

3.6 SOLUBLE POISON WORTHS

Soluble poison in the form of dissolved boric acid is added to the moderator to provide additional reactivity control beyond that available from the control

rods. The primary function of the soluble poison control system is to control the excess reactivity of the fuel throughout each core life cycle.

Measurement of the soluble poison differential worth has been completed at 532°F and 2155 Psig. The measured value was determined by summing the incremental reactivity values measured during the rod worth measurements over a known boron concentration range from 1106 to 772 ppmB.

The result of the differential soluble poison worth measurement is tabulated in Table 3.6-1.

In summary, the measured differential boron worth at zero power was within 5.4 percent of the predicted worth which is within the acceptance criteria of $\pm$ 10 percent.

3.7 EJECTED CONTROL ROD WORTH

Pseudo ejected control rod reactivity worth was measured at hot zero power conditions of 532°F, and 2155 Psig for control rod 6-4. The purpose of this measurement was to verify the safety analysis calculations relating to the assumed accidental ejection of the most reactive control rod which is fully inserted in the core during normal power operation. The acceptance criteria for the ejected control rod test are that the average measured ejected rod worth be within 20% of the predicted value, and that the worst case measured ejected rod worth be less than 1.0% $\Delta k/k$.

The ejected rod worth measurements were done with control rod groups 6 and 7 fully inserted and group 8 at 37.5% withdrawn. Group 5 was 0% withdrawn for the boron swap measurement and 28% withdrawn for the rod swap measurements. The ejected worth of control rod 6-4 was measured by the boron swap method. Control rod 6-4 and the rods in the quarter core symmetric locations were also measured by the rod swap method. The rod swap test method was to initially position group 5 at 28% withdrawn with the "ejected" control rod 0% withdrawn. The "ejected" rod and group 5 were then swapped until the "ejected" rod was 100% withdrawn. The change in group 5 position was then converted to reactivity by using a previously generated group 5 rod worth curve to determine the measured ejected rod worth.

Since the measured rod position did not precisely equal the minimum rod index, the measured ejected rod worth was adjusted to the Technical Specifications position by using the equation below:

Where: $\rho(ER) = F \times \rho(R) + \rho(M)$ EQ. (3.7.1)

$\rho(ER)$ = The measured ejected rod worth at the maximum allowable inserted group worth.

F = Empirically derived constant of 0.45 which relates ejected rod worth and total inserted group worth.

$\rho(R)$ = The worth of rods at the minimum rod index minus the worth of rods at the final rod configuration.

$\rho(M)$ = The measured worth of the ejected rod which corresponds to the worth changes in group 5.

The measured ejected rod worth was not within the acceptance criterion of $\pm 20\%$ from the predicted value. The Plant Review Committee and Babcock and Wilcox both agreed that the results were acceptable and would have no adverse effect on the safe and efficient operation of the reactor. To ensure that the measured worst case ejected rod worth was conservative, it was multiplied by an uncertainty factor of 1.10. The measured worst case ejected rod worth was within the acceptance criterion of less than 1.0% $\Delta k/k$. The relative ejected rod worth measurements were within the acceptance criterion of $\pm$ 0.05% $\Delta k/k$ of the average worth. The results of the measurements are given in Table 3.7-1.

3.8 TEMPERATURE COEFFICIENTS OF REACTIVITY

The temperature coefficient of reactivity is defined as the fractional change in the excess reactivity of the core per unit change in core temperature. The temperature coefficient is normally divided into two components as shown in Equation (3.8-1).

$$\alpha T = \alpha M + \alpha D \qquad \text{EQ. (3.8-1)}$$

Where: αT = Temperature Coefficient of Reactivity

αM = Moderator Coefficient of Reactivity

αD = Doppler Coefficient of Reactivity

The technique used to measure the 532°F and 2155 psig isothermal temperature coefficient at zero power was to first establish steady state conditions by maintaining reactor flux, reactor coolant pressure, turbine header pressure and core average temperature constant, with the reactor critical between 7 X 10^{-10} amps and the upper zero power physics test current limit set in the sensible heat determination experiment. Equilibrium boron concentration was established in the reactor coolant system, make-up tank and pressurizer to eliminate reactivity effects due to boron changes during the subsequent temperature swings. The reactimeter and the brush recorders were connected to monitor selected core parameters with the reactivity value calculated by the reactimeter and the core average temperature displayed on a two channel recorder.

Once steady state conditions were established, a negative heatup rate was started by opening the turbine bypass valves. As the reactivity changed about $\pm$ 40 PCM, control rods were moved in step changes to keep the reactivity values on scale and also to keep core power in the range necessary to produce an adequate intermediate range signal. After the core average temperature decreased by about 5°F, coolant temperature and reactivity were stabilized. This process was then reversed except that the core average temperature was increased by about 10°F. After stabilizing coolant temperature and reactivity, the core average temperature was then returned to its original value. The measurement of the temperature coefficient from the data obtained was then performed by dividing the change in reactivity by the corresponding changes in core temperature over a specific time period.

Isothermal temperature coefficient measurements were conducted at two different reactor coolant boron concentrations during the Zero Power Physics test program. The results of the measurements are summarized in Table 3.8-1 along with the predicted values which are included for comparison. In all cases the measured

results compared favorably with the predicted values. All measured temperature coefficients of reactivity were within the acceptance criteria of ± 0.40 X 10^{-4} Δ k/k/°F of the predicted value.

The moderator coefficient cannot be directly measured in an operating reactor because a change in moderator temperature causes a similar change in the fuel temperature. However, since the moderator coefficient has safety implications, it is an important reactivity coefficient. As specified in Technical Specifications the moderator temperature coefficient shall not be positive at power levels above 95 percent full power and shall be less than +0.9 X 10^{-4} Δ k/k/°F at all other power levels. To obtain the moderator coefficient from the measured temperature coefficient, a Doppler correction of -0.2 X 10^{-4} Δ k/k/°F must be subtracted. Determination of the moderator coefficient has been completed and shows that this coefficient is well within the limits stated above.

In conclusion, all temperature coefficients of reactivity that were measured at the 532°F and 2155 Psig plateau were within the acceptance criteria of ± 0.40 X 10^{-4} Δ k/k/°F of the predicted value. In addition, calculation of the moderator coefficient indicates that it is well within the requirements of the Technical Specification 3.1.1.3.

3.9 BIOLOGICAL SHIELD SURVEY

A Biological Shield Survey was not done at Hot Zero Power since no plant modifications were made which would invalidate the Biological Shield Surveys made during the Initial Startup Testing Program.

3.10 EFFLUENT AND EFFLUENT MONITORING

No effluent or effluent monitoring testing was performed as these systems have been performing normally since initial startup and no further testing was required.

3.11 CHEMICAL AND RADIOCHEMICAL TESTS

Chemical and radiochemical testing was not performed during this startup. These tests were conducted at initial startup and no plant modifications have been made which would invalidate the results of those tests.

Table 3.0-1

Acceptance Criteria Deviation Limits Between Measured and Predicted Values

	Core Physics Parameters	Allowable Deviation Between Predicted and Measured Values
A.	Control Rod Worths	
	Group Worths (5 - 7)	± 15%
	Total Worth (5 - 7)	± 10%
	Ejected Rod Worth	± 20%
B.	Temperature Coefficients	$\pm 0.4 \times 10^{-4}$ Δ k/k/°F
C.	Differential Boron Worth	± 15%
D.	All Rods Out Boron Concentration	± 100 ppmB

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Table 3.0-2

Comparison Of Measured And Predicted Results Obtained During Zero Power Physics Test

	Physic Parameter	Units	Measured	Predicted	Acceptance Requirements	Comparison
1.	All Rods Out Critical Boron	ppmB	1294	1260	Measured Value must be within 100 ppmB of predicted value	OK (34)
2.	Sensible Heat	amps	5.00×10^{-8}		None	
3.	NI Overlap	decades	2.10		Greater than 1.0 decade	OK (>1.0)
4.	Control Rod Group Worth	% Δ k/k			Percent deviation of group worth should be less than 15%	
	Group 7		-0.84	-0.76		OK (-9.5%)
	Group 6		-0.97	-0.94		OK (-2.9%)
	Group 5		-1.15	-1.09		OK (-4.9%)
	TOTAL		-2.96	-2.79	Percent deviation of total worth should be less than 10%	OK (-5.7%)
5.	Ejected Rod Worth (6-4)	% Δ k/k				
	Measured		0.41	0.55	Percent deviation of ejected rod worth must be within 20% of the predicted value.	(32.7%)*
	Measured Worst Case		0.44		Value must be less than 1.0% Δk/k	OK (>1.0%)
6.	Temperature Coefficient	10^{-4} Δ k/k°F			Measured value must be within 0.4×10^{-4} Δ k/k°F of the predicted value.	
	At 1287 ppmB		+0.103	-0.019		OK (0.122)
	At 1015 ppmB		-0.624	-0.681		OK (0.057)

* Deemed safe and acceptable by Plant Review Committee

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Table 3.0-2 (Cont'd)

Comparison Of Measured And Predicted Results Obtained During Zero Power Physics Test

	Physic Parameter	Units	Measured	Predicted	Acceptance Requirements	Comparison
7.	Moderator Coefficient	10^{-4} Δk/k/°F			Maximum positive moderator coefficient must be less than 0.9 X $10x^{-4}$ Δk/k/°F	
	At 1287 ppmB		+0.303	+0.172		OK
	At 1015 ppmb		-0.424	-0.490		OK
8.	Differential Boron Worth	% Δk/k/ppmB				
	At 1148 ppmB		-0.01067	-0.01008	Percent deviation must be less than 10%	OK (5.5%)

NOTE: Percent Deviation is defined by the following formula:

$$\% \text{ Deviation} = \frac{\text{Predicted Value} - \text{Measured Value}}{\text{Measured Value}} \times 100$$

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Table 3.2-1

Summary of Nuclear Instrumentation Overlap Measurements During Zero Power Physics Test

Data Set	Source Range Indication NI-1 (CPS)	NI-2 (CPS)	Intermediate Range Indication NI-3 (AMPS)	NI-4 (AMPS)	Average SR Indication (CPS)	Average IR Indication (AMPS)	Overlap (Decades)
01	8.0×10^{4}	8.0×10^{4}	1.0×10^{-10}	1.0×10^{-10}	8.0×10^{4}	1.0×10^{-10}	2.10
02	8.0×10^{5}	8.0×10^{5}	1.0×10^{-9}	1.0×10^{-9}	8.0×10^{5}	1.0×10^{-9}	2.10
						Average =	2.10

Note(1): Overlap is obtained between the Source and Intermediate Range by using the average indications in the equation below.

$$\text{Overlap} = (6 - \text{Log SR}) + (\text{Log IR} + 11)$$

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Table 3.4-1

Comparison Of Reactimeter and Doubling Time (DT) Reactivity Measurements

Case No.	DT (Sec)	DT Converted Reactivity (% Δk/k)	Reactimeter Reactivity (% Δk/k)	Absolute Error (%)
1	189.3	-0.0315	-0.0320	1.6
2	131.7	+0.0332	+0.0320	3.8
3	122.7	-0.0539	-0.0520	3.6
4	81.3	+0.0497	+0.0500	0.4
5	87.3	-0.0901	-0.0890	1.2
6	32.0	+0.0988	+0.0990	0.2

NOTE: The Absolute Error Between DT Converted Reactivity and Reactimeter Reactivity is Defined By the Equation Below:

$$E\ (\%) = 100 \left| (\rho R - \rho DT) \ / \ (\rho DT) \right|$$

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Table 3.4-2

All Rods Out Critical Boron Concentration (PPM Boron)

Moderator Temperature	Predicted Results	Measured Results
532°F	1260	1294 (1)

Note (1): This number is based on the measured boron concentration with a 5 ppmB correction to account for the fact the rods were not quite at 100% withdrawn.

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Comparison Of Predicted And Measured Control Rod Group Reactivity Worth
At BOC-2

A. Moderator Temperature at 532°F, APSR's at 37.5% Withdrawn

Rod Group	Number Of Rods	Predicted Worth (% $\Delta k/k$)	Measured Worth (% $\Delta k/k$)	Percent Deviation (%)
5	8	-1.09	-1.15	-4.9
6	8	-0.94	-0.97	-2.9
7	8	-0.76	-0.84	-9.5
	Total	-2.79	-2.96	-5.7

B. Moderator Temperature at 579°F, APSR's at 37.5% Withdrawn

Rod Group	Number Of Rods	Predicted Worth (% $\Delta k/k$)	Estimated Worth (% $\Delta k/k$)
5	8	-1.17	-1.23
6	8	-1.02	-1.05
7	8	-0.85	-0.94
		-3.04	-3.22

NOTE: Estimated worth is determined by applying the percent deviations between predicted to measured results at 532°F to the predicted results at 579°F.

Table 3.5-1

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Differential Boron Reactivity Worth Measurements During Zero Power Physics Test

Rod Position, % wd								Measured Boron Conc (ppm)	Average Boron Conc (ppm)	Delta Boron Conc (ppm)	Boron Worth (% Δk/k)	Differential Worth %Δk/k/ppmB	
1	2	3	4	5	6	7	8					Measured	Predicted
100	100	100	100	100	100	94	37.5	1284	1148	272	2.902	0.01067	0.01008
100	100	100	100	0	0	0	37.5	1012					

Table 3.6-1

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Table 3.7-1

Determination of The Relative Symmetric Ejected Rod Worth at Zero Power, BOL, And 532°F Conditions

Rod Position, % wd								Inserted Reactivity (% Δk/k)	RCS Boron Concentration (ppm)	Ejected Rod Position	Change In Reactivity (% Δk/k)	Ejected Rod Work, % Δk/k	
1	2	3	4	5	6	7	8					Measured	Worst Case
100	100	100	100	28.1	0	0	37.5	2.77	1069	100	0.33	0.41	0.44
100	100	100	100	60.0	0	0	37.5	2.44		0			
100	100	100	100	30.2	0	0	37.5	2.73		100	0.29	0.39	
100	100	100	100	61.1	0	0	37.5	2.44		0			
100	100	100	100	32.3	0	0	37.5	2.69		100	0.25	0.37	
100	100	100	100	60.0	0	0	37.5	2.44		0			
100	100	100	100	28.1	0	0	37.5	2.77		100	0.33	0.41	
100	100	100	100	60.0	0	0	37.5	2.44		0			

Note: The measured ejected rod worth has been normalized to the rod configuration of Banks 5-7 0% withdrawn.

Note: The worst case ejected rod worth is obtained by application of an uncertaininty factor of approximately 1.10.

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Summary of Measured, Calculated, And Predicted Coefficients of Reactivity At Zero Power

Rod Position, % wd								Average Temperature (Deg F)	RCS Boron Concentration (ppmB)	Reactivity Coeffients, 1.0E-04 %k/k/°F			
										Temperature Coefficient		Moderator Coefficient	
1	2	3	4	5	6	7	8			Measured	Predicted	Calculated	Predicted
100	100	100	100	100	100	94	100	532	1287	+.103	-0.19	+.303	+0.172
100	100	100	100	0	0	0	38	532	1015	-0.624	-0.681	-0.424	-0.490

Note: The calculated moderator coefficient has been determined from the measured temperature coefficient by subtracting off the isothermal Doppler coefficient of -0.20 $10^{-4} \frac{k}{k}$/°F.

Table 3.8-1

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INCORE MONITOR AND CONTROL ROD MAP

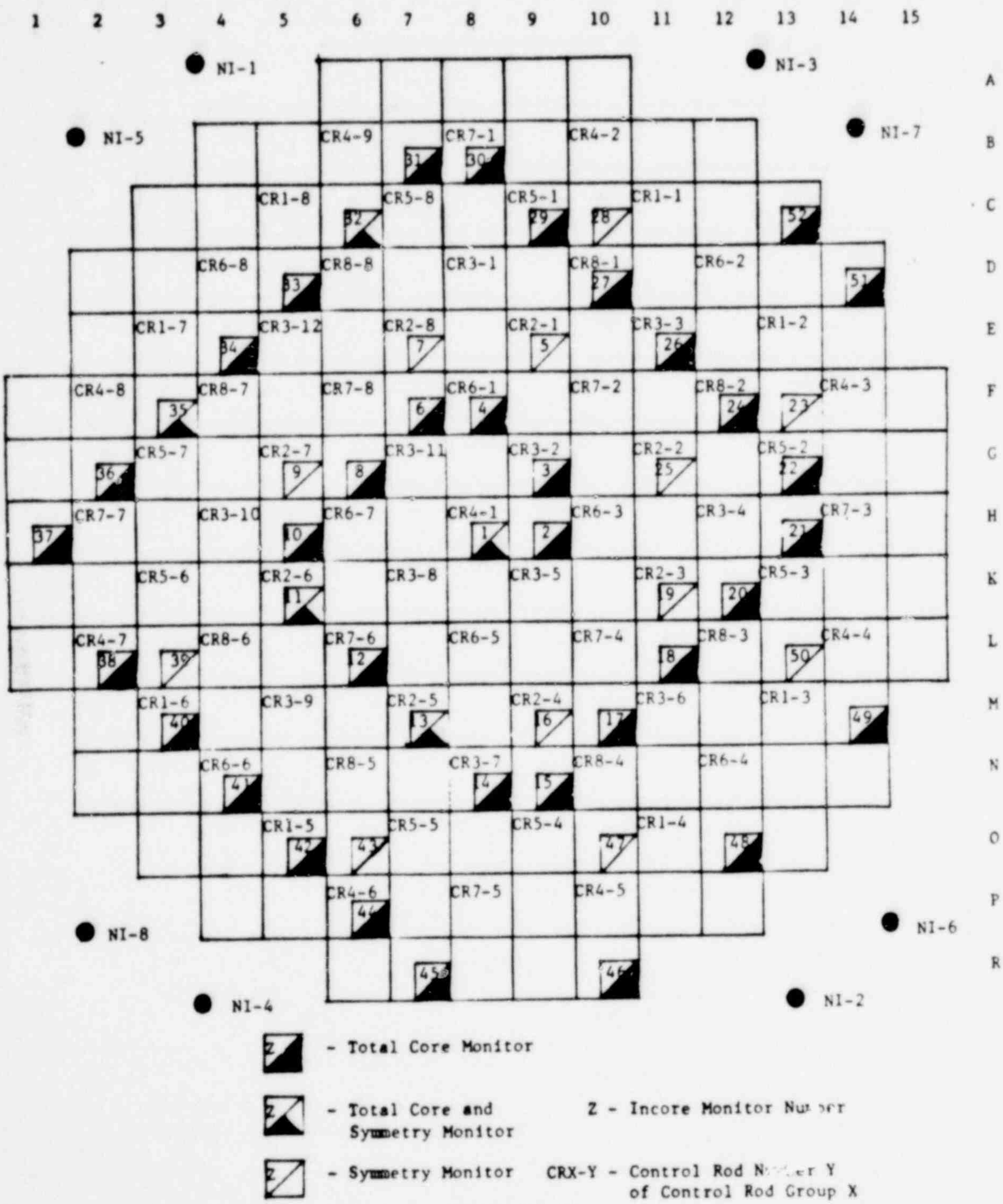

Figure 3.5-1

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CONTROL ROD GROUP 5
INTEGRAL WORTH
BOC-2,HZP

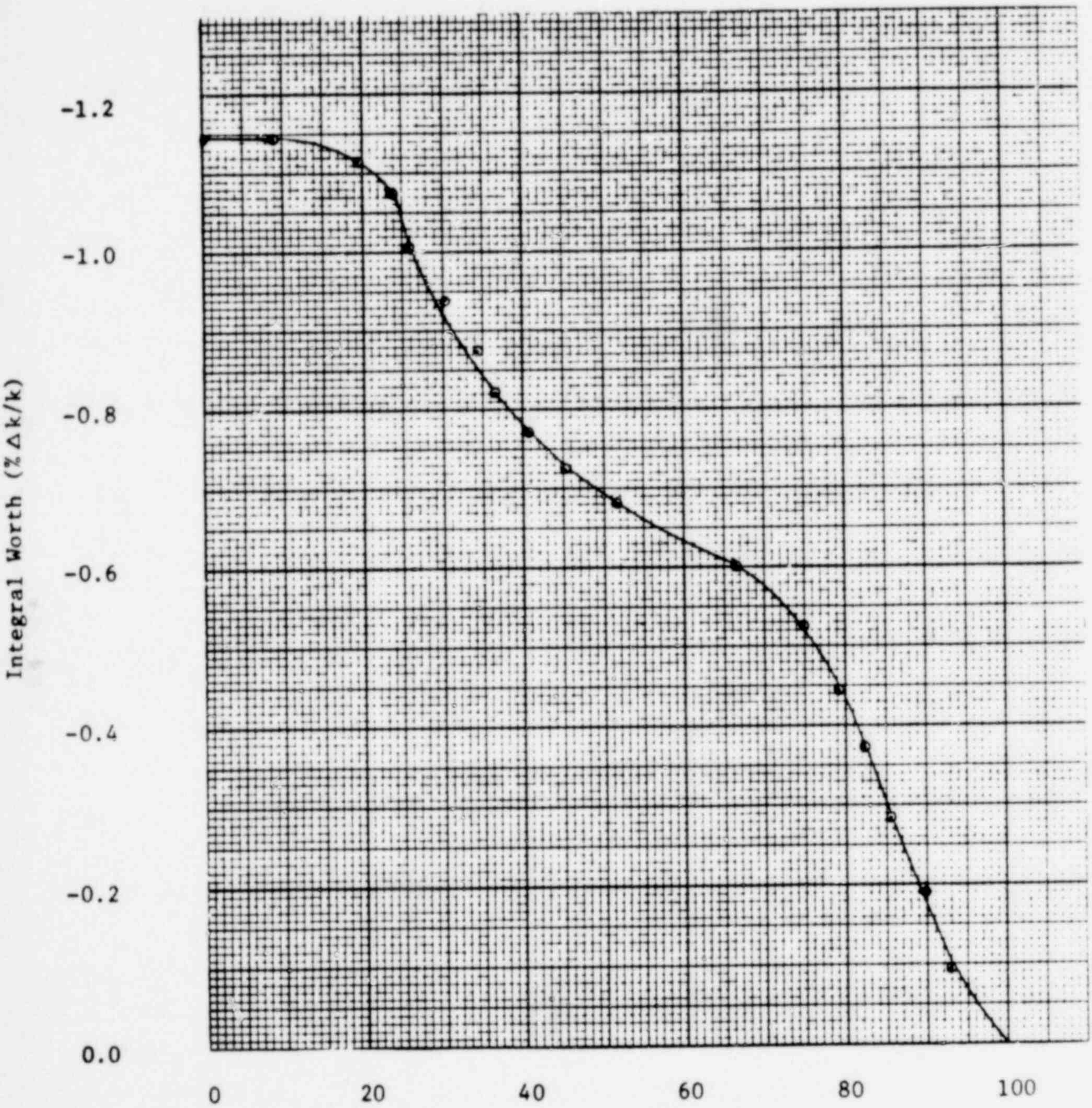

Figure 3.5-2

POOR ORIGINAL

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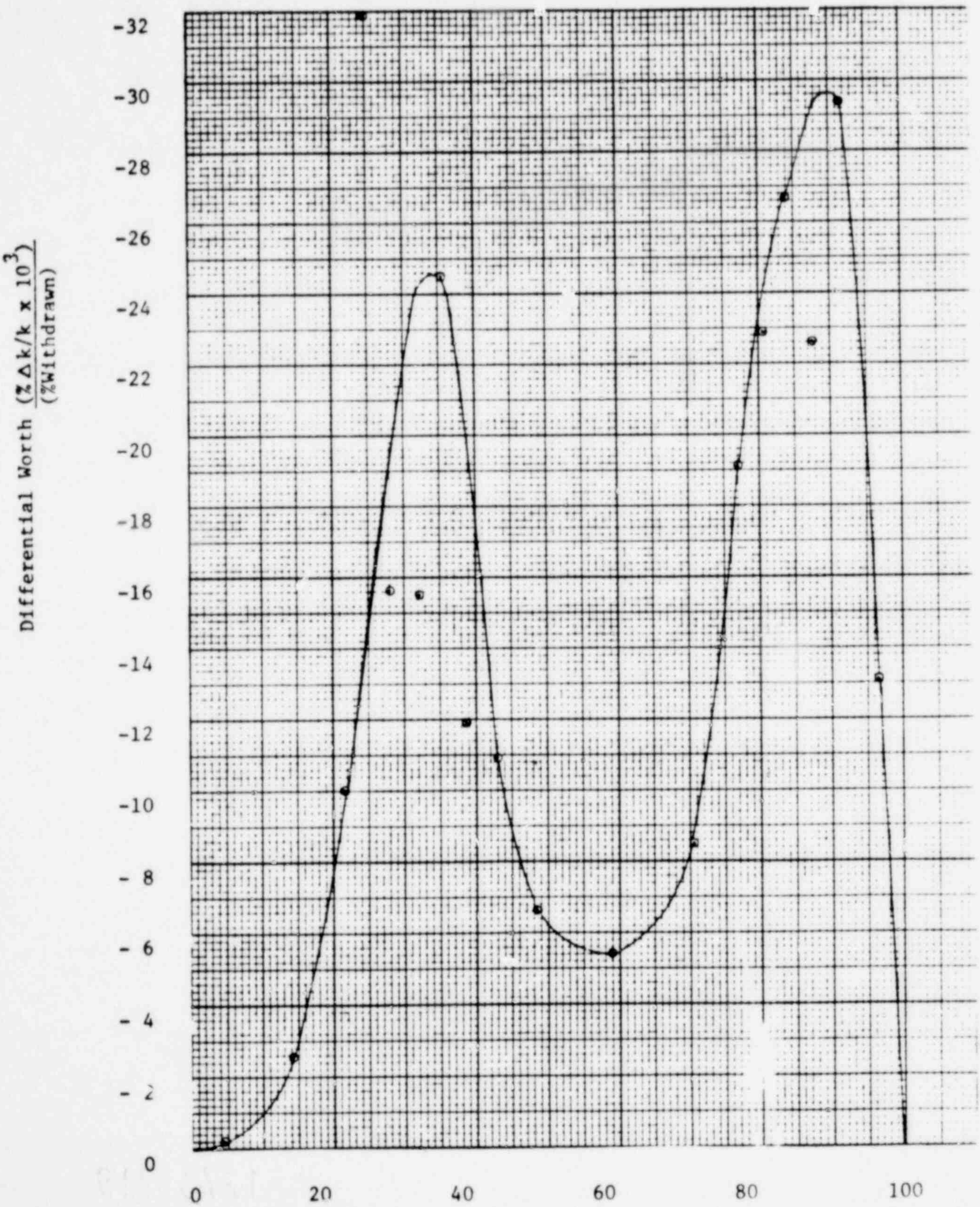
CONTROL ROD GROUP 5
DIFFERENTIAL WORTH
BOC-2, HZP



POOR ORIGINAL

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Figure 3.5-3

CONTROL ROD GROUP 6
INTEGRAL WORTH
BOC-2,HZP

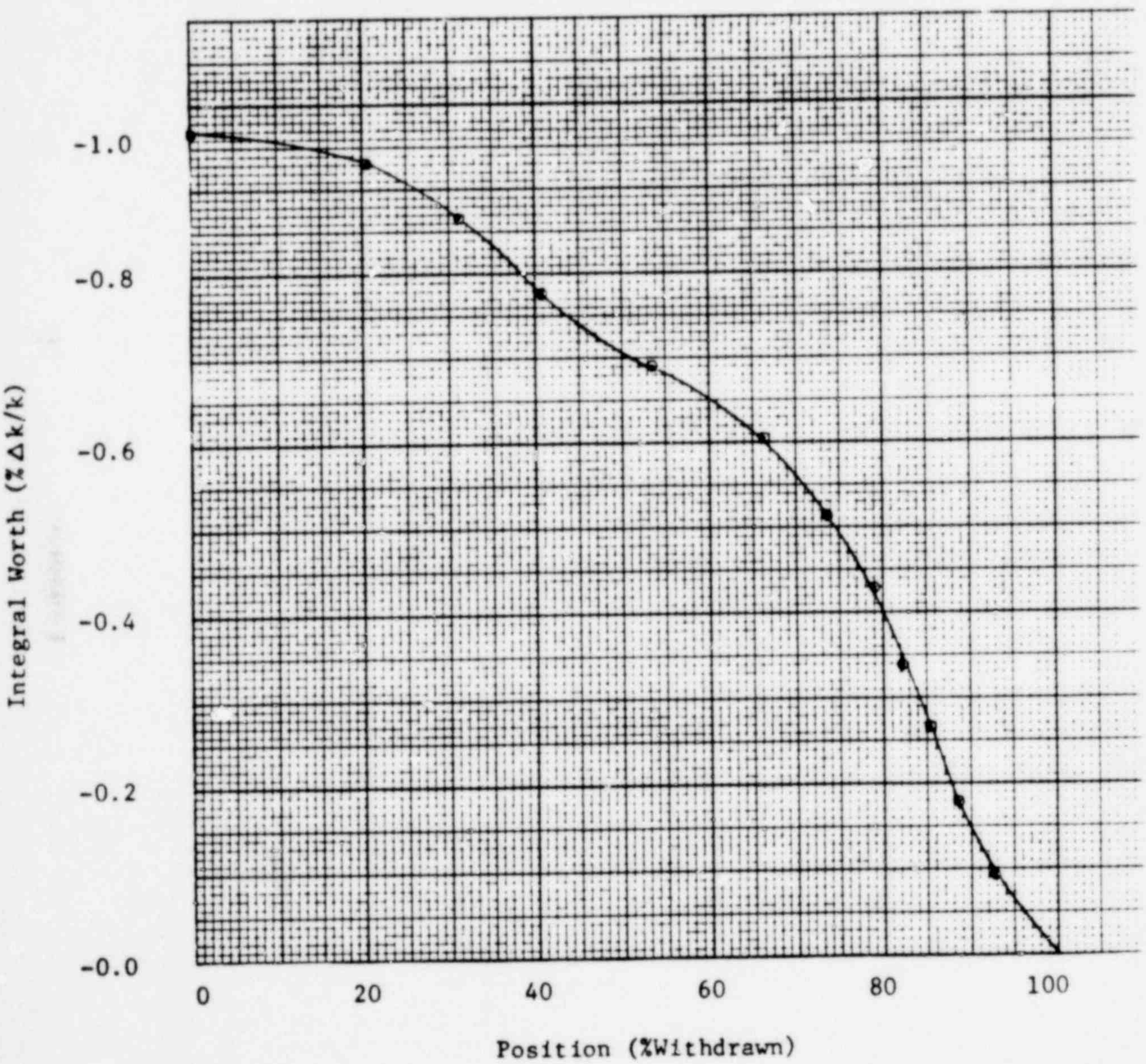

Figure 3.5-4

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CONTROL ROD GROUP 6
DIFFERENTIAL WO.TH
BOC-2,HZP

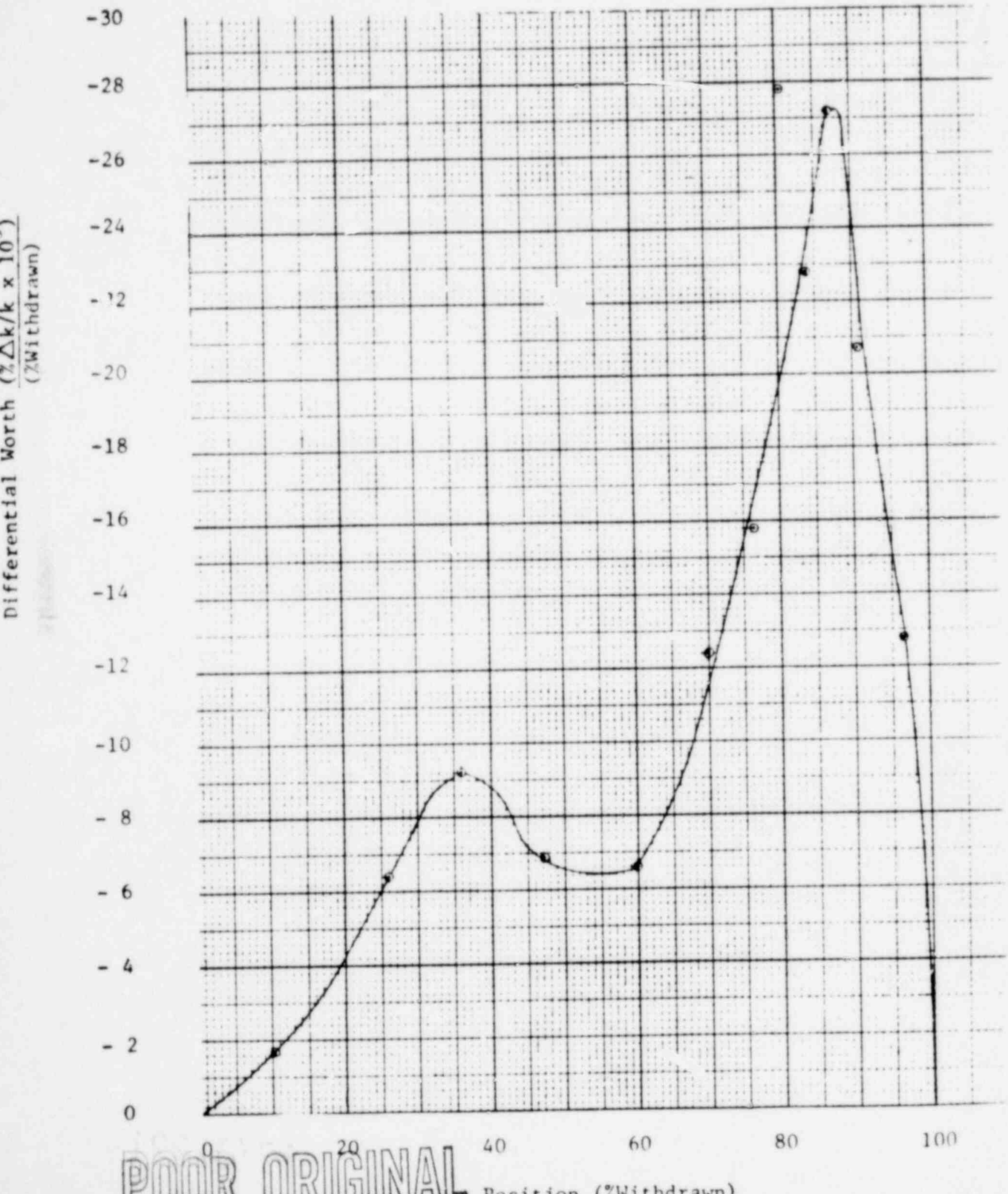

Figure 3.5-5

1273 ?22

CONTROL ROD GROUP 7
INTEGRAL WORTH
BOC-2,HZP

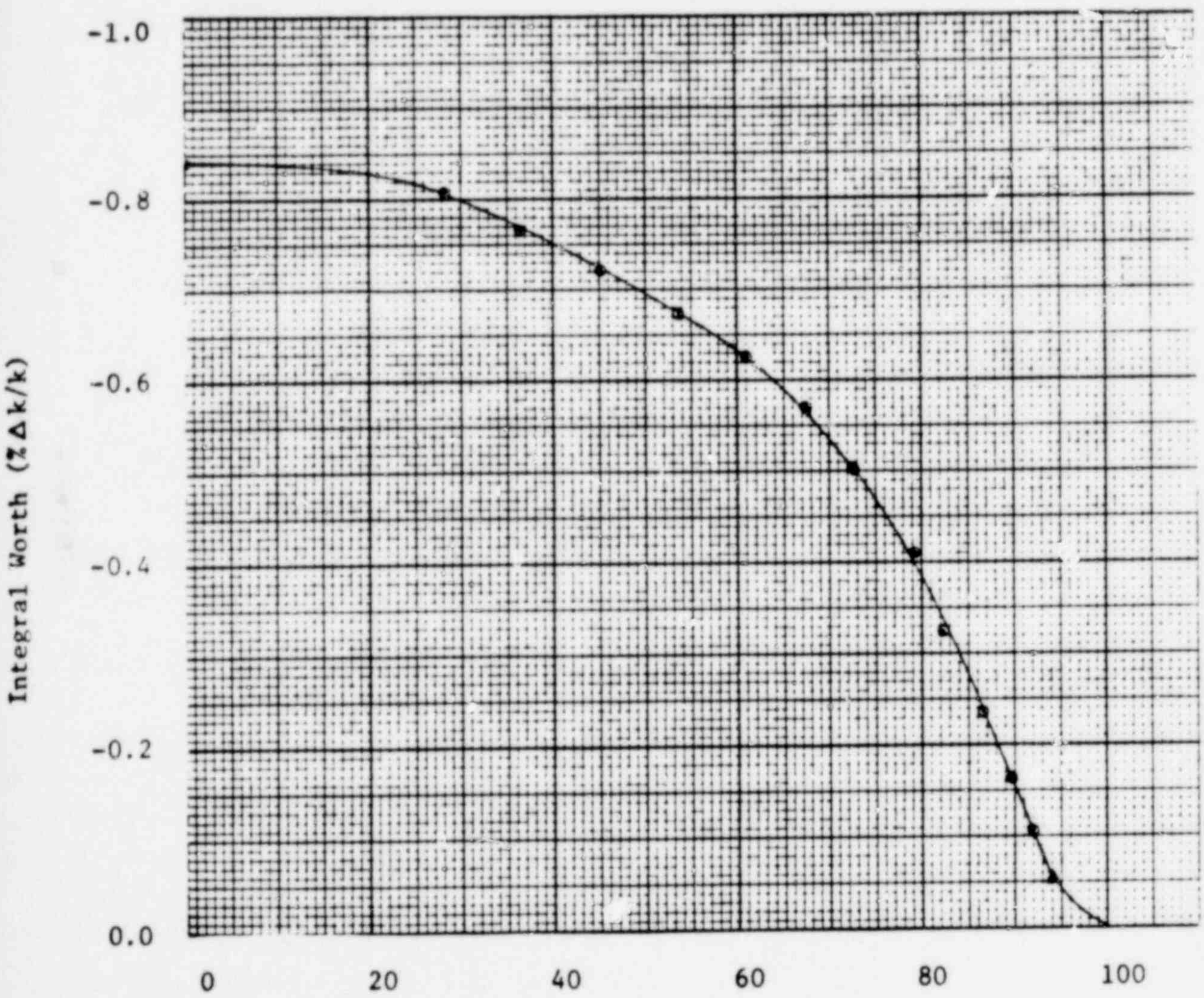

Figure 3.5-6

POOR ORIGINAL

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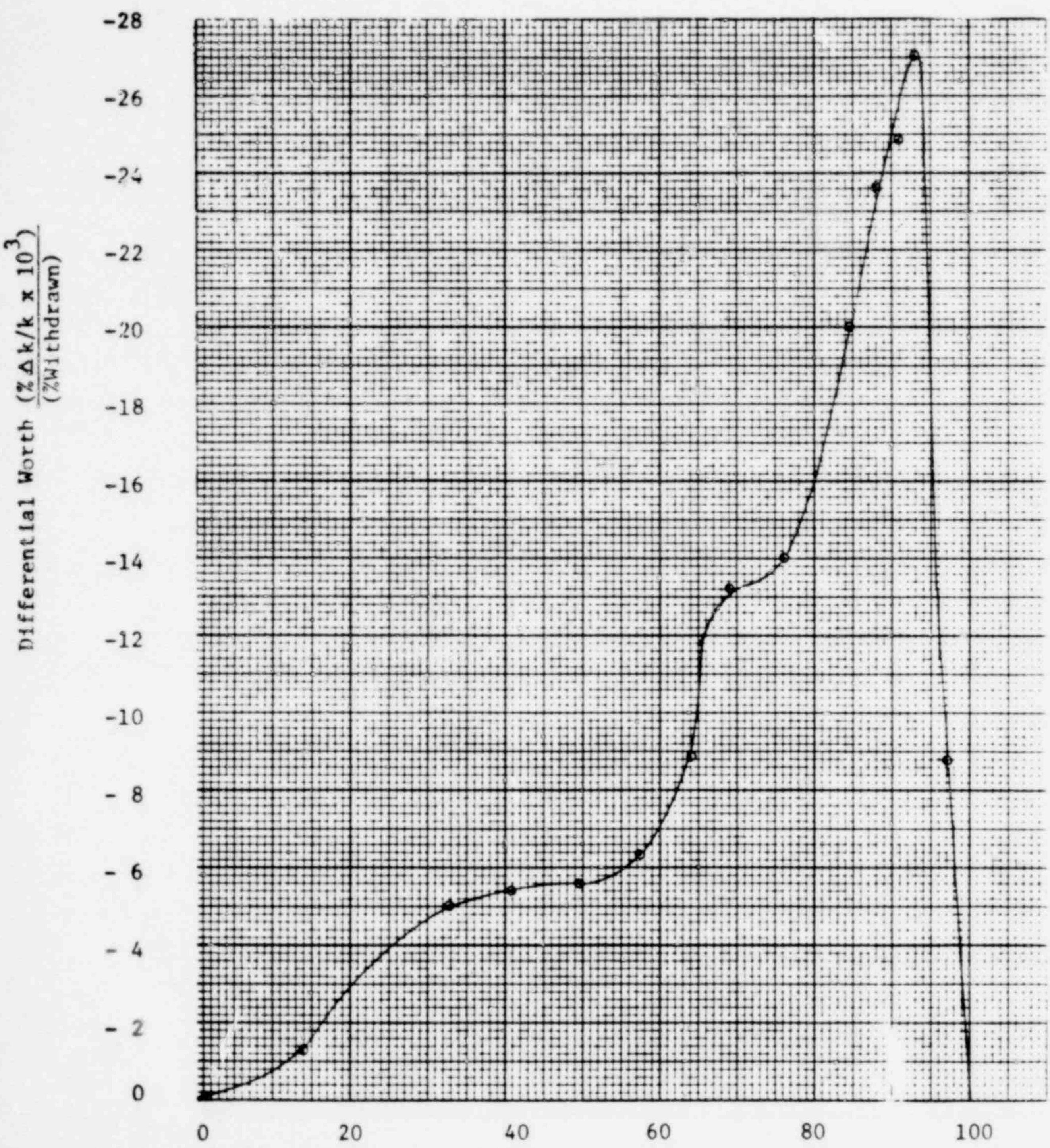
CONTROL ROD GROUP 7
DIFFERENTIAL WORTH
BOC-2,HZP



POOR ORIGINAL

Figure 3.5-7

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4.0 POWER TESTING

4.1 REACTIVITY COEFFICIENTS AT POWER TEST

4.1.1 PURPOSE

The purpose of this test was to determine reactivity coefficients during power operation at 100 percent full power, and to verify that they were conservative with respect to the FSAR. The following coefficients were either measured or calculated from the data obtained.

(a) Temperature coefficient of reactivity, defined as the fractional change in the reactivity of the core per unit change in fuel and moderator temperature.

(b) Moderator coefficient of reactivity, defined as the fractional change in the reactivity of the core per unit change in moderator temperature.

(c) Power Doppler coefficient of reactivity, defined as the fractional change in the reactivity of the core per unit change in power.

Acceptance criteria are specified for the Reactivity Coefficients at Power Test and are listed below:

(1) The moderator coefficient of reactivity measured at 100% full power shall be negative.

(2) The power Doppler coefficient of reactivity shall be more negative than -0.55×10^{-4} $\Delta k/k/\%FP$.

4.1.2 TEST METHOD

Reactivity coefficient measurements were made during the power escalation test program at 100% full power.

Differential rod worth measurements were performed during the reactivity coefficient measurement in order to generate rod worth data for the specific test conditions. For temperature coefficients, average reactor coolant temperature was increased and decreased about 5°F and data recorded. For power Doppler coefficients, power was increased and decreased about 5 percent full power and data recorded. From the measured temperature and power Doppler coefficients, the moderator and Doppler coefficients were calculated.

4.1.3 TEMPERATURE AND MODERATOR COEFFICIENTS

The temperature coefficient of reactivity is defined as the fractional change in the reactivity of the core per unit change in fuel and moderator temperature. The temperature coefficient is normally divided into two components as shown in equation 4.1-1.

$$\alpha T = \alpha M + \alpha D \qquad \text{EQ. (4.1-1)}$$

Where: αT = Temperature Coefficient of Reactivity
αM = Moderator Coefficient of Reactivity
αD = Doppler Coefficient of Reactivity

The moderator coefficient cannot be directly measured in an operating reactor because a change in the moderator temperature also causes a similar change in the fuel temperature. Therefore, the moderator coefficient must be calculated using equation 4.1-1 after the temperature and Doppler coefficients have been determined. Technical Specifications, section 3.1.1.3, requires that the moderator coefficient be less than $+0.90 \times 10^{-4}$ $\Delta k/k/°F$ at power levels below 95% full power and non-positive at or above 95% full power.

Temperature and moderator coefficients were theoretically predicted as shown in Figures 4.1-1 and 4.1-2 using the distributed moderator and fuel temperatures instead of the isothermal values which were used for zero power physics predictions. For these predictions, the normal mode of operation with critical boron and rod conditions was assumed which set the average core moderator temperature equal to 579°F.

The measurement method used at power is to change the reactor coolant temperature setpoint at the reactor control station with the integrated control system in automatic effecting an approximate 5°F change in the reactor coolant temperature. The reactivity change caused by the temperature change of the core was measured by recording the change in the position of the controlling control rod group and converting this change to reactivity using differential rod worth values measured during the test. Prior to running the test, steady state equilibrium xenon conditions including a stable boron concentration and no significant control rod motion during the last 30 minutes prior to taking data were required as prerequisite system conditions.

The power Doppler coefficient relates the change in core reactivity to a corresponding change in power. Theoretical predictions of the power Doppler coefficient were made using the PDQ code with thermal feedback. The predicted power Doppler coefficients using this code are presented in Figure 4.1-3.

The measurement method used was to change the reactor power level 5 percent full power. This change in power level was initiated by manually decreasing the reactor power at the reactor master control station. After obtaining approximately ten minutes of steady state data at the reduced power level, reactor power was returned to the initial power.

The calculation of the power Doppler coefficient uses the measured change in the controlling rod group position converted to an equivalent reactivity value and the measured change in reactor power determined by using the normalized core ΔT which is the primary side heat balance.

4.1.4 DIFFERENTIAL ROD WORTH AT POWER

The method by which the differential rod worth was determined at power is the fast insertion/withdrawal method. In this measurement, the controlling rod group is inserted for approximately six seconds, followed immediately by a withdrawal for approximately six seconds. Since the total elapsed time is on the order of the primary loop recirculation time, the moderator temperature effects are eliminated and the reactivity versus time is essentially a combination of the effects due to the control rod motion and the fuel power variation.

Determination of the differential rod worth was then found by using the measured reactivity and rod positions, compensating the data by a predicted fuel power correction factor.

The fuel power correction factor accounts for the time delay involved in fuel temperature change during the measurement.

4.1.5 EVALUATION OF TEST RESULTS

The results of the measured temperature and calculated moderator coefficients at power are plotted in Figures 4.1-1 and 4.1-2 which also shows the predicted temperature and moderator coefficient results.

Examination of the calculated moderator coefficient as plotted in Figure 4.1-2 indicates that the limit of a non-positive value is not to be exceeded at 95 percent full power. The soluble poison concentration for equilibrium-xenon all-rods-out condition is about 805 ppmB. Thus during power operation at or above 95 percent full power the moderator coefficient will be negative even if all control rods are withdrawn from the core.

The results of the measured and predicted power Dopper coefficient of reactivity are plotted in Figure 4.1-3. A list of the measured reactivity coefficients along with the conditions under which they were measured is given in Table 4.1-1.

The acceptance criterion for the measured power Doppler coefficient is that the coefficient must be more negative than -0.55×10^{-4} $\Delta k/k/\%FP$. Figure 4.1-3 shows that the measured coefficient is below this value and that the acceptance criterion is adequately met.

4.2 UNIT HEAT BALANCE

Heat balance calculations were performed using both the Bailey 855 unit computer, and the IBM 5100 computer. The accuracy of these two methods was verified during initial startup by performing hand calculations. No modifications were made during this shutdown which would require reverification of the heat balance calculation.

4.3 CORE POWER DISTRIBUTION TEST

Core power distributions were taken as required during the power escalation test program as a part of the following individual tests.

Core Power Distribution Test
Power Imbalance Detector Correlation Test
Incore Detector Test

Continuous monitoring of the core power density at 364 core locations is accomplished by the incore monitoring system. This system is comprised of 52 detector strings each having 7 individual neutron detectors equally spaced at seven axial elevations in the center of 52 fuel assemblies. This system is capable of producing detailed core power distributions for either eighth core or quarter core symmetry conditions. A detailed description of the incore monitoring system and the calibration test results are presented in Section 4.13. The output of the incore detectors is connected to the unit computer and is corrected for background, fuel depletion and the as-built dimensions to provide accurate ouputs of relative neutron flux. The computer output of the corrected signals is used to develop core power distributions which provide power peaking information necessary to determine core performance in terms of DNBR and LHR.

Implementation of the core power and core power imbalance safety limits, in terms of the reactor protection setpoints, is shown in Figure 4.3-1.

The out-of-core nuclear instrumentation provides the core power and core power imbalance signals to the RPS, since the incore monitoring system does not immediately respond to prompt changes in core conditions.

The results of the core power distributions taken during the test program as part of the power escalation sequence are discussed by dividing this section into three subsections as follows:

(1) Normal Operating Core Power Distributions

(2) Worst Case Minimum DNBR and Maximum LHR Calculations

(3) Quadrant Power Tilt and Axial Power Imbalance

4.3.1 PURPOSE

The purposes of the steady state core power distribution measurements are as follows:

(a) To measure core power distribution and thermal-hydraulic data at the major test plateaus as required by the power escalation test program.

(b) To compare the measured and predicted core power distributions at 40, 75, and 100% full power.

(c) To verify acceptable core thermal-hydraulic parameters at 40, 75, and 100% full power.

Four acceptance criteria are specified for the Core Power Distribution Test and are listed below:

(1) The core power distribution and thermal-hydraulic parameters have been measured, evaluated, and deemed reasonable.

(2) The highest measured radial and total peaking factors are not more than 8.0 and 12 percent greater than predicted, respectively, at 40% FP and 5.0 and 7.5 percent greater than predicted, respectively, at 75% FP and 100% FP.

(3) The measured worst case minimum DNBR is greater than 1.30 and the measured worst case maximum LHR is less than 18.00 kW/ft.

(4) The extrapolated worst case minimum DNBR is greater than 1.30 and the extrapolated worst case maximum LHR is less than 19.70 kW/ft., or the extrapolated imbalance falls outside the power imbalance trip envelope as shown in Figure 4.3-1.

A review criterion was also added requiring the location of the measured radial and total peaks to match the locations of the predicted peaks.

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4.3.2 TEST METHOD

Computer printouts of the core power distribution and thermal hydraulics conditions were obtained after establishing steady state conditions at the required power level and rod configurations as determined by the Controlling Procedure for Power Escalation, PT-120. Three-dimensional equilibrium xenon was required, the APSR's were maintained at a constant position and the axial incore imbalance was maintained within +/-2% FP of zero.

4.3.3 EVALUATION OF THE TEST RESULTS

4.3.3.1 NORMAL OPERATING CORE POWER DISTRIBUTIONS

Normal operating, equilibrium xenon core power distributions were measured and predicted for each of the three major power escalation test plateaus. The results of these measured core power distributions, given in Figures 4.3-2 to 4.3-7, at operating control rod configurations indicate a maximum radial peaking factor of 1.39, and a maximum total peaking factor of 1.71 for the three cases studied. The total peaking factor observed during normal operation was well below the design maximum total peaking factor of 2.67.

At the 40% plateau, both acceptance criteria of 8.0 and 12.0% for radial and total peak differences between calculated and measured values, were met. At 75% FP, where the acceptance criteria narrowed to 5.0 and 7.5%, the measured radial and total power distributions did not meet acceptance criteria. Babcock and Wilcox revised their calculation to include the actual EOC-1 exposure and the measured position of the axial power shaping rods. Using these new calculated power distributions, the acceptance criterion of 7.5% agreement for the total peak was met, however, the radial peak remained outside the acceptance criterion of 5.0% agreement. Babcock and Wilcox concluded that since Cycle 2 is limited according to the <u>total</u> peaking factor and the total peak was within the acceptance criterion of 7.5%, Crystal River 3 could be safely operated at 100% FP.

At the 100% FP plateau, the acceptance criterion of 7.5% agreement for the total peak was met, however, the radial peak acceptance criterion of 5.0% agreement was not. As in the 75% FP case, Babcock and Wilcox re-ran the core power distribution calculations at 100% FP, using the actual EOC-1 exposure and the measured position of the axial power shaping rods. The radial power peak agreement between the measured and calculated values improved but not enough to meet the acceptance criterion. Again, it was concluded that since the <u>total</u> peaking factor, being the Cycle 2 limiting factor, was within the acceptance criterion, there would be no adverse effects caused by the discrepancy between the measured and calculated radial peaks at 100% FP.

Introduced in the Cycle 2 Startup Physics Test was the review criterion requiring the measured radial and total peaks to be where predicted. Only the total peak at 100% FP met this review criterion, however, Table 4.3-1 reveals that the measured peaking factors occurring at the predicted peak locations were relatively close ($\leq$1.8%) to the actual measured peak. Since the accuracy required by the acceptance criteria for predicting the radial and total peaks was 5.0% and 7.5%, respectively, it was not unexpected when the predicted and measured peak locations did not agree.

4.3.3.2 WORST CASE MINIMUM DNBR AND MAXIMUM LHR CALCULATIONS

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal peaking conditions. The two primary core thermal limits which are indicative of fuel thermal performance are fuel melting and departure from nucleate boiling. These limits are independent, each must be evaluated to ensure core safety for a given power peaking situation. Fuel melting is basically a function of the local power generated in the fuel which is a combination of radial and axial peaking. The maximum allowable linear heat rate limit is 19.70.

The upper boundary of the nucleate boiling region is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperature and the possibility of cladding failure. The local DNB ratio, (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB.

Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature and pressure can be related to DNB through the use of the B&W-2 correlation. The B&W-2 correlation has been developed to predict DNB and the location of DNB for uniform and non-uniform axial heat flux distributions. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to 94.5% probability at a 99% confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operational conditions.

WORST CASE MINIMUM DNBR DETERMINATION

The worst case minimum DNBR values were calculated by the unit computer for each core power distribution taken as part of the power escalation test program. The results of various worst case minimum DNBR values calculated at each test plateau under normal rod configurations are plotted in Figure 4.3-8. These results indicate that all measured values were above the design worst case minimum DNBR versus power level and well above the minimum acceptable value of 1.30.

Three normal operating equilibrium xenon core power distributions, as required by the Core Power Distribution Test, were obtained during the power escalation sequence. Each distribution was subjected to the following analysis:

(a) From each core power distribution, the worst case measured minimum DNBR was selected.

(b) The measured worst case minimum DNBR's were then extrapolated to the overpower trip setpoint and corrected for axial peak location and magnitude.

(c) Verification of acceptable core conditions at the present and next power level of escalation were then performed relative to 1.30.

(d) The measured worst case minimum DNBR's were then extrapolated to the LOCA and design overpower power level and corrected for axial peak location and magnitude.

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WORST CASE MAXIMUM LHR DETERMINATION

Worst case maximum LHR values were calculated using SP-104 "Hot Channel Factors Calculations" for each standard core power distribution taken as part of the power escalation test program. In all cases, the above limit was met during the power escalation test program, Table 4.3-2.

4.3.3.3 QUADRANT POWER TILT AND AXIAL POWER IMBALANCE

QUADRANT POWER TILT

Quadrant power tilt limits have been established in the Technical Specifications. These limits, when used in conjunction with the control rod position limits, assure that the design peak heat rate criterion is not exceeded during normal power operation.

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$\text{Quadrant Power Tilt} = \left[\frac{\text{Power in Any Core Quadrant}}{\text{Average Power of All Quadrants}} - 1\right] \times 100 \quad \text{EQ. (4.3-1)}$$

During the startup testing program, maximum quadrant power tilt was determined using the corrected signals from the 16 symmetric incore monitoring assemblies. Figures 4.3-2 through 4.3-7 include the maximum quadrant power tilt for each standard core power distribution taken as required by the Core Power Distribution Test. Technical Specification tilt limits were not exceeded at any of the power plateaus.

AXIAL POWER IMBALANCE

Results from the Standard Core Power Distributions taken at 40% FP during the performance of the Power Imbalance Detector Correlation Test, show that the imbalance trip envelope (Figure 4.3-1) of the reactor protective system is sufficient to protect the unit from exceeding the DNBR and the LHR limits under all core imbalance conditions when a gain factor of 4.50 is set into the delta flux amplifier. In addition, analyses indicate that the largest thermal margins (measured by DNBR and LHR) exist when a negative 5.0% to 10.0% incore axial offset is present.

The core imbalances measured in conjunction with the core power distribution of this section are included in Figures 4.3-2 through 4.3-7.

4.4 BIOLOGICAL SHIELD SURVEY

A biological shield survey was not conducted as part of this power testing program since no plant modifications were made which would invalidate the biological shield surveys made during the initial startup testing program.

4.5 PSEUDO ROD EJECTION TEST

This test was not performed during the power escalation test program. This test was performed during zero power physics testing.

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4.6 TURBINE/REACTOR TRIP TEST

No turbine/reactor trip testing was performed during this startup testing program since no modifications were made which would invalidate the original testing results.

4.7 INTEGRATED CONTROL SYSTEM TEST

Since no modifications were made to the Integrated Control System during this outage no specific ICS testing was done during the startup testing program. Minor adjustments were made to the ICS during startup under normal maintenance and calibration procedures. The ICS functioned normally during the entire testing program.

4.8 UNIT LOSS OF ELECTRICAL LOAD

No loss of electrical load testing was performed during this startup testing program since no modifications were made which would invalidate the original testing results.

4.9 UNIT LOAD TRANSIENT TEST

No specific unit load transient testing was performed since no modifications were made to invalidate the results of previous tests. No major problems were encountered during transient operations during the testing program.

4.10 SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

No shutdown from outside the control room testing was performed since no modifications were made which would invalidate the results during initial startups.

4.11 LOSS OF OFFSITE POWER

No loss of offsite power testing was performed since no modifications were made which would invalidate the results of the tests performed during initial startup.

4.12 POWER IMBALANCE DETECTOR CORRELATION TEST

Imbalance of the neutron flux in a reactor results from temperature distributions, fuel depletion, xenon oscillations, or control rods positioned in the core. The amount of imbalance* which is allowed in the core so that DNBR or LHR limits are not exceeded is set in the reactor protection system and is a function of the power level and the reactor coolant flow. Since this imbalance is determined using input signals from the out-of-core detectors, it is essential that they are calibrated to read the true imbalance as determined from the incore detectors.

This section of the report represents the results of incore imbalance to out-of-core imbalance induced on Crystal River Unit 3, Cycle 1 at 40 percent full power during the performance of Power Imbalance Detector Correlation Test.

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* Imbalance = % power in top of core - % power in bottom of core

4.12.1 PURPOSE

The Power Imbalance Detector Correlation Test had the following four objectives:

(a) To measure the relationship between core offset* as indicated by the out-of-core power range nuclear instrumentation detectors and as indicated by the full incore monitoring system.

(b) To provide sufficient information to adjust the NI/RPS differential amplifiers should the measured correlation indicate a need.

(c) To verify that an acceptable relationship between the backup recorder system and the full incore monitoring system indication of core offset was observed.

(d) To verify acceptable core thermal-hydraulic parameters at each imbalance condition induced.

Three acceptance criteria are specified for the Power Imbalance Detector Correlation-Test and are listed below:

(1) The measured correlation between each out-of-core detector offset to full incore monitoring system offset lies within the acceptance region shown in Figure 4.12-1. The correlation slope shall be greater than or equal to 1.00 and the correlation intercept should be less than $\pm$ 3.5% offset.

(2) The measured relationship between the full incore monitoring sysem offset and the backup recorder offset lies within the acceptable region shown in Figure 4.12-2.

(3) The measured worst case minimum DNBR is greater than 1.30 and the measured worst case maximum LHR is less than 18.00 kW/ft.

4.12.2 TEST METHOD

During the Power Escalation Sequence at Crystal River 3, as part of the test program, imbalance measurements were made to determine the acceptability of the out-of-core detectors to detect imbalance and to establish a basis for verifying that DNBR and LHR limits would not be exceeded while operating within the flux/delta flux/flow envelope set in the reactor protection system. These imbalance measurements were made at 40% FP with a gain factor of 4.50 applied to the multiplier on the delta flux amplifier output.

In performing the test, APSRs were positioned to obtain the desired full incore imbalance with reactivity compensations made by control rod groups 6 and/or 7. At 40% FP, offset as indicated by the full incore system, out-of-core system, and backup recorder system was recorded as given in Table 4.12-1. From this data, plots of average out-of-core offset and backup recorder offset versus full incore offset were maintained as shown in Figures 4.12-1 through 4.12-2. Worst case minimum DNBR and maximum LHR data was monitored during the test. The results are plotted versus full incore offset in Figure 4.12-3.

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$$\text{* Offset} = \frac{\text{Imbalance}}{\text{Total Core Power}}$$

Based upon previous startup experience, the relationship between incore offset and out-of-core offset was determined to be a linear equation of the form below:

$$OCO = M \times ICO + B \qquad \text{EQ. (4.12-1)}$$

Where: OCO = Out-of-Core Offset (Percent)
ICO = Incore Offset (Percent)
M = Slope of Relationship
B = Intercept at Zero ICO

The experimental slope and intercept could then be obtained using a linear least squares fit from the data obtained. If the measured slope of the relationship of ICO to OCO was determined to be unsatisfactory; i.e. <1.15, the gain of the out-of-core power range detector should be adjusted. The relationship of measured slope to gain factor is as follows:

$$GF = (M2/M1) \times GF_0 \qquad \text{EQ. (4.12-2)}$$

Where: GF = Desired Gain Factor
M2 = Desired Slope (i.e. >1.15)
M1 = Measured Slope
GF_0 = Present Gain Factor (i.e. 4.50)

Verification of the adequacy of the power imbalance system trip setpoint was performed in conjunction with worst case analysis on each minimum DNBR and maximum LHR measured. The technique used was to extrapolate each measured point to the power/imbalance/flow envelope boundary limits given in Figure 4.3-1. In this way, the adequacy of the imbalance system trip setpoints to protect the unit from exceeding thermal-hydraulic limits could be verified.

4.12.3 EVALUATION OF TEST RESULTS

The measurement of the offset correlation function between the full incore system and each out-of-core detector was determined, during imbalance scans by APSR's and control rod group 6 and/or 7 at 40% full power, to be a linear relationship on all power range detectors. Figure 4.12-1 shows the average response in offset between the out-of-core power range detectors and the full incore system. Test data indicated that all measured offsets from the out-of-core power range detectors fell within the acceptable areas of the curve during the performance of the test. For each power range detector, a linear least squares fit was applied to the measured data points to obtain a value for the slope and intercept of the observed relationship. The results of these calculations are tabulated in Table 4.12-2. In all cases, the measured slopes were greater than the minimum allowable value of 1.15 which verified the utilization of a 4.50 gain factor for the difference amplifiers.

The ability of the backup recorder to follow full incore offset was also verified as part of this test by collecting backup recorder data and performing the necessary calculations. The results of this analysis are plotted in Figure 4.12-2.

During all phases of testing, worst case minimum DNBR and maximum LHR were recorded against incore offset and a plot maintained as shown in Figure 4.12-3. The most limiting value observed on the worst case minimum DNBR and maximum LHR was 7.85 and 5.77 kW/ft., respectively, which is well within the procedural acceptance criteria. As can be seen from the plot, both worst case minimum DNBR and maximum LHR are functions of incore offset.

4.13 NUCLEAR INSTRUMENTATION CALIBRATION AT POWER TEST

The purpose of the Nuclear Instrumentation Calibration at Power Test was to calibrate the power range nuc'ear instrumentation to within 2.0% full power of the thermal power as determined y a heat balance and to within 3.5% of the axial offset as determined by the incore monitoring system. Additional purposes during the power escalation program were as follows:

(a) To adjust the high power level trip setpoint when required by the power escalation test procedure.

(b) To confirm that the nuclear instrumentation calibration procedure can adequately adjust the power range channels to a heat balance and an incore offset.

(c) To verify that at least one decade overlap exists between the intermediate and power range nuclear instrumentation.

Four acceptance criteria are specified for the Nuclear Instrumenation calibration at Power Test as listed below:

(1) The power range nuclear instrumentation indicates the power level within 2.0% full power of the heat balance and within 3.5% incore axial offset. The incore axial offset criteria does not apply below 30% full power.

(2) The high power level trip bistable is set to trip at the desired value within a tolerance of +0.00 to -0.31% full power.

(3) The power range nuclear instrumentation can be calibrated adequately using the nuclear instrumentation calibration procedure.

(4) The overlap between the intermediate and power range is in excess of one decade overlap.

The power range instrumentation channels were calibrated several times during the power escalation test program. The results of the calibrations met the criteria specified for the test.

4.14 EMERGENCY FEEDWATER FLOW TEST

4.14.1 PURPOSE

The purpose of this test was to demonstrate the turbine-driven and motor-driven emergency feedwater pumps' ability to deliver adequate feedwater flow to the steam generators and to verify that the integrated control system (ICS) signal could be safely overridden to manually control emergency feedwater flow to the steam generator. Acceptance criteria for this test are listed below:

(1) Emergency feedwater flow can be controlled manually by overriding the integrated control system.

(2) Both the motor-driven and the steam-driven emergency feedwater pumps will deliver a minimum of 550 gallons per minute to either steam generator at a steam generator pressure of 1050 psig.

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4.14.2 TEST METHOD

With reactor power at approximately 5% full power, the steam-driven emergency feedwater pump is allowed to take over supplying feedwater flow to the "A" steam generator. With the "A" steam generator at a pressure of between 900 and 950 psig and a feedwater flow rate of approximately 740 gallons per minute, the emergency feedwater pump's discharge and suction pressure and the "A" steam generator's flow rate and steam pressure are recorded. From these parameters, an emergency feedwater pump flow rate at a steam pressure of 1050 psig was extrapolated.

The above procedure is repeated for the motor-driven emergency feedwater pump supplying feedwater flow to the "B" steam generator. In both these procedures, manual control of emergency feedwater flow independent of the integrated control system is demonstrated.

4.14.3 TEST RESULTS

In the Emergency Feedwater Flow Test, the extrapolated emergency feedwater flow at 1050 psig was 570 and 620 gallons per minute for the motor-driven and steam-driven emergency feedwater pumps, respectively. This satisfied the acceptance criterion of a minimum of 550 gallons per minute. In both procedures, manual control of emergency feedwater flow independent of the integrated control system was demonstrated, thereby satisfying the remaining acceptance criterion.

4.15 TURBINE/GENERATOR OPERATION

No Turbine/Generator operational testing was performed during this testing program since no changes were made to the Turbine/Generator which would invalidate the results of the tests conducted during initial startup.

4.16 DROPPED CONTROL ROD TEST

The dropped rod test was not performed because sufficient thermal margin exists as indicated by B&W-2 correlation analyses.

4.17 INCORE DETECTOR TEST

Incore detector output was verified by comparing the corrected detector response from similar core locations. All detector outputs were normalized to the average detector output per assembly. Table 4.17-1 shows the results of this comparison at 40% full power.

4.18 RCS HOT LEAKAGE TEST

RCS hot leakage is monitored on a regular basis during plant operation as required by Technical Specifications. No additional RCS leakage testing wa performed at this time.

4.19 PIPE AND COMPONENT HANGER HOT INSPECTION AT POWER

No pipe and component hanger hot inspection at power was done during this startup since no modifications had been made which would invalidate the results of the testing conducted at initial startup.

4.20 CHEMICAL AND RADIOCHEMICAL TESTS

Chemical and Radiochemical testing was not performed during this startup. These tests were conducted at initial startup and no plant modifications have been made which would invalidate the results of those tests.

4.21 EFFLUENT AND EFFLUENT MONITORING

No effluent or effluent monitoring testing was performed as these systems have been performing normally since initial startup and no furtner testing was required.

SUMMARY OF TEMPERATURE, MODERATOR, AND DOPPLER
COEFFICIENTS OF REACTIVITY

COEFFICIENT	VALUE	CONDITION
Temperature (10^{-4} $\Delta k/k/°F$)	-1.051	96% FP, 780 PPM, 6 EFPD
Moderator (10^{-4} $\Delta k/k/°F$)	-0.9009	96% FP, 780 PPM, 6 EFPD
Power Doppler (10^{-4} $\Delta k/k/\%FP$)	-1.248	94% FP, 780 PPM, 6 EFPD

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Table 4.1-1

Table 4.3-1

Power	Predicted Radial Peak (Location)	Measured Radial Peaks (Location)	% Difference *	Predicted Total Peak (Location)	Measured Total Peaks (Location)	% Difference *
40% FP	1.307 (K-[illegible])	1.394 (M-11)	-	1.595 (L-12)	1.702 (H-12)	-
		1.382 (H-12)	-		1.678 (L-12)	1.4
		1.369 (K-11)	1.8			
75% FP	1.301 (K-11)	1.388 (M-11)	-	1.606 (L-12)	1.706 (M-11)	-
		1.379 (H-12)	-		1.676 (L-12)	1.8
		1.365 (K-11)	1.7			
100% FP	1.287 (K-11)	1.379 (H-12)	-	1.575 (L-12)	1.627 (L-12)	-
		1.378 (M-11)	-			
		1.370 (K-11)	0.7			

* % difference between highest measured peak and the measured peak at the predicted peak location (P_L)

$$\frac{(P_{max} - P_L)}{P_L} \times 100\%$$

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Maximum Linear Heat Rate Per Incore Detector Level Versus Power Level

IC Level	Maximum Linear Heat Rate (KW/ft.) 40% FP	75% FP	100% FP	LOCA Limit (KW/ft.)
1	3.44	6.95	9.81	14.7
2	5.05	9.26	11.71	15.6
3	4.43	8.16	10.17	16.7
4	4.14	7.47	9.70	18.0
5	4.65	8.09	10.90	17.2
6	5.09	8.93	12.26	16.2
7	3.62	7.06	8.38	15.4

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Table 4.3-2

Table 4.12-1

SUMMARY OF TEST DATA
POWER IMBALANCE DETECTOR CORRELATION TEST
40% FP

Date Time	Power Level (% FP)	Rod Position (% WD) 1-5	6	7	8	Incore Offset(%) Full	Backup	Out-of-Core Offset (%) NI-5	NI-6	NI-7	NI-8	Worst Case Max. LHR (kw/ft)	Worst Case Min. DNBR
8/4/79 0326	41.1	100	84.6	12.4	37.8	-1.88	+1.86*	-2.72	-2.50	-2.93	-2.79	4.77	9.23
8/4/79 0401	41.15	100	84.6	12.4	39.0	-7.82	+2.40*	-10.04	-9.71	-10.00	-9.88	4.96	10.25
8/4/79 0545	41.1	100	84.6	12.4	43.0	-15.64	-----*	-19.79	-19.56	-19.65	-19.33	5.52	9.13
8/4/79 0633	41.2	100	84.6	12.4	45.0	-24.38	-22.22**	-30.37	-29.88	-29.91	-29.51	5.77	8.69
8/4/79 0203	41.2	100	84.6	12.4	33.8	+8.80	+4.09*	9.81	9.95	9.31	9.30	5.38	8.26
8/4/79 0010	41.3	100	84.6	12.4	27.5	+16.45	+11.07*	18.52	18.71	17.89	17.63	5.66	7.85
8/4/79 0720	41.2	100	NA	NA	NA	-12.14	-8.13**						
8/4/79	40.9	100	NA	NA	NA	+1.58	+4.89**						

* Backup recorders malfunctioned resulting in suspect data for backup offset values

** Backup recorders operational. These points will be used for comparison to acceptance criteria.

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SUMMARY OF LEAST SQUARES LINEAR REGRESSION ANALYSIS FOR EACH POWER RANGE CHANNEL AND BACKUP RECORDERS DURING THE POWER IMBALANCE CORRELATION TEST

Detector System	Gain Factor	Data Sets	Intercept Calculated	Intercept Allowable	Slope Calculated	Slope Allowable
Power Imbalance Test at 40% Full Power						
NI-5	4.5	6	-0.889	±3.5	1.201	>1.15
NI-6	4.5	6	-1.274	±3.5	1.194	>1.15
NI-7	4.5	6	-1.403	±3.5	1.174	>1.15
NI-8	4.5	6	-1.219	±3.5	1.15[illegible]	>1.15
Backup	---	3	3.654	±5.0	1.042	1.00±0.20

Table 4.12-2

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Table 4.17-1

Comparison of Incore Monitoring Assemblies Flux Shapes at 40% FP
Using the Indicated Grouping Given in Incore Detector Test

Group Number	Incore Detector Number	Fuel Assembly Location	Detector Current to Average Detector Current per Assembly							
			Level 1	Level 2	Level 3	Level 4	Level 5	Level 6	Level 7	(BG)
01	05	E-09	0.893	1.131	1.058	0.940	1.073	1.182	0.722	
	07	E-07	0.842	1.074	1.056	0.927	1.085	1.266	0.751	0.149
	09	G-05	0.865	1.091	1.025	0.913	1.094	1.220	0.792	0.162
	11	K-05	0.865	1.060	1.065	0.922	1.019	1.281	0.788	0.171
	13	M-07	0.859	1.093	1.006	0.894	1.097	1.256	0.804	0.131
	16	M-09	0.864	[illegible]	1.095	[illegible]	1.045	1.253	0.750	0.127
	19	K-11	[illegible]	[illegible]	1.090	[illegible]	1.084	1.179	0.775	0.151
	25	G-11	0.824	[illegible]	1.060	[illegible]	1.050	1.229	0.895	0.152
02	23	F-13	[illegible]	[illegible]	1.079	[illegible]	1.036	1.124	0.942	0.178
	28	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.114
	32	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	[illegible]	[illegible]	[illegible]	[illegible]	.092	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	43	[illegible]	[illegible]	[illegible]	.084	[illegible]	[illegible]	[illegible]	0.862	0.160
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
03	15	N-09	[illegible]	[illegible]	1.057	[illegible]	1.044	[illegible]	0.855	0.144
	17	M-10	0.926	.144	1.044	0.836	1.016	1.212	0.822	0.148
	18	L-11	0.813	.146	1.071	0.845	0.981	1.219	0.824	0.152
	20	K-12	0.831	.135	1.045	0.853	[illegible]	1.221	0.875	0.150
	33	D-05	[illegible]	.167	1.075	[illegible]	[illegible]	1.212	0.726	0.171
	34	E-04	0.829	.175	1.047	[illegible]	1.058	1.250	0.758	0.161
										0.147
04	24	F-12	0.907	1.275	1.177	0.622	0.914	1.249	0.856	0.135
	27	D-10	0.906	.232	1.093	0.598	0.958	1.332	0.881	0.149

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Table 4.17-1 (Cont'd)

Comparison of Incore Monitoring Assemblies Flux Shapes at 40% FP
Using the Indicated Grouping Given in Incore Detector Test

Group Number	Incore Detector Number	Fuel Assembly Location	Detector Current to Average Detector Current per Assembly							
			Level 1	Level 2	Level 3	Level 4	Level 5	Level 6	Level 7	(BG)
05	01	H-08	0.743	1.057	1.007	1.005	1.177	1.226	0.735	0.134
	38	L-02	0.742	1.104	1.109	0.99	1.137	1.164	0.744	0.144
	44	P-06	0.771	1.133	1.105	1.014	1.118	1.178	0.682	0.138
06	14	N-08	0.792	1.036	1.049	0.917	1.078	1.224	0.903	0.145
	26	E-11	0.852	1.121	1.102	0.841	1.054	1.208	0.812	0.108
07	03	G-09	0.880	0.988	1.103	0.909	1.096	1.278	0.746	0.124
	04	F-08	0.873	1.126	1.172	1.044	1.223	1.141	0.420	0.121
	06	F-07	0.938	1.018	1.069	0.972	1.082	1.249	0.672	0.154
	08	G-06	0.917	1.073	1.080	0.977	1.096	1.175	0.681	0.160
	12	L-06	1.286	1.039	0.934	0.900	0.978	1.155	0.709	0.157
	30	B-08	1.225	1.025	1.015	0.900	0.990	1.095	0.750	0.190
08	10	H-05	0.849	1.126	1.080	0.932	1.110	1.164	0.739	0.122
	21	H-13	0.809	1.040	1.055	0.953	1.068	1.189	0.885	0.161
09	22	G-13	0.809	1.091	1.046	0.929	1.044	1.216	0.866	0.164
	29	G-09	0.809	1.109	1.052	0.879	1.05a2	1.235	0.864	0.162
10	31	B-07	0.839	1.085	1.086	1.003	1.068	1.104	0.815	0.137
	36	G-02	0.899	1.158	0.519	1.031	1.162	1.334	0.896	0.175
	46	R-10	0.742	1.015	1.090	1.076	1.154	1.179	0.743	0.142
	49	M-14	0.752	1.124	1.067	1.057	1.115	1.171	0.714	0.107
	02	H-09	0.747	1.041	1.061	1.031	1.165	1.248	0.707	0.154
11	37	H-01	0.845	1.064	1.068	1.060	1.075	1.179	0.709	0.128
	45	R-07	0.803	1.133	1.088	1.093	1.113	1.208	0.561	0.110
	48	O-12	0.780	1.123	1.246	1.055	1.042	1.183	0.571	0.139
	51	D-14	0.722	1.122	1.161	1.104	1.133	1.204	0.554	0.154
	52	C-13	0.726	1.155	1.27	1.101	1.156	1.193	0.54[illegible]	0.137

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Table 4.17-1 (Cont'd)

Comparison of Incore Monitoring Assemblies Flux Shapes at 40% FP
Using the Indicated Grouping Given in Incore Detector Test

Group Number	Incore Detector Number	Fuel Assembly Location	Detector Current to Average Detector Current per Assembly							
			Level 1	Level 2	Level 3	Level 4	Level 5	Level 6	Level 7	(BG)
12	40	M-03	0.768	1.130	1.074	0.933	1.103	1.239	0.753	0.171
	41	N-04	0.816	1.186	1.172	1.042	1.159	1.202	0.423	0.099
	42	O-05	0.827	1.180	1.081	0.965	1.072	1.198	0.678	0.144

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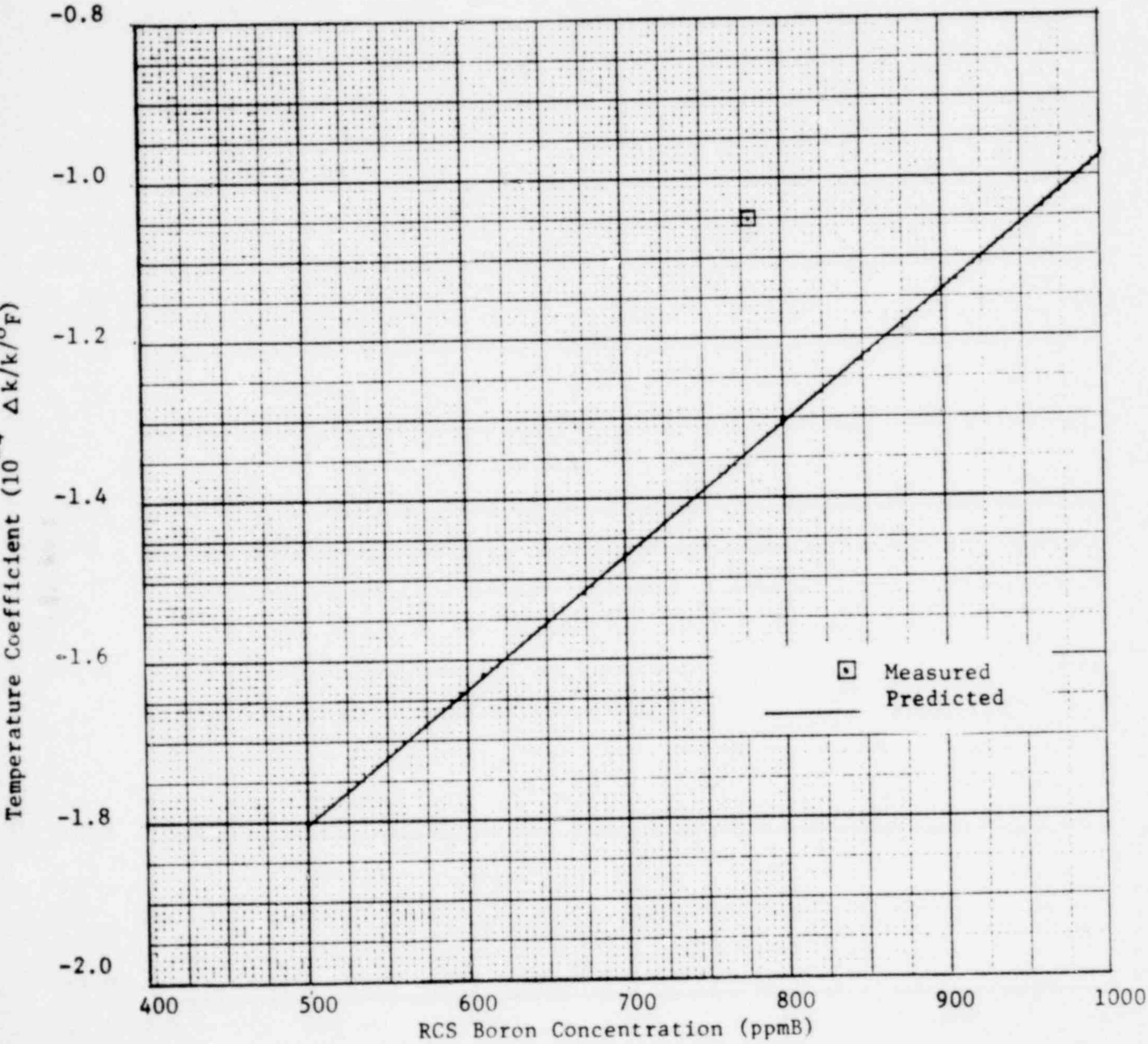
Temperature Coefficient of Reactivity versus Boron Concentration
Cycle 2, 4 EFPD, HFP



POOR ORIGINAL

Figure 4.1-1

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Moderator Coefficient of Reactivity versus Boron Concentration
Cycle 2, 4EFPD, HFP

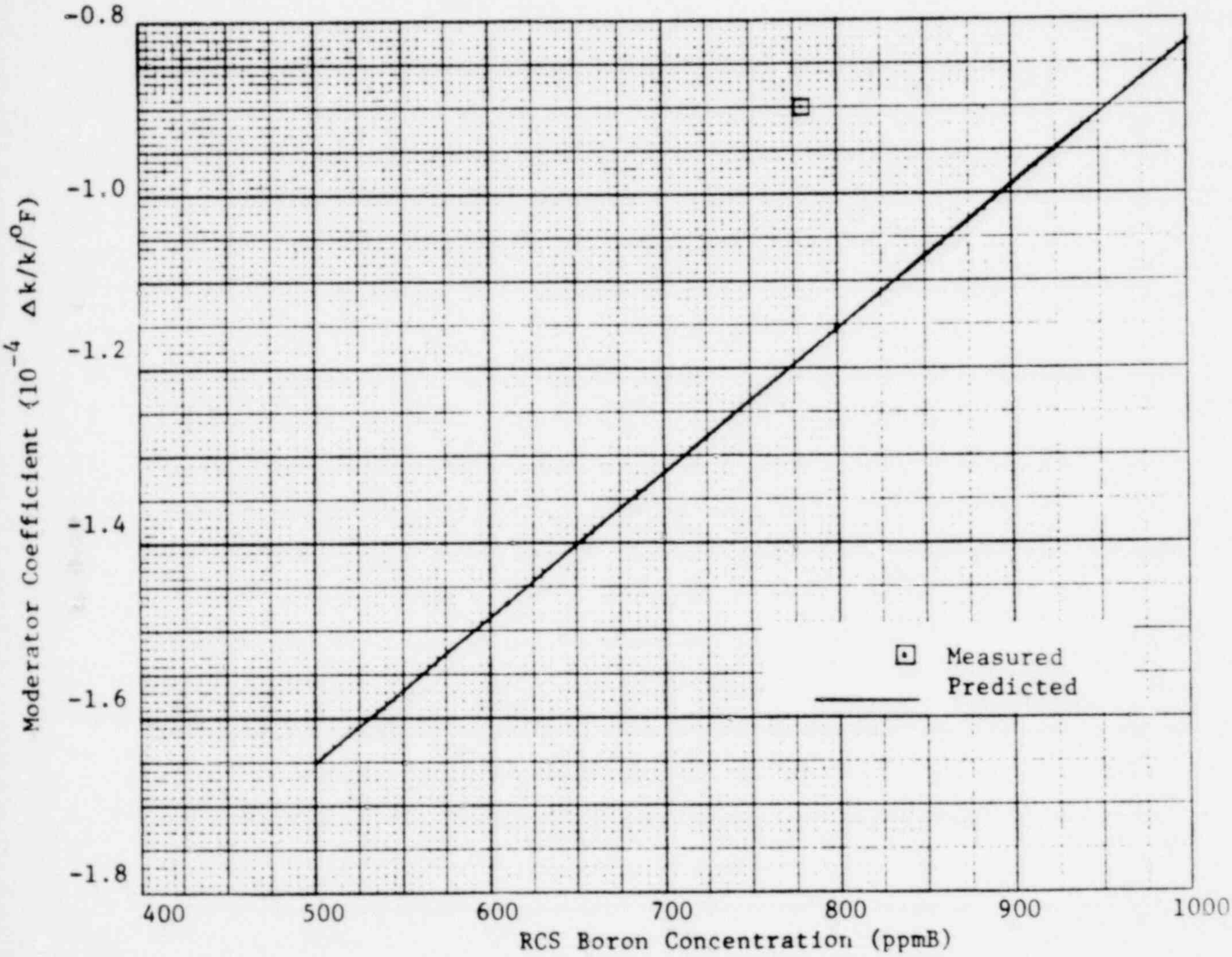

Figure 4.1-2

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Power Doppler Coefficient of Reactivity versus Power Level
Cycle 2, 4 EFPD, HFP

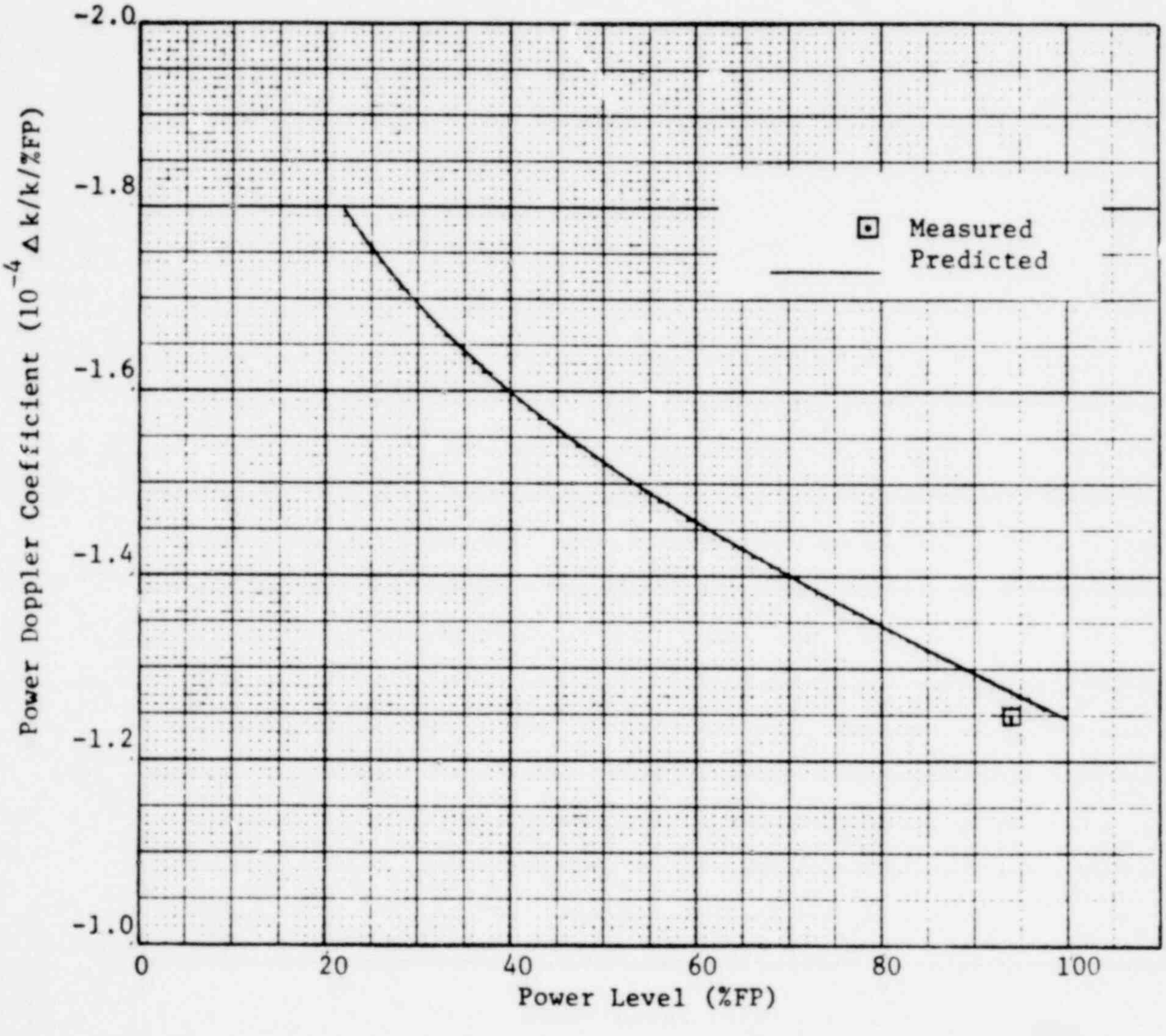

Figure 4.1-3

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Reactor Protective System Maximum Allowable Envelope
for Four Pump Operation

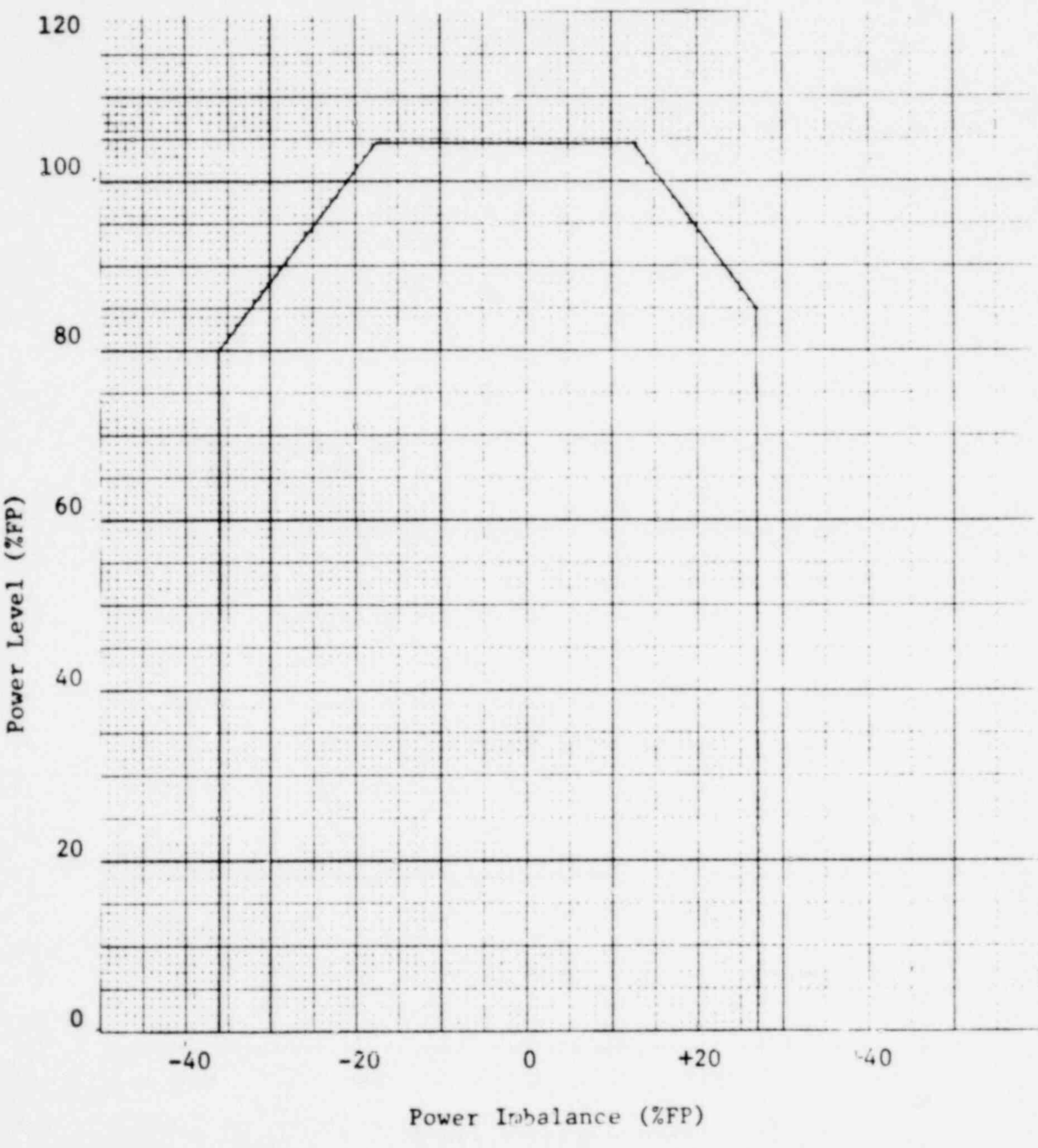

Figure 4.3-1

Comparison of Predicted and Measured Radial Peaking Core Power Distribution Results at Steady State, 3-D Equilibrium Xenon, 40% FP Conditions

Predicted Conditions

Control Rod Group Positions			Core Parameter	Value	Unit
Gps 1-5	100.0	% wd	Core Power Level	40.0	%FP
Gp 6	87.1	% wd	Boron Concentration	NA	ppmB
Gp 7	12.6	% wd	Core Burnup	2.00	EFPD
Gp 8	35.3	% wd	Axial Imbalance	-0.53	%FP
			Max Quadrant Tilt	0.0	%

Measured Conditions

Control Rod Group Positions			Core Parameter	Value	Unit
Gps 1-5	100.0	% wd	Core Power Level	41.3	%FP
Gp 6	84.2	% wd	Boron Concentration	930	ppmB
Gp 7	11.4	% wd	Core Burnup	0.86	EFPD
Gp 8	35.8	% wd	Axial Imbalance	+0.15	%FP
			Max Quadrant Tilt	1.36	%

	H	K	L	M	N	O	P	R
8	0.904 0.847 6.7							
9	0.870 0.919 -5.3	0.903 0.874 3.3						
10	1.011 0.980 3.2	1.069 1.152 -7.2	0.667 0.755 -11.7					
11	1.116 1.172 -4.8	1.307 1.369 -4.5	1.021 1.096 -6.8	1.280 1.394 -8.2				
12	1.250 1.382 -9.6	1.282 1.316 -2.6	1.199 1.271 -5.7	1.219 1.227 -0.7	1.066 0.998 6.8			
13	0.929 0.975 -4.7	1.119 1.161 -3.6	1.065 1.050 1.4	1.018 0.952 6.9	1.242 1.115 11.3	0.865 0.733 18.0		
14	0.503 0.532 -5.5	0.825 0.770 7.4	1.172 1.116 5.0	0.989 0.964 2.6	0.730 0.628 16.2			
15	0.579 0.570 1.6	0.637 0.616 3.4	0.604 0.588 2.7					

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X.XXX	Predicted Results
X.XXX	Measured Results
XX.X	Deviation $\frac{(\text{Predicted} - \text{Measured})}{\text{Measured}} \times 100\%$

Figure 4.3-2

Comparison of Predicted and Measured Total Peaking Core Power Distribution Results at Steady State, 3-D Equilibrium Xenon, 40% FP Conditions

Predicted Conditions

Control Rod Group Positions			Core Power Level	40.0	%FP
Gps 1-5	100.0	% wd	Boron Concentration	NA	ppmB
Gp 6	87.1	% wd	Core Burnup	2.00	EFPD
Gp 7	12.6	% wd	Axial Imbalance	-0.53	%FP
Gp 8	35.3	% wd	Max Quadrant Tilt	0.0	%

Measured Conditions

Control Rod Group Positions			Core Power Level	41.3	%FP
Gps 1-5	100.0	% wd	Boron Concentration	930	ppmB
Gp 6	84.2	% wd	Core Burnup	0.86	EFPD
Gp 7	11.4	% wd	Axial Imbalance	+0.15	%FP
Gp 8	35.8	% wd	Max Quadrant Tilt	1.36	%

	H	K	L	M	N	O	P	R
8	1.070 1.041 2.8							
9	1.021 1.126 -9.3	1.042 1.065 -2.2						
10	1.220 1.139 7.1	1.245 1.334 -6.7	0.831 0.881 -5.7					
11	1.317 1.371 03.9	1.542 1.651 -6.6	1.247 1.310 -4.8	1.544 1.665 -7.3				
12	1.511 1.702 -11.2	1.567 1.610 -2.7	1.595 1.678 -5.0	1.476 1.500 -1.6	1.297 1.212 7.0			
13	1.103 1.175 -6.1	1.333 1.414 -5.7	1.277 1.270 0.6	1.190 1.151 3.4	1.479 1.334 10.9	1.037 0.881 17.7		
14	0.575 0.600 -4.2	0.951 0.955 -0.4	1.380 1.298 6.3	1.160 1.114 4.1	0.868 0.747 16.2			
15	0.672 0.649 3.5	0.745 0.735 1.4	0.702 0.686 2.3					

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X.XXX	Predicted Results
X.XXX	Measured Results
XX.X	Deviation (Predicted - Measured) / Measured x 100%

Figure 4.3-3

Comparison of Predicted and Measured Radial Peaking Core Power Distribution
Results at Steady State, 3-D Equilibrium Xenon, 75% FP Conditions

Predicted Conditions

Control Rod Group Positions				
Gps 1-5	100.0 % wd		Core Power Level	75.0 %FP
Gp 6	87.0 % wd		**Boron** Concentration	NA ppmB
Gp 7	13.0 % wd		Core Burnup	3.0 EFPD
Gp 8	32.0 % wd		Axial Imbalance	0.4 %FP
			Max Quadrant Tilt	0.0 %

Measured Conditions

Control Rod Group Positions				
Gps 1-5	100.0 % wd		Core Power Level	75.31 %FP
Gp 6	85.6 % wd		Boron Concentration	826 ppmB
Gp 7	11.7 % wd		Core Burnup	2.4 EFPD
Gp 8	32.5 % wd		Axial Imbalance	-3.35 %FP
			Max Quadrant Tilt	1.33 %

	H	K	L	M	N	O	P	R
8	0.955 0.864 10.6							
9	0.916 0.941 -2.7	0.942 0.890 5.9						
10	1.041 0.987 5.5	1.091 1.154 -5.5	0.685 0.768 -10.8					
11	1.134 1.169 -3.0	1.301 1.365 -4.7	1.031 1.095 -5.9	1.170 1.388 -15.7				
12	1.253 1.379 -9.1	1.280 1.309 -2.2	1.191 1.257 -5.3	1.210 1.213 -0.2	1.058 1.001 5.7			
13	0.944 0.956 -1.4	1.122 1.162 -3.4	1.067 1.048 1.8	1.041 0.948 .9	1.203 1.114 8.0	0.904 0.736 27.6		
14	0.514 0.541 05.0	0.833 0.775 7.4	1.148 1.109 3.5	0.967 0.963 0.4	0.714 0.631 13.1			
15	0.582 0.583 -0.2	0.645 0.623 3.5	0.601 0.600 -0.2					

1273 252

X.XXX	**Predicted Results**
X.XXX	**Measured Results**
XX.X	Deviation $\frac{\text{(Predicted - Measured)}}{\text{Measured}} \times 100\%$

Figure 4.3-4

Comparison of Predicted and Measured Total Peaking Core Power Distribution Results at Steady State, 3-D Equilibrium Xenon, 75% FP Conditions

Predicted Conditions

Control Rod Group Positions					
Gps 1-5	100.0	% wd	Core Power Level	75.0	%FP
Gp 6	87.0	% wd	Boron Concentration	NA	ppmB
Gp 7	13.0	% wd	Core Burnup	3.0	EFPD
Gp 8	32.0	% wd	Axial Imbalance	0.4	%FP
			Max Quadrant Tilt	0.0	%

Measured Conditions

Control Rod Group Positions					
Gps 1-5	100.0	% wd	Core Power Level	75.31	%FP
Gp 6	85.6	% wd	Boron Concentration	826	ppmB
Gp 7	11.7	% wd	Core Burnup	2.4	EFPD
Gp 8	32.5	% wd	Axial Imbalance	-3.35	%FP
			Max Quadrant Tilt	1.33	%

	H	K	L	M	N	O	P	R
8	1.164 1.005 15.8							
9	1.101 1.093 0.7	1.111 1.032 7.7						
10	1.278 1.221 4.7	1.288 1.359 -5.2	0.842 0.991 -15.0					
11	1.356 1.430 -5.2	1.556 1.688 -4.7	1.253 1.362 -8.0	1.532 1.706 -10.2				
12	1.527 1.592 -4.1	1.578 1.585 -0.4	1.606 1.676 -4.2	1.482 1.580 -6.2	1.258 1.281 -1.8			
13	1.136 1.126 0.9	1.348 1.379 -2.2	1.287 1.274 1.0	1.183 1.234 -4.1	1.427 1.416 0.8	0.990 0.911 8.7		
14	0.588 0.661 -11.0	0.967 0.951 1.7	1.348 1.342 0.4	1.179 1.223 -3.6	0.835 0.776 7.6			
15	0.672 0.681 -1.3	0.741 0.762 -2.8	0.695 0.681 2.1					

1273 253

X.XXX	Predicted Results
X.XXX	Measured Results
XX.X	Deviation $\frac{(\text{Predicted - Measured})}{\text{Measured}} \times 100\%$

Figure 4.3-5

Comparison of Predicted and Measured Radial Peaking Core Power Distribution
Results at Steady State, 3-D Equilibrium Xenon, 100% FP Conditions

Predicted Conditions

Control Rod Group Positions			Core Power Level	100.0	%FP
Gps 1-5	100.0	% wd	**Boron** Concentration	NA	ppmB
Gp 6	87.0	% wd	Core Burnup	6	EFPD
Gp 7	13.0	% wd	Axial Imbalance	-2.83	%FP
Gp 8	32.0	% wd	Max Quadrant Tilt	0.0	%

Measured Conditions

Control Rod Group Positions			Core Power Level	99.85	%FP
Gps 1-5	100.0	% wd	Boron Concentration	778	ppmB
Gp 6	90.0	% wd	Core Burnup	4.35	EFPD
Gp 7	15.0	% wd	Axial Imbalance	-2.10	%FP
Gp 8	28.0	% wd	Max Quadrant Tilt	1.21	%

	H	K	L	M	N	O	P	R
8	0.957 0.890 7.6							
9	0.918 0.968 -5.2	0.944 0.910 3.7						
10	1.039 1.002 3.7	1.086 1.123 -3.3	0.689 0.791 12.9					
11	1.127 1.117 0.9	1.287 1.370 -6.1	1.027 1.096 -6.3	1.259 1.378 -8.6				
12	1.241 1.379 -10.0	1.267 1.304 -2.8	1.181 1.229 -3.9	1.203 1.203 0.0	1.060 0.996 6.4			
13	0.945 0.981 3.7	1.119 1.160 -3.5	1.067 1.038 2.7	1.016 0.943 7.8	1.202 1.103 9.0	0.847 .730 16.0		
14	0.524 0.551 5.0	0.841 0.779 7.9	1.150 1.103 4.2	0.972 0.953 2.0	0.723 0.624 15.9			
15	0.596 0.589 1.2	0.658 0.624 5.5	0.613 0.601 2.0					

X.XXX	Predicted Results
X.XXX	Measured Results
XX.X	Deviation $\frac{\text{(Predicted - Measured)}}{\text{Measured}} \times 100\%$

1273 254

Figure 4.3-6

Comparison of Predicted and Measured Total Peaking Core Power Distribution Results at Steady State, 3-D Equilibrium Xenon, 100% FP Conditions

Predicted Conditions

Control Rod Group Positions					
Gps 1-5	100.0	% wd	Core Power Level	100.0	%FP
Gp 6	87.0	% wd	Boron Concentration	NA	ppmB
Gp 7	13.0	% wd	Core Burnup	6.0	EFPD
Gp 8	32.0	% wd	Axial Imbalance	-2.83	%FP
			Max Quadrant Tilt	0.0	%

Measured Conditions

Control Rod Group Positions					
Gps 1-5	100.0	% wd	Core Power Level	99.85	%FP
Gp 6	90.0	% wd	Boron Concentration	778	ppmB
Gp 7	15.0	% wd	Core Burnup	4.35	EFPD
Gp 8	28.0	% wd	Axial Imbalance	-2.10	%FP
			Max Quadrant Tilt	1.21	%

	H	K	L	M	N	O	P	R
8	1.129 1.063 6.2							
9	1.068 1.154 -7.5	1.081 1.083 0.2						
10	1.225 1.204 1.8	1.269 1.352 -6.1	0.883 1.063 -16.9					
11	1.328 1.392 -4.6	1.538 1.612 -4.6	1.260 1.296 -2.8	1.537 1.609 4.5				
12	1.454 1.624 10.5	1.527 1.544 -1.1	1.575 1.627 3.2	1.466 1.450 1.1	1.308 1.204 8.7			
13	1.093 1.128 -31.	1.316 1.359 -312	1.302 1.233 5.6	1.211 1.129 6.8	1.489 1.336 11.5	1.037 0.865 19.9		
14	0.626 0.708 -11.5	0.977 0.952 2.6	1.408 1.270 10.9	1.184 1.088 8.8	0.878 0.739 18.7			
15	0.712 0.683 4.3	0.784 0.754 4.0	0.735 0.673 9.2					

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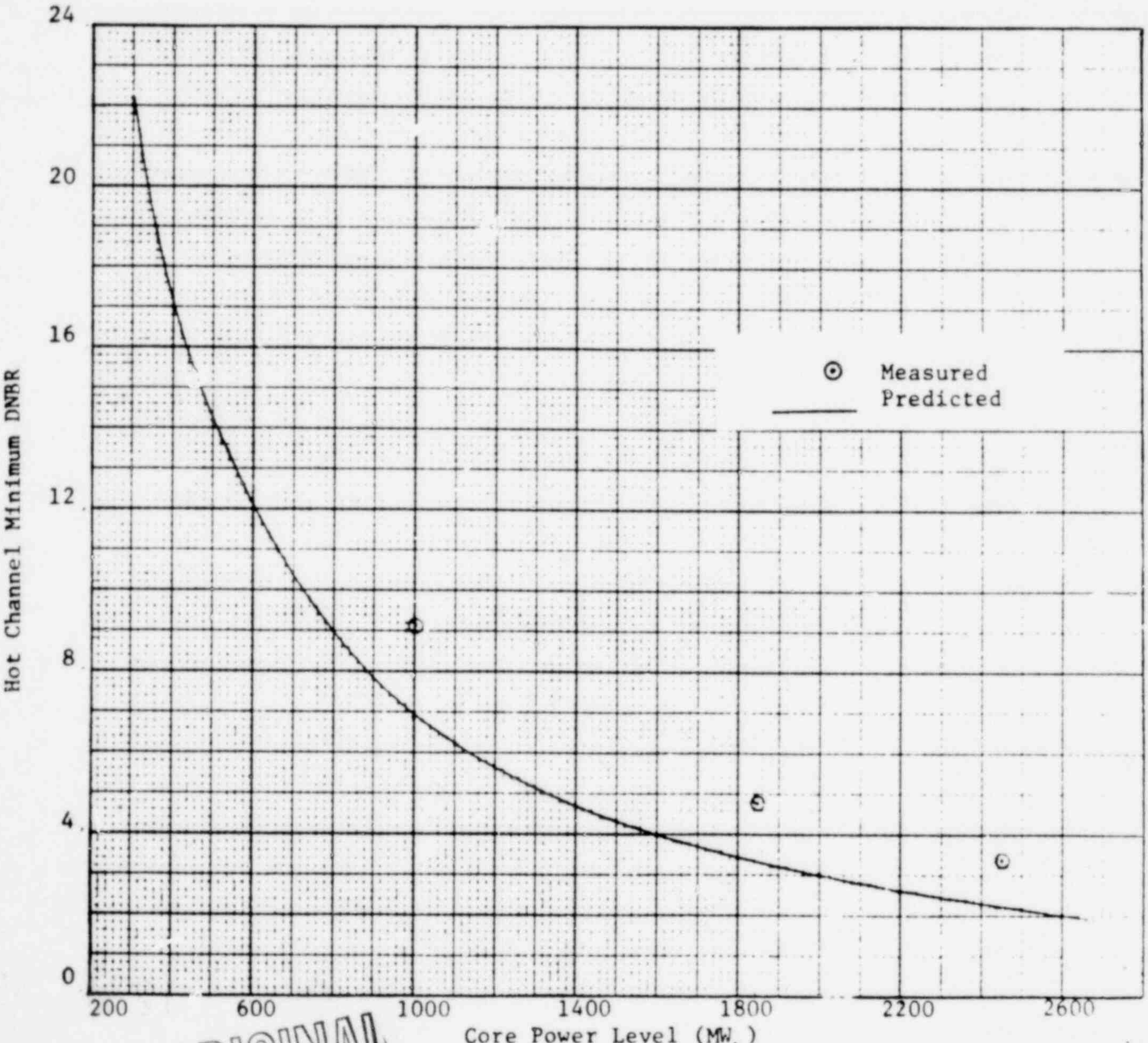
X.XXX	Predicted Results
X.XXX	Measured Results
XX.X	Deviation $\frac{(\text{Predicted} - \text{Measured})}{\text{Measured}} \times 100\%$

Figure 4.3-7

Hot Channel Minimum DNBR versus Core Power Level

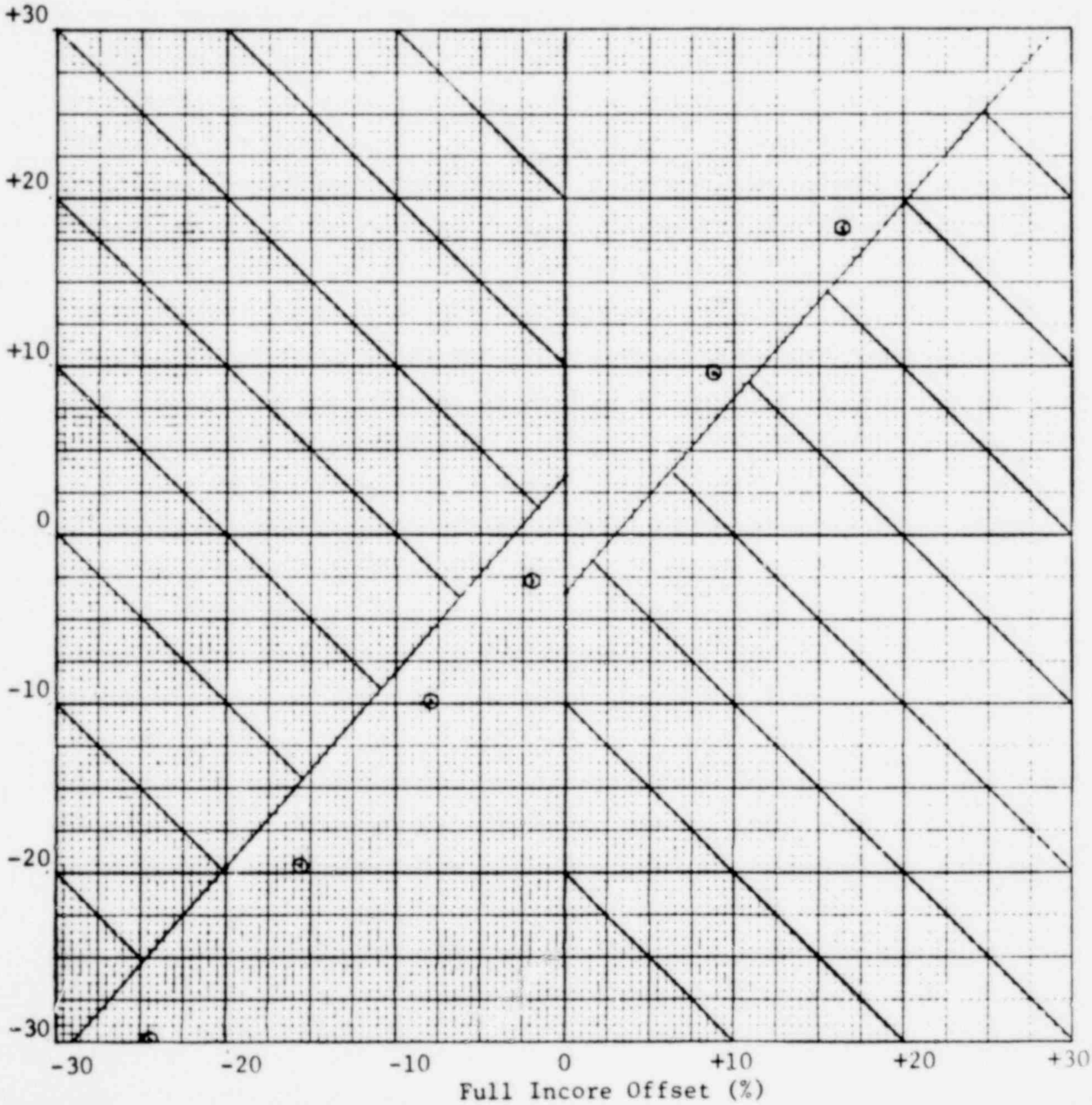


POOR ORIGINAL

Figure 4.3-8

1273 256

Average Out-of-Core Offset versus Full Incore Offset
Cycle 2, 40% FP



POOR ORIGINAL

Figure 4.12-1

1273 257

Backup Incore Offset versus Full Incore Offset
Cycle 2, 40% FP

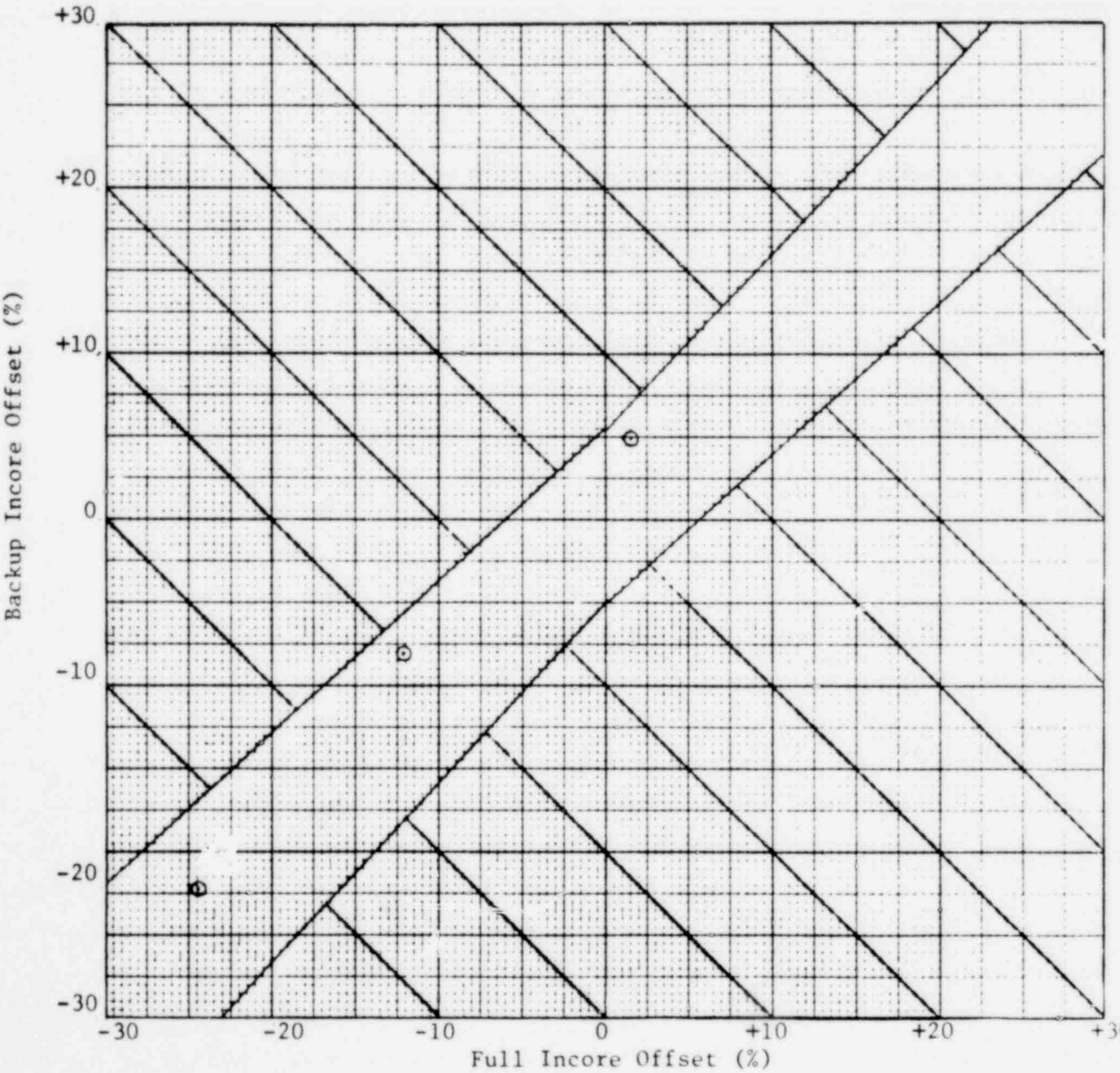
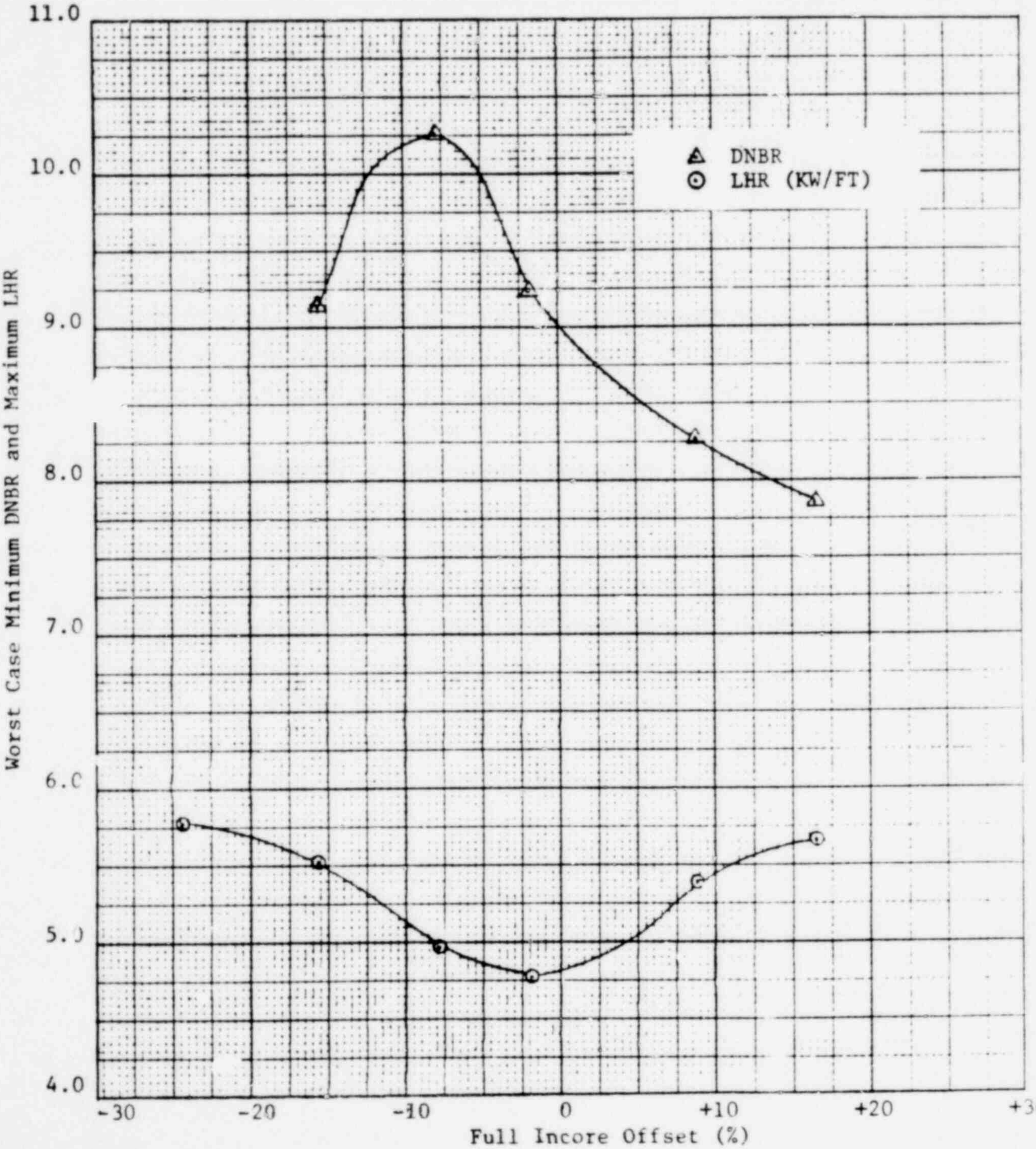


Figure 4.12-2

Worst Case Minimum DNBR and Maximum LHR
Versus Full Incore Offset
Cycle 2, 40% FP



POOR ORIGINAL

Figure 4.12-3

1273 259