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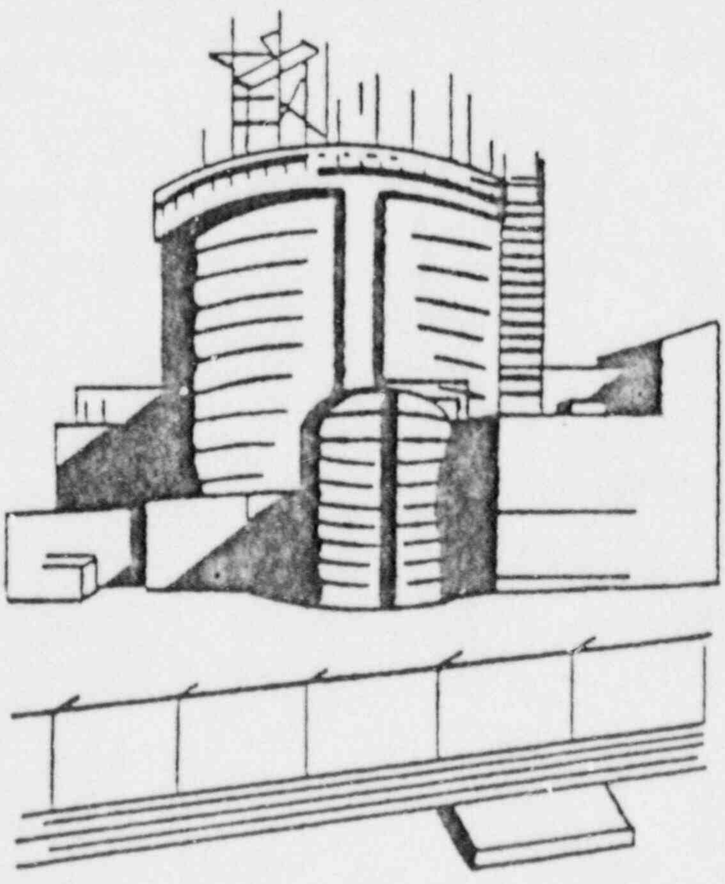
Crystal River Unit 3

CYCLE 7 STARTUP REPORT

APRIL 1988



Florida Power CORPORATION

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1.0 INTRODUCTION

On September 18, 1987, Crystal River Unit 3 was shut down to refuel for Cycle 7. The Cycle 6 fuel had attained a core average exposure of 412.07 effective full power days (EFPD).

Cycle 7 has been designed for a cycle lifetime of 550 ± 10 EFPD and operation in a feed-and-bleed mode. The cycle utilizes burnable poison rod assemblies and gray axial power shaping rods (APSRs). The final Cycle 7 core loading pattern is shown in Figure 1.0-1. The fuel assembly and control component identifications for Cycle 7 are shown in Figure 1.0-2.

This report, prepared and submitted in accordance with Technical Specification 6.9.1, describes the precritical, zero power, and power testing performed during the Cycle 7 start up and also summarizes the results of these tests.

Crystal River Unit 3 (CR3) was brought to Mode 3 (Hot Standby) for precritical testing. The results of these tests are reported in Section 2.0. Zero power physics tests began after criticality was achieved on January 8, 1988 at 2042 hours. The zero power physics testing results are given in Section 3.0. Power escalation testing began after breaker closure on January 10, 1988 at 1034 hours. These test results are summarized in Section 4.0. The all-rods-in (ARI) temperature coefficient, the hot zero power (HZP) ejected rod worth test, and one intermediate power distribution test plateau were not performed during Cycle 7 physics testing in accordance with a program for improving plant availability through the elimination of unnecessary tests.[1]

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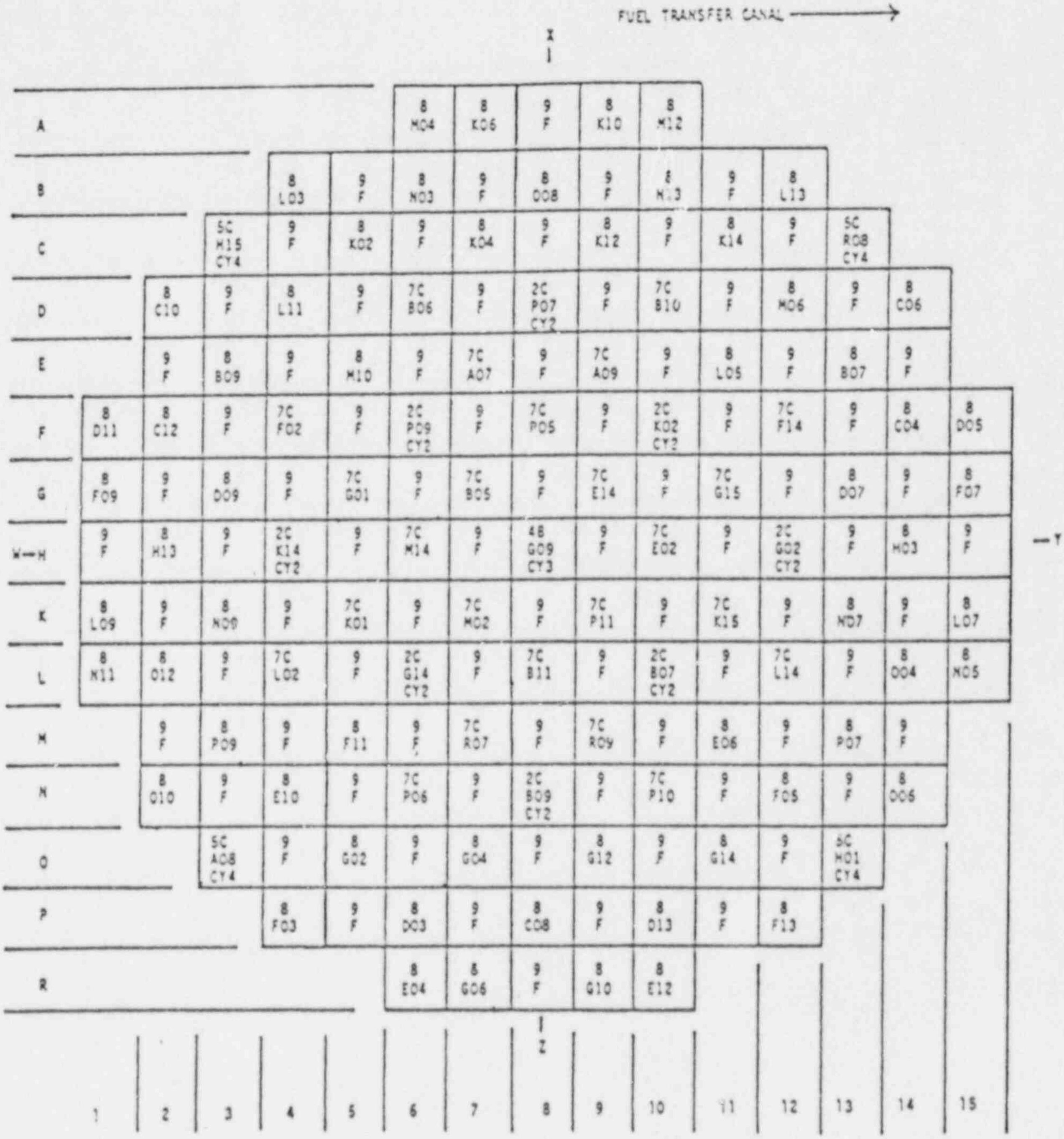
[1] "Reduced Physics Testing, Task Summary Report" document by Babcock and Wilcox, B&W document number 86-1164722-00, dated March 1987 for B&W Owners' Group Performance Committee.

CORE LOADING DIAGRAM, CYCLE 7



Source: B&W Cy 7 Fuel
Design Report

XX Batch Number
YY Previous Cycle Location
ZZ Cycle Out For Reinsertion

FIGURE 1.0-1

FUEL ASSEMBLY and CONTROL ASSEMBLY COMPONENT IDENTIFICATIONS

FUEL TRANSFER CANAL ⟶

Z

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						3UQ	3UR	48B	3TS	3TN					
B				3TM	497	3V7 C042	48J B5A8 RTXX	3TM C007	494 B5AA RTXX	3U4 C032	489 R014 LXXX	3TJ			
C			204	48X B5A0 RTXX	3UM C039	49H B5B8 RTXX	3UT C20Y	4A3 B5B0 RTXX	3UH C20R	4A7 B5BB RTXX	3V6 C036	48F B5A1 RTXX	217		
D		3UB	490 B59Z RTXX	3UV C056	4A6 B5BV RTXX	2YG A062	49A B5AJ RTXX	3B45 C054	49W B5AY RTXX	2YH A05V	490 B5BS RTXX	3US C022	48B B5A2 RTXX	3TK	
E		487	3UN C049	49B B5BT RTXX	3UZ C035	49K B5BE RTXX	2ZG C040	48P B5AK RTXX	2ZJ C047	48U B5BL RTXX	3UU C005	49E B5BX RTXX	3V5 C048	49B	
F	3TT	3V3 C019	49J B5BA RTXX	306 A061	49L B5BG RTXX	3B27 C053	48H B5AS RTXX	2Z1 C016	4AA B5AQ RTXX	3B54 C029	49G B5BJ RTXX	2YY A05W	49T B5AU RTXX	3U9 C018	3UK
G	3TX	48D B5A8 RTXX	3UJ C20X	49S B5AM RTXX	2ZH C050	48G B5BC RTXX	2Z5 C060	4A0 B5B2 RTXX	2YV C028	493 B5AW RTXX	30B C037	4A0 B5B1 RTXX	3TZ C205	48C B5A9 RTXX	3TV
W — H	485	3TL C011	4A2 B5BP RTXX	3B56 C010	48N B5AN RTXX	2ZU C004	4A9 B5B3 RTXX	1BP	48L B5B9 RTXX	2Z3 C058	48T B5AL RTXX	3B68 C052	4A4 B5BN RTXX	3V0 C051	484 — Y
K	3TP	48Y B5A7 RTXX	3UD C20W	49P B5B3 RTXX	2ZK C025	4AB B5AG RTXX	2ZY C034	491 B5AZ RTXX	2Z9 C002	4A8 B5AR RTXX	2YT C012	49Z B5AV RTXX	3UG C20T	499 B5AC RTXX	3U0
L	3TQ	3UA C044	49Q B5B7 RTXX	2YE A060	48S B5B0 RTXX	3B38 C033	492 B5AT RTXX	2ZE C046	4AC B5AH RTXX	3B04 C009	48Q B5BF RTXX	2YK A05X	49M B5AX RTXX	3U5 C043	3TR
M		486	3V1 C014	49Y B5BW RTXX	3TY C057	49N B5BH RTXX	2YQ C015	48R B5AP RTXX	2ZR C024	48M B5BK RTXX	3UY C027	49C B5BY RTXX	3V4 C013	482	
N		3TG	48K B5A6 RTXX	3UP C038	4A1 B5BU RTXX	2YD A052	49V B5B4 RTXX	3B40 C008	49U B5AF RTXX	2YF A05Y	49X B5BR RTXX	3UW C006	48Z B5A3 RTXX	3U6	
O			216	48E B5A5 RTXX	3U2 C026	49F B5B6 RTXX	3UE C20V	4A5 B5BM RTXX	3UF C20U	49R B5B5 RTXX	3U3 C023	48W B5A4 RTXX	215		
P				3UL	496 R013 LXXX	3U8 C030	495 B5AD RTXX	3TW C055	48V B5AE RTXX	3V2 C020	48A	3U7			
R						3TU	3UX	483	3U1	3UC					

Z

Fuel ID (3B**=Batch 2; 1BP=Batch 4; 204-217=Batch 5; 2Y*, 2Z* & 30*=Batch 7;
3T*, 3U* & 3V*=Batch 8; 48*, 49* & 4A*=Batch 9)
Control Comp. ID(C=CRA; A=APSRA; B=BPRA) or Source ID(R=Regenerative)
Retainers (L for Source; RT for BPRA)

Source: B&W Cy 7 Fuel Design Report

Note: NJO prefixes all but the Batch 2 Fuel Assembly ID Numbers.

FIGURE 1.0-2

2.0 PRECRITICAL TESTING

During all precritical testing required by Section 13.4 Table 13-2 of the CR3 Final Safety Analysis Report (FSAR), the actual reactor power was maintained at a subcritical level. The results of the precritical tests are summarized in Table 2.0-1.

2.1 Calibration and Neutron Response of Source Range Monitoring System

Precritical testing began following response checkout of the neutron source range detectors. These detectors are two BF_3 proportional counters which measure flux from the subcritical to start up level. This test verified that the source range instrumentation provided a count rate of more than 2 counts/sec. These tests were completed in accordance with Performance Testing Procedure PT-100.

2.2 Reactor Coolant Flow Test

The Reactor Coolant Flow test was performed to measure reactor coolant (RC) flow. The results were compared with design calculations to verify adequate core coolant flow. The test was executed in accordance with Surveillance Procedure SP-224.

The acceptance criteria of the test are as follows:[1]

Steady state total reactor coolant system (RCS) flow (three or four pump operation) at 532±2 °F and 2155±100 psig shall be within the following limits:

4 RCS Pumps, Flow $\geq 139.7 \times 10^6\ lb_m/hr$
3 RCS Pumps, Flow $\geq 104.4 \times 10^6\ lb_m/hr$

Additionally, with all 4 reactor coolant pumps in operation, steady state loop flows shall be within 2% of each other. To account for measurement uncertainty, the following must be true:

Measured Flow x 1.025 $\leq$ 411,840 gpm (Max. Flow)
Measured Flow x 0.975 $\geq$ 374,880 gpm (Min. Flow)

2.2.1 Method

During this test, the RCS temperature was maintained at 532±2°F with a pressure of 2155±100 psig.

The steady state temperature, pressure, and flow of the two coolant flow loops were taken every minute for ten minutes. This data was then used to calculate the average reactor coolant flow for each flow loop and for the entire system.

2.2.2 Results

The total RC flow was calculated to be $153.765 \times 10^6\ lb_m/hr$. The flow difference between loops was calculated to be 0.661%. The calculation for measurement uncertainty was 411,077 gpm for the maximum flow and 391,024 gpm for the minimum flow. Therefore, the acceptance criteria was met.

[1] SP-224, Section 4, Rev. 4

2.3 Control Rod Drop Time Tests

The Control Rod Drop Time tests demonstrated that the drop times are in accordance with Technical Specifications requirements. The acceptance criteria for these tests, per Surveillance Procedure SP-102, are:[1]

> The individual safety and regulating rod drop times from the fully withdrawn position shall be $\leq$ 1.66 seconds from power interruption at the control rod drive breakers to three-fourths insertion, 25% position, with $T_{avg} \geq$ 525°F and either 3 or 4 reactor coolant pumps operating.

In order to check for uncoupled control rods, the individual rod drop times were compared to their respective group average drop times. The individual rod drop time should not exceed its group average drop time by more than 0.05 sec.

2.3.1 Method

The Control Rod Drive (CRD) Drop Time tests were 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 chart recorders for each control rod assembly from a contact on the manual trip switch. A second signal to indicate three-fourths insertion was furnished by a reed switch located on the position indicator tube of each CRD.

2.3.2 Results

The results of the Control Rod Drop Time tests are presented in Table 2.3-1. The average drop time by group is summarized in Table 2.3-2.

By examination of Table 2.3-1, it is found that the shortest drop time is 1.26 seconds for CR 5-5 and CR 7-6. The longest drop time was 1.33 seconds for CR 2-2. The weighted average drop time for groups 1 through 7 was 1.293 seconds.

Based on the results shown in Table 2.3-1, the acceptance criteria were met by all of the control rods.

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[1] SP-102, Section 4, Rev. 17

2.4 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.

2.5 Pressurizer Effectiveness Test

No Pressurizer Effectiveness testing was done this startup since no modifications have been made to the pressurizer. Therefore, the results of the testing done at initial startup are still valid.

2.6 In-Service Loose Parts and Vibration Monitoring System Tests

The Loose Parts Monitoring Subsystem was calibrated during the course of the outage per Surveillance Procedure SP-152. The normal schedule of neutron noise readings, combined with routine core physics and Loose Parts Monitoring Systems, is sufficient to alert plant personnel of degrading core internals, if it occurs.

TABLE 2.0-1 SUMMARY OF PRECRITICAL TESTING RESULTS
Precritical Testing

Test (Reference)	Units	Results	Acceptance Criteria	Acceptability
1. Calibration and Neutron Response of Source Range Monitoring Tests (PT-100)	cps	$\geq$ 2 cps	Source range channels indicate $\geq$ 2 cps	OK
2a. Reactor Coolant Flow (SP-224)	lb_m/hr	153.765×10^6	Flow for four pumps must be greater than 139.7×10^6 lb_m/hr	OK
		0.661%	Loop flow within 2% of each other	OK
b. Reactor Coolant Flow Measurement Uncertainty	gpm	411,077	Max flow must be < 411,840	OK
		391,024	Min flow must be > 374,880	OK
3. Control Rod Drive Time Tests (SP-102)	seconds			
Least (CR 5-5, 7-6)		1.26	All rod drop times must be less than or equal to 1.66 seconds	OK
Greatest (CR 2-2)		1.33		OK
Group 1 Average		1.30	None	
Group 2 Average		1.31		
Group 3 Average		1.29		
Group 4 Average		1.31		
Group 5 Average		1.28		
Group 6 Average		1.29		
Group 7 Average		1.28		
Groups 1-7 Average		1.293	None	

TABLE 2.3-1 CONTROL ROD DRIVE DROP TIME TESTS RESULTS
Precritical Testing

Control Rod	Core Position	Drop Time (seconds)
CR 1-1	B-10	1.30
CR 1-2	F-14	1.32
CR 1-3	L-14	1.30
CR 1-4	P-10	1.29
CR 1-5	P-6	1.30
CR 1-6	L-2	1.30
CR 1-7	F-2	1.31
CR 1-8	B-6	1.28
CR 2-1	C-11	1.29
CR 2-2	E-13	1.33
CR 2-3	M-13	1.31
CR 2-4	O-11	1.31
CR 2-5	O-5	1.32
CR 2-6	M-3	1.32
CR 2-7	E-3	1.29
CR 2-8	C-5	1.32
CR 3-1	F-8	1.29
CR 3-2	G-9	1.29
CR 3-3	H-10	1.29
CR 3-4	K-9	1.29
CR 3-5	L-8	1.29
CR 3-6	K-7	1.31
CR 3-7	H-6	1.29
CR 3-8	G-7	1.30
CR 4-1	E-9	1.31
CR 4-2	G-11	1.29
CR 4-3	K-11	1.31
CR 4-4	M-9	1.32
CR 4-5	M-7	1.31
CR 4-6	K-5	1.32
CR 4-7	G-5	1.32
CR 4-8	E-7	1.31

TABLE 2.3-1 CONTROL ROD DRIVE DROP TIME TESTS RESULTS
Precritical Testing
(Continued)

Control Rod	Core Position	Drop Time (seconds)
CR 5-1	C-9	1.29
CR 5-2	E-11	1.28
CR 5-3	G-13	1.28
CR 5-4	K-13	1.30
CR 5-5	M-11	1.26
CR 5-6	O-9	1.28
CR 5-7	O-7	1.28
CR 5-8	M-5	1.29
CR 5-9	K-3	1.30
CR 5-10	G-3	1.28
CR 5-11	E-5	1.28
CR 5-12	C-7	1.28
CR 6-1	B-8	1.29
CR 6-2	F-10	1.29
CR 6-3	H-14	1.29
CR 6-4	L-10	1.30
CR 6-5	P-8	1.28
CR 6-6	L-6	1.29
CR 6-7	H-2	1.29
CR 6-8	F-6	1.29
CR 7-1	D-8	1.28
CR 7-2	D-12	1.28
CR 7-3	H-12	1.28
CR 7-4	N-12	1.28
CR 7-5	N-8	1.27
CR 7-6	N-4	1.26
CR 7-7	H-4	1.29
CR 7-8	D-4	1.29

Source: SP-102

TABLE 2.3-2 GROUP AVERAGE CONTROL ROD DROP TIME
Precritical Testing

Group	Number of Rods	Average Drop Time (Seconds)
1	8	1.30
2	8	1.31
3	8	1.29
4	8	1.31
5	12	1.28
6	8	1.29
7	8	1.28
1-7	60	Average = 1.293

3.0 ZERO POWER PHYSICS TESTING

The zero power physics tests (ZPPT) required by Section 13.4 Table 13-3 of the CR3 FSAR were performed in accordance with Performance Testing Procedure PT-110. These tests were done to verify nuclear design parameters used in the safety analysis, operational parameters, and limits set in the Technical Specifications.

Acceptance criteria deviation limits for the ZPPT are given in Table 3.0-1. A summary of measured and predicted values obtained for the ZPPT is given in Table 3.0-2.

3.1 Initial Criticality

3.1.1 Method

The reactor coolant conditions at criticality were 532°F, 2155±100 psig, and 1965 ppmB. In order to verify that the source range instrumentation was operating properly, a Chi-Square Test was performed which confirmed that a good fit to a Poisson distribution existed. The approach to criticality began by withdrawing control rod group 8 to 25% withdrawn and groups 5 and 6 to 100% withdrawn. Control rod groups 1 through 4 remained at 100% withdrawn. Rod group 7 was then withdrawn until criticality was achieved (40% withdrawn).

Throughout the approach to criticality, curves of inverse multiplication versus rod reactivity worth removed were maintained. The data was taken from two source range detectors by two people, independently. At the end of each control rod group withdrawal, the count rate was taken from each source range detector using the scaler-counters. The ratio of the initial average count rate to the average count rate at the end of each reactivity addition was plotted and the criticality point determined by extrapolation.

3.1.2 Results

Initial criticality for Cycle 7 was achieved on January 8, 1988 at 2042 hours.

3.2 Nuclear Instrumentation Overlap

The nuclear instrumentation (NI) detectors were used throughout testing to provide continuous reactor power information. The instrumentation consisted of eight measuring channels divided into three ranges: source (subcritical to startup), intermediate (startup to 150% full power), and power (0 to 125% full power). The location of these NIs is shown in Figures 3.2-1 and 3.2-2.

Technical Specifications state that:

> NI Source range must overlap the intermediate range by a factor of 10 or more.[1]

This means that before the source range count rate equals 10^5 cps, the intermediate range must be on scale. If the required one decade is not observed, the approach to the intermediate range cannot be continued until the situation has been corrected.

3.2.1 Method

For this test, reactor coolant conditions were 532°F and 2155 psig. To verify the 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 channels. This was then repeated for another decade.

3.2.2 Results

The results of the Nuclear Instrumentation overlap test are given in Table 3.2-1. This table shows that the average overlap between the source and intermediate range is 2.155 decades, which is above the minimum one decade specified as the acceptance criterion.

3.3 Sensible Heat Determination

By determining the intermediate range current level at which the production of sensible nuclear heat occurs, the upper zero power physics test current limit is established. Thus, by restricting reactor power operation to a level reduced by a conservatism factor of 3.3 below the sensible heat level, the effects of temperature feedback are eliminated in the measurement of physics parameters. The test for sensible heat was done according to Performance Testing Procedure PT-116.

3.3.1 Method

For this test, the intermediate range current level was increased in one-third decade increments until sensible heat was detected. The production of sensible heat in the core is indicated by an increase in the RCS T_{avg}, RCS loop T_{hot}, and makeup tank level.

3.3.2 Results

The point at which there was a definite heatup rate was 1.1×10^{-7} amps measured on channel NI-4. This was defined as the sensible heat point. From this, the upper current limit was established at 3.33×10^{-8} amps. This was found by dividing the sensible heat point by a 3.3 conservatism factor.

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[1] PT-110, Section 10.1, Rev. 12; Tech Specs Table 4.3-1, Note 5

3.4 Reactimeter Check

The Reactimeter is the reactivity computer manufactured by Babcock and Wilcox 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 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.

The Reactimeter check is subject to the following acceptance criterion:

> The reactivity values computed from measured doubling times must agree within ±5% of the reactivity values measured on the Reactimeter.[1]

3.4.1 Method

After steady state conditions with a constant neutron flux were established, approximately 25 pcm of negative reactivity was inserted into the core by inserting control rod group 7. Stop watches were used to measure the doubling time of the neutron flux and the inserted reactivity was determined from period-reactivity curves. The measurements were repeated for several values of reactivity ranging from -79.8 to +69.4 percent millirho.

3.4.2 Results

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 The reactivity calculated by the Reactimeter was within the acceptance criterion limit of ±5% of the reactivity determined from doubling times in each case.

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[1] PT-110, Section 10.2, Rev. 12

3.5 All Rods Out Critical Boron Test

This test was used to provide information relating to core excess reactivity by determining the amount of soluble boron that must be added to the coolant water to maintain a critical level with all control rods removed.

This test, performed in accordance with PT-111, is subject to the following acceptance criterion:

> The measured all rods out critical boron concentration... should be within ±50 ppmB of the value given in the Physics Test Manual.[1]

3.5.1 Method

The test measurements were made at a reactor coolant temperature of 532°F and a system pressure of 2155 psig. The portion of control rod group 7 which remained following deboration was withdrawn and the excess reactivity measured using the Reactimeter. From this value and the differential boron worth given in the Physics Test Manual, the all rods out critical boron concentration was determined.

3.5.2 Results

The excess reactivity was found to be 0.0225 %Δk/k. From this and the differential boron worth from the Physics Test Manual, it was determined that the all rods out critical boron concentration was 2033 ppmB. Since the predicted value is 2032 ppmB, the acceptance criterion was met.

3.6 Moderator and Temperature Coefficients Measurements

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 below.

$$\alpha_T = \alpha_M + \alpha_D$$

where: α_T = Temperature Coefficient of Reactivity
α_M = Moderator Coefficient of Reactivity
α_D = Doppler Coefficient of Reactivity

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[1]PT-111, Section 10.1, Rev. 7

For this test, performed in accordance with PT-114, the temperature and moderator coefficients at hot zero power were measured. Furthermore, an extrapolated hot full power moderator coefficient was calculated. Each of the above coefficients was determined for an all-rods-out (ARO) control rod configuration, with rod bank 8 remaining at 25% withdrawn. Previously, these calculations were repeated with regulating rod banks 5-7 inserted. This all-rods-in (ARI) calculation was eliminated because:[1]

1. There is good agreement between predicted and measured coefficients.
2. There is a very good correlation between the direction of predicted versus measured deviation for the ARO and ARI coefficients for a given reload cycle.

The acceptance criterion for the hot zero power temperature coefficients is:

> The calculated hot zero power temperature coefficient...shall be within $\pm 0.4 \times 10^{-4}$ $\Delta k/k/°F$ of the value given in the Physics Test Manual.[2]

The acceptance criteria for the hot zero power moderator coefficients of reactivity are:

> The hot zero power moderator coefficient shall be greater than -3.0×10^{-4} $\Delta k/k/°F$ but less than 0.9×10^{-4} $\Delta k/k/°F$.[3]

For the extrapolated moderator coefficient, the criterion is as follows:

> The hot full power moderator coefficient shall be less than zero.[4]

3.6.1 Method

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. The reactor was maintained at a critical level between 5×10^{-10} amps and the upper zero power physics test current limit as defined by the sensible heat determination experiment. Equilibrium boron concentration was established in the reactor coolant system, makeup tank, and

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[1] "Reduced Physics Testing, Task Summary Report" document by Babcock and Wilcox, B&W document number 86-1164722-00, dated March 1987 for the B&W Owners' Group Performance Committee.
[2] PT-114, Section 10.4, Rev. 11
[3] PT-114, Sections 10.1, 10.3, Rev. 11
[4] PT-114, Section 10.2, Rev. 11

pressurizer to eliminate reactivity effects from 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-pen recorder.

Once steady state conditions were established, a positive heatup rate was maintained by adjusting the turbine header pressure set point. As the reactivity changed, the controlling rod group was moved as necessary to produce an adequate intermediate range signal. After the core average temperature increased by about 5°F, coolant temperature and reactivity were stabilized. This process was then reversed, and core average temperature was decreased by about 10°F. Finally, after stabilizing coolant temperature and reactivity, the core average temperature was returned to its original value. The measurement of the temperature coefficient from the data obtained was performed by dividing the change in reactivity by the corresponding changes in core temperature for a specific time period.

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. To obtain the moderator coefficient from the measured temperature coefficient, a Doppler correction of -0.151×10^{-4} $\Delta k/k/°F$ must be subtracted.

The extrapolated hot full power moderator coefficient is calculated based on the hot zero power moderator coefficient. Added to this are both a calculated control rod effect and a boron change effect, which include the Doppler and xenon effects.

3.6.2 Results

The results of the hot zero power temperature and moderator coefficients measurements are summarized in Table 3.6-1 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 criterion of $\pm 0.40 \times 10^{-4}$ $\Delta k/k/°F$ of the predicted value. In addition, calculation of the moderator coefficient indicates that it is well within the requirements of the Technical Specification.

The extrapolated hot full power moderator coefficient, as given in Table 3.6-2, is -1.41×10^{-5} $\Delta k/k/°F$ for all rods out. From this, it is seen that the acceptance criterion of a hot full power moderator coefficient of less than zero is met.

3.7 Control Rod Group Worths

The layout of the core showing the location of the control rod groups and the location of the 52 incore detector strings is given in Figure 3.7-1. 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	8	Safety
4	8	Safety
5	12	Power Doppler
6	8	Power Doppler
7	8	Transient
8	8	Axial Power Shaping
	Total = 68	

The control rod worth tests were run in accordance with Performance Testing Procedure PT-112 to provide information about the reactivity worths of banks 5, 6, and 7. The results are also used to verify rod worth curves used during plant operation. The tests are subject to the following acceptance criteria:

> The predicted reactivity worth of rod groups, 5, 6, and 7 in the Physics Test Manual shall each be within ±15% of the measured value.[1] Additionally, the predicted total reactivity worth of the sum of rod groups 5, 6, and 7 in the Physics Test Manual shall be within ±10% of the measured total.[2]

3.7.1 Method

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 continuous calculation of reactivity was made by the Reactimeter. The reactivity in percent millirho (PCM) from the Reactimeter and the rod position were recorded on a strip chart and magnetic tape.

3.7.2 Results

The results of both the predicted rod group worths and the measured group worths are tabulated in Table 3.0-2 for a reactor coolant system temperature of 532°F and pressure of 2155 psig. Comparison between measured and predicted rod worths shows groups 5-7 were well within the specified 15% deviation, and the sum of groups 5-7 was within 3.029% percent of the predicted value. Therefore, all acceptance criteria were met.

Integral measured rod worth curves for control rod groups 5, 6, and 7 at 532°F and 2155 psig are plotted in Figures 3.7-2 through 3.7-4.

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[1] PT-112, Section 10.1, Rev. 7
[2] PT-112, Section 10.2, Rev. 7

3.8 Differential Boron Worth Determination

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 the life of the cycle.

The differential boron worth was measured in accordance with Performance Testing Procedure PT-112. From PT-112, the following acceptance criterion must be met:

> The predicted differential boron worth in the Physics Test Manual shall be within ±15% of the measured value.[1]

3.8.1 Method

The test measurements of the boron differential worth was completed at reactor coolant conditions of 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 2028 to 1635 ppmB.

3.8.2 Results

The measured differential boron worth was calculated as -7.8117×10^{-3} $\Delta k/k$/ppmB, as compared to the predicted value of -7.60×10^{-3} $\Delta k/k$/ppmB. The deviation is therefore -2.71%, which is well within the acceptance criteria of ±15%. The results of the differential soluble poison worth measurements are tabulated in Table 3.8-1.

3.9 Ejected Control Rod Worth Measurement

In previous cycles, this test was performed to verify the safety analysis calculations relating to the assumed accidental ejection of the most reactive control rod. From the existing B&W data base, there is good agreement between the measured and predicted ejected rod worths. It has been determined[2] that the HZP ejected rod worth (ERW) test can be eliminated if the predicted value of maximum HZP ejected rod worth is less than 0.8 %$\Delta k/k$. Since the predicted ejected rod worth from the Physics Test Manual is 0.434 %$\Delta k/k$, the ejected control rod worth test has been eliminated from zero power physics testing for Cycle 7.

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[1] PT-112, Section 10.3, Rev. 7
[2] "Reduced Physics Testing, Task Summary Report" document by Babcock and Wilcox, document number 86-1164722-00, dated March 1987 for the B&W Owners' Group Performance Committee.

3.10 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.

3.11 Effluent and Effluent Monitoring

No Effluent or Effluent Monitoring testing was performed as these systems have been performing normally since initial startup. Therefore, no further testing was required.

3.12 Chemical and Radiochemical Tests

Chemical and Radiochemical testing was not performed during this start-up. 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

Zero Power Physics Testing

Core Physics Parameters	Allowable Deviation Between Measured & Predicted Values
o All Rods Out Boron Concentration	± 50 ppmB
o Temperature Coefficient of Reactivity	$\pm 0.4 \times 10^{-4}$ $\Delta k/k/°F$
o Control Rod Worths	
Individual Group Worths (Groups 5, 6, & 7)	± 15%
Total Group Worth (Groups 5-7)	± 10%
o Differential Boron Worth	± 15%

Table 3.0-2 SUMMARY OF ZERO POWER PHYSICS TESTING RESULTS
Zero Power Physics Testing

Physics Parameter (Reference)	Units	Measured	Predicted	Acceptance Criteria	Comparison
1. NI Overlap (PT-110)	decades	2.155		Overlap must be greater than 1.0 decade	OK (2.155)
2. Sensible Heat (PT-116)	Amps				
NI-3		1.1×10^{-7}		None	
NI-4		1.1×10^{-7}			
3. All Rods Out Critical Boron (PT-111)	ppmB	2033	2032	Measured value must be within 50 ppmB of predicted value	OK (1)
4. Temperature Coefficient of Reactivity (PT-114)	$\Delta k/k/°F$				
ARO, 2030 ppmB		$+0.225\times10^{-4}$	$+0.286\times10^{-4}$	Measured value must be within 0.4×10^{-4} $\Delta k/k/°F$ of the predicted value	OK
5. Moderator Coefficient of Reactivity (PT-114)	$\Delta k/k/°F$				
ARO, 2030 ppmB		$+0.376\times10^{-4}$	$+0.453\times10^{-4}$	Maximum positive moderator coefficient must be greater than -3×10^{-4} $\Delta k/k/°F$ and less than 0.9×10^{-4} $\Delta k/k/°F$.	OK OK

TABLE 3.0-2 SUMMARY OF ZERO POWER PHYSICS TESTING RESULTS
Zero Power Physics Testing (Continued)

Physics Parameter (Reference)	Units	Measured	Predicted	Acceptance Criteria	Comparison
6. Extrapolated Moderator Coefficient of Reactivity (PT-114)	$\Delta k/k/^{o}F$				
ARO		-1.41×10^{-5}		Extrapolated hot full power moderator coefficient must be less than 0.0	OK
7. Control Rod Group Worth (PT-112)	% $\Delta k/k$				
Group 7		-0.8375	-0.881	Percent deviation of group worth must be less than 15%	OK (5.13)
Group 6		-0.8320	-0.927		OK (11.42)
Group 5		-1.4005	-1.355		OK (3.25)
Total		-3.071	-3.163	Percent deviation of total worth must be less than 10%	OK (3.029)
8. Differential Boron Worth (PT-112)	% $\Delta k/k$/ppmB				
1831.5 ppmB		-7.812×10^{-3}	-7.60×10^{-3}	Percent deviation must be less than 15%	OK (-2.71)

Percent Deviation is calculated as follows:

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

TABLE 3.2-1 NUCLEAR INSTRUMENTATION OVERLAP TEST RESULTS
Zero Power Physics Testing

Case Number	Source Range (SR) Indication		Intermediate Range (IR) Indication		Average SR Indication (CPS)	Average IR Indication (Amps)	Overlap* (Decades)
	NI-1 (CPS)	NI-2 (CPS)	NI-3 (Amps)	NI-4 (Amps)			
1	6×10^{4}	6×10^{4}	7×10^{-11}	7×10^{-11}	6×10^{4}	7×10^{-11}	2.067
2	7×10^{5}	7×10^{5}	7×10^{-10}	7×10^{-10}	7×10^{5}	7×10^{-10}	2.000
3	2×10^{5}	2×10^{5}	4×10^{-10}	6×10^{-10}	2×10^{5}	5×10^{-10}	2.398

Average Overlap = 2.155

*Overlap between the Source (SR) and Intermediate Range (IR) average indications is obtained using the following equation:

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

Source: PT-110, Enclosure 4, Attachment 1, Console Values

TABLE 3.4-1 REACTIMETER AND DOUBLING TIME REACTIVITY COMPARISON

Zero Power Physics Testing

Case Number	Doubling Time (DT) (Sec)	Average Reactivity from DTs (PCM)	Reactimeter Reactivity (PCM)	Absolute Error * (%)
1	223.3	-26.8	-26	3.1
2	202.0	+23.5	+24	2.1
3	118.7	-58.2	-56	3.9
4	97.7	+43.9	+44	0.2
5	95.7	-79.8	-76	5.0
6	54.7	+69.4	+69	0.6

* Absolute error (E%) between the Doubling Time reactivity (ρ_{DT}) and Reactimeter reactivity (ρ_R) is given by the following equation:

$$E\ (\%) = 100^* \left| (\rho_{DT} - \rho_R) / (\rho_R) \right|$$

Source: PT-110, Enclosure 5, Attachment 1

TABLE 3.6-1 COMPARISON OF MEASURED AND PREDICTED REACTIVITY COEFFICIENTS

Zero Power Physics Testing

Rod Configuration	Control Rod Group (Group Position, % Withdrawn)								Average Temp (°F)	RCS Boron Concentration (ppm)	Reactivity Coefficients ($\Delta k/k/°F$)			
	1	2	3	4	5	6	7	8			Temperature Coefficient		Moderator Coefficient	
											Measured	Predicted	Measured	Predicted
ARO	100	100	100	100	100	100	90	25	532	2030	$+0.225\times10^{-4}$	$+0.286\times10^{-4}$	$+0.376\times10^{-4}$	$+0.453\times10^{-4}$

... = All Rods Out

... .1-114, Enclosure 1

TABLE 3.6-2 EXTRAPOLATED HOT FULL POWER MODERATOR COEFFICIENT OF REACTIVITY

Zero Power Physics Testing

Rod Configuration	Predicted Hot Full Power Moderator Coefficient of Reactivity ($\Delta k/k/^{\circ}F$)
All Rods Out	-1.41×10^{-5}

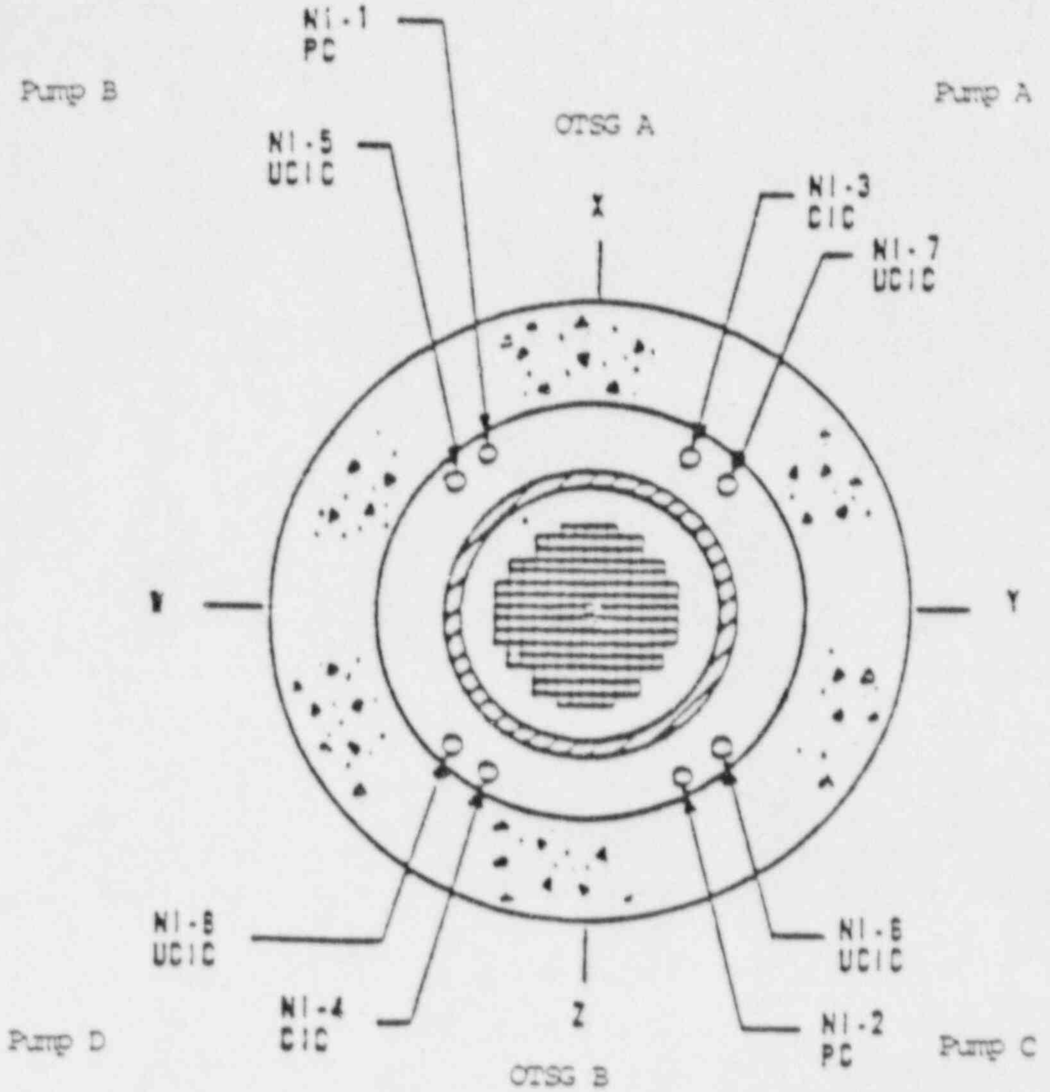
Source: PT-114, Enclosure 2

TABLE 3.8-1 DIFFERENTIAL BORON REACTIVITY WORTH TEST RESULTS

Zero Power Physics Testing

Control Rod Group (Group Position, % Withdrawn								Measured Boron Conc. (ppm)	Average Boron Conc. (ppm)	Delta Boron Conc. (ppm)	Boron Worth (%Δk/k)	Differential Boron Worth (%Δk/k/ppmB)	
1	2	3	4	5	6	7	8					Measured	Predicted
100	100	100	100	100	100	89	25.0	2028					
100	100	100	100	0	0	0	25.0	1635	1831.5	-393	-3.070	-7.8117×10^{-3}	-7.60×10^{-3}

Source: PT-112, Enclosures 1 and 4



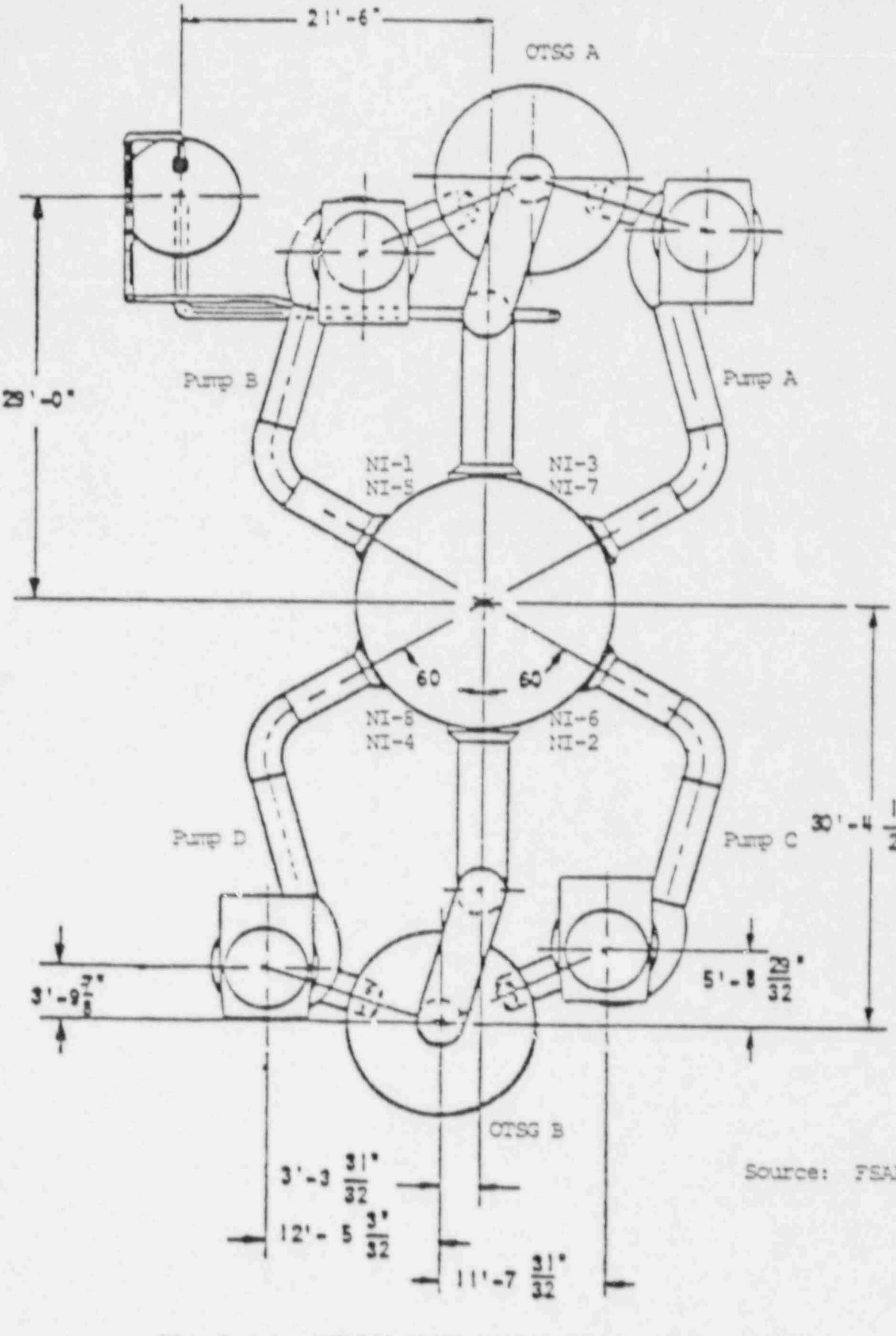
PC - PROPORTIONAL COUNTER - SOURCE RANGE 1 DETECTOR

CIC - COMPENSATED ION CHAMBER - INTERMEDIATE RANGE DETECTOR

UCIC - UNCOMPENSATED ION CHAMBER - POWER RANGE DETECTOR

Source: FSAR, Fig. 7-18

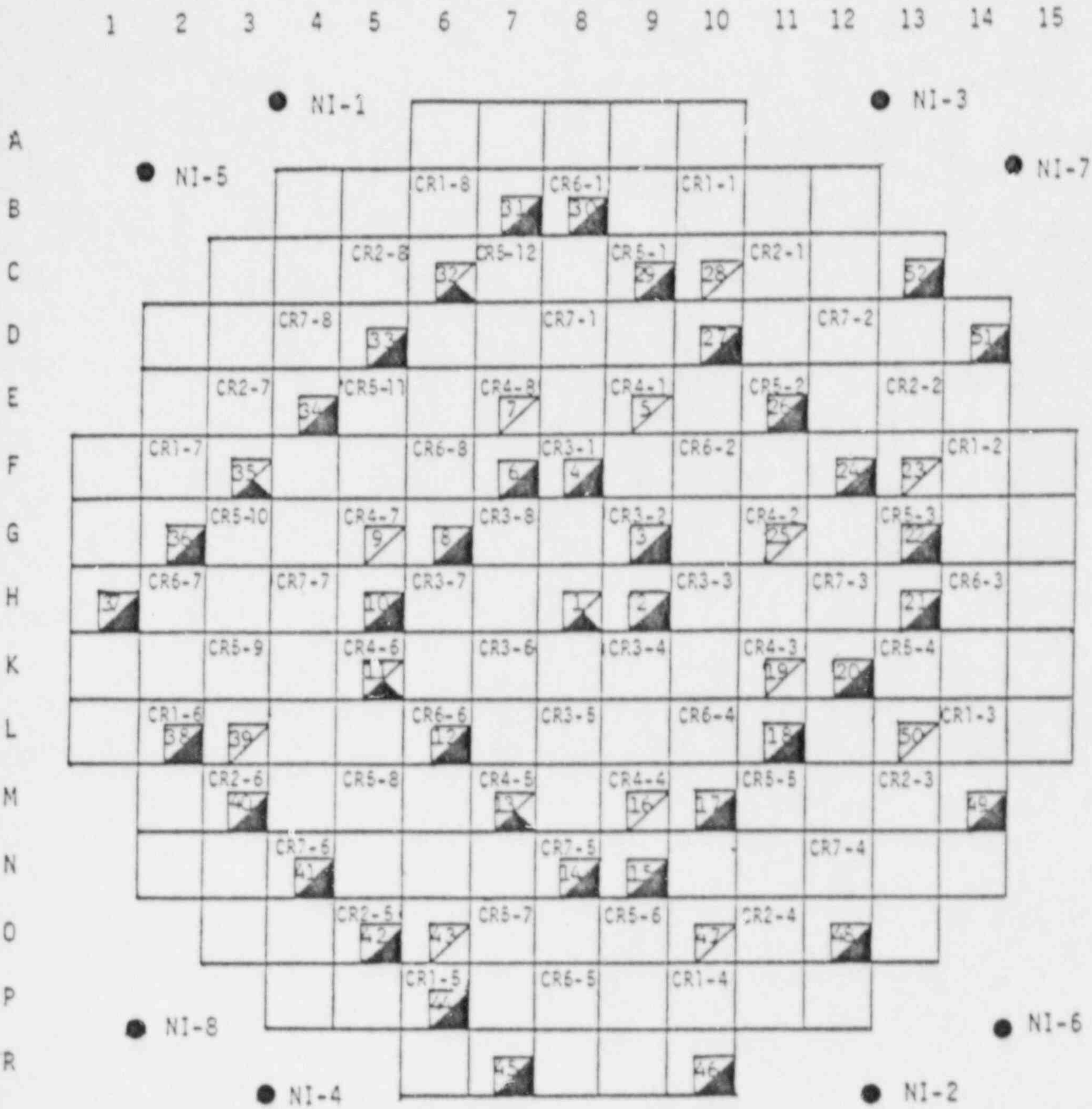
Fig. 3.2-1 NUCLEAR INSTRUMENTATION DETECTOR LOCATIONS



Source: FSAR, Fig. 4-3

Fig. 3.2-2 NUCLEAR INSTRUMENTATION DETECTOR LOCATIONS

INCORE MONITOR AND CONTROL ROD MAP



Total Core Monitor

Total Core & Symmetry Monitor

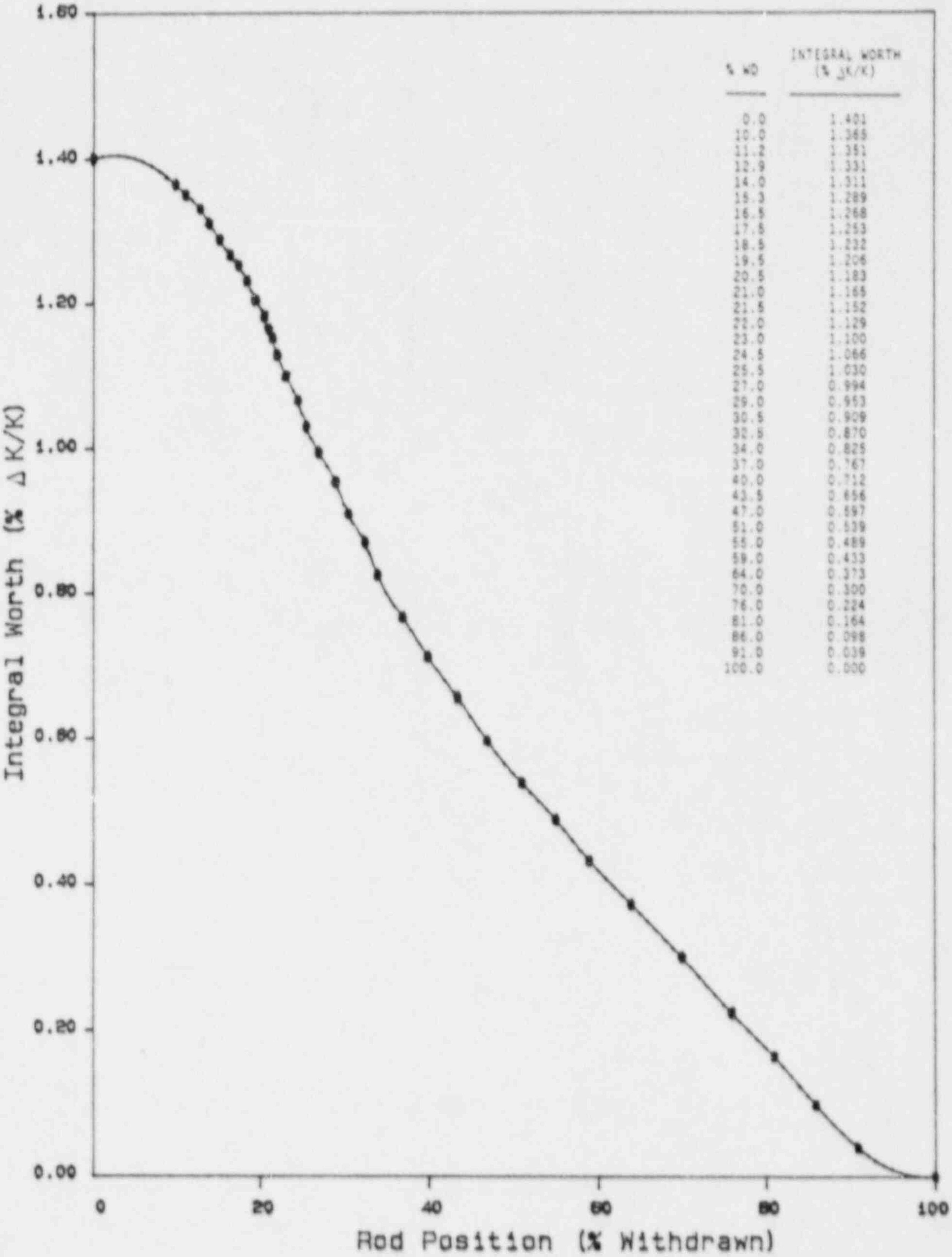
Symmetry Monitor

z - Incore Monitor Number

CRX-Y - Control Rod Y of Control Rod Group X

Figure 3.7-1

Control Rod Group 5
Integral Worth
BOC 7 at HZP

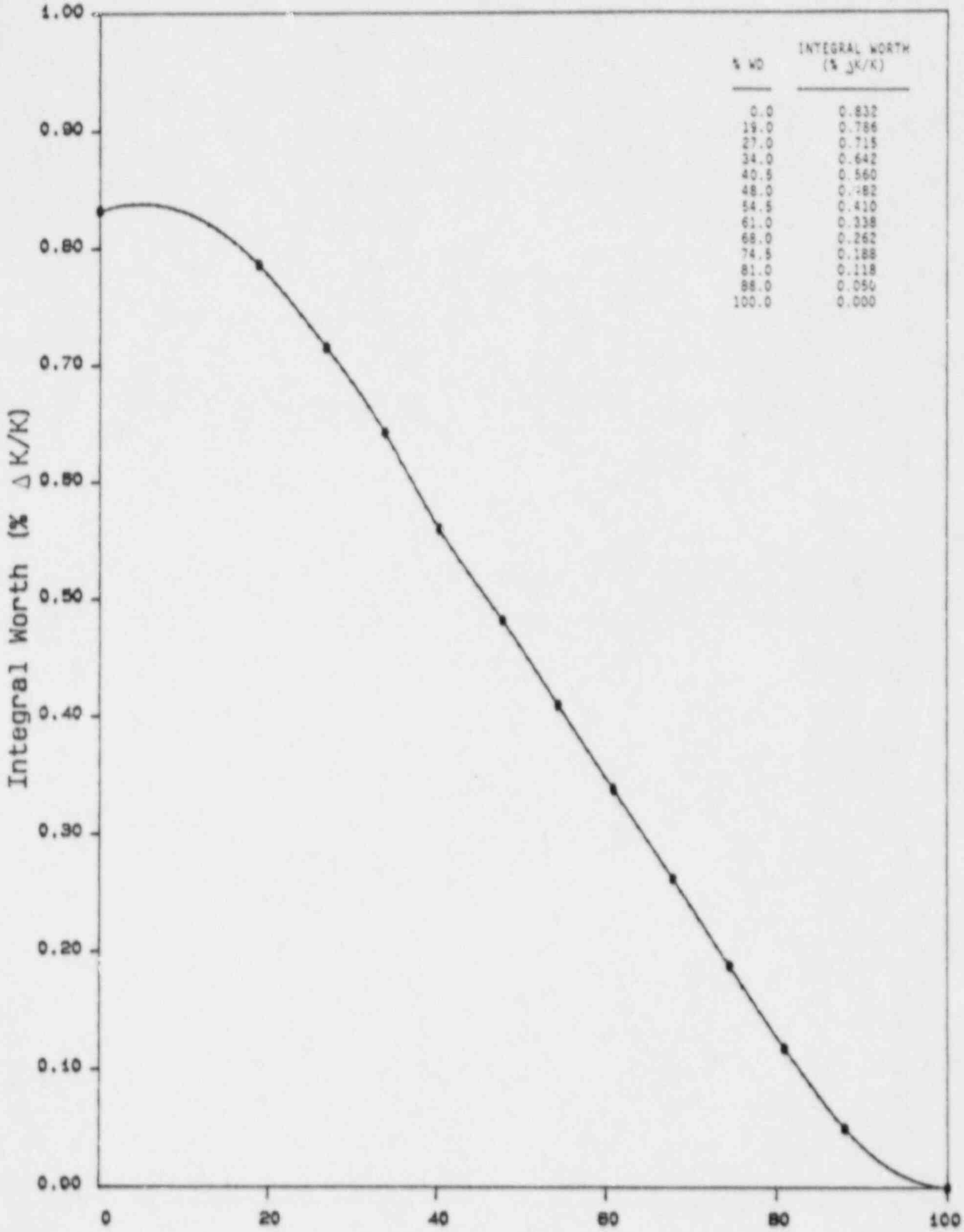


% WD	INTEGRAL WORTH (% ΔK/K)
0.0	1.401
10.0	1.365
11.2	1.351
12.9	1.331
14.0	1.311
15.3	1.289
16.5	1.268
17.5	1.253
18.5	1.232
19.5	1.206
20.5	1.183
21.0	1.165
21.5	1.152
22.0	1.129
23.0	1.100
24.5	1.066
25.5	1.030
27.0	0.994
29.0	0.953
30.5	0.909
32.5	0.870
34.0	0.825
37.0	0.767
40.0	0.712
43.5	0.656
47.0	0.597
51.0	0.539
55.0	0.489
59.0	0.433
64.0	0.373
70.0	0.300
76.0	0.224
81.0	0.164
86.0	0.098
91.0	0.039
100.0	0.000

Source: PT-112, Encl. 10

Figure 3.7-2

Control Rod Group 6
Integral Worth
BOC 7 at HZP

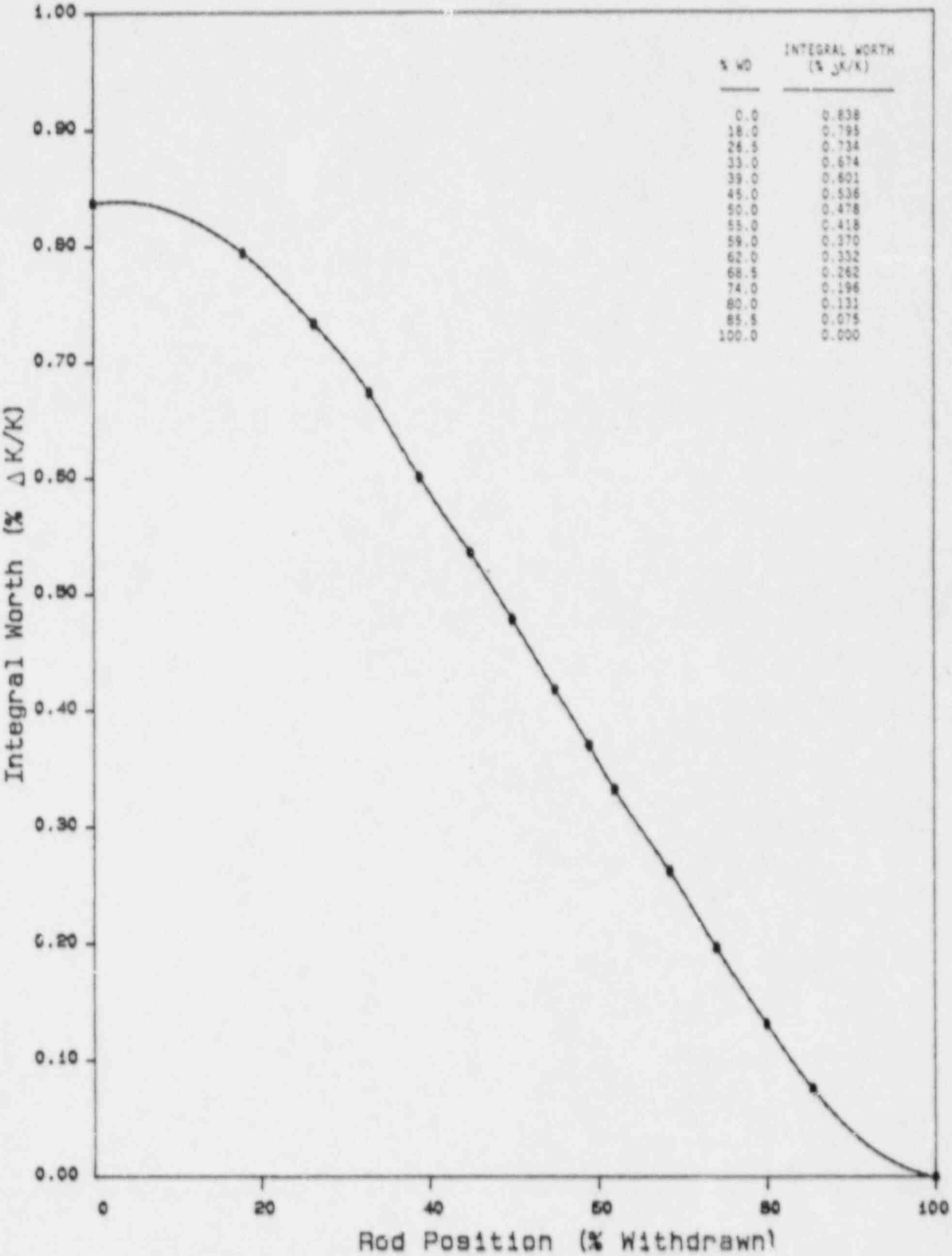


% WD	INTEGRAL WORTH (% ΔK/K)
0.0	0.832
19.0	0.786
27.0	0.715
34.0	0.642
40.5	0.560
48.0	0.482
54.5	0.410
61.0	0.338
68.0	0.262
74.5	0.188
81.0	0.118
88.0	0.050
100.0	0.000

Source: PT-112, Encl. 10

Figure 3.7-3

Control Rod Group 7
Integral Worth
BOC 7 at HZP



Source: PT-112, Encl. 10

Figure 3.7-4

4.0 FULL POWER ESCALATION TESTING

Full power escalation tests were performed to verify the validity of the safety analysis assumptions and, therefore, the acceptability of the core design. Testing was performed at two major power plateaus; 75% and 95-100% full power (FP). In previous cycles testing was performed at three major power plateaus, but one intermediate power distribution test plateau was eliminated as explained in the Introduction (Section 1.0)

4.1 Turbine/Reactor Trip Test

No Turbine/Reactor Trip testing was performed during this startup since no modifications were made which would invalidate the original testing results.

4.2 Integral Control System Test

Since no modifications were made to the Integrated Control System (ICS) during this outage, no specific ICS testing was done during this start-up. Minor adjustments were made to the ICS under normal maintenance and calibration procedures.

4.3 Unit Loss of Electrical Load

No Loss of Electrical Load testing was performed since no modifications were made which would invalidate the original testing results.

4.4 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 throughout the testing program.

4.5 Reactivity Coefficients At Power Test

The purpose of this test was to determine reactivity coefficients at 100% 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.

d. Fuel temperature Doppler coefficient of reactivity, defined as the fractional change in the reactivity of the core per unit change in fuel temperature.

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

1. The moderator coefficient of reactivity shall be less positive than 0.9×10^{-4} $\Delta k/k/°F$ at power levels below 95% full power, less positive than 0.0×10^{-4} $\Delta k/k/°F$ at power levels at or above 95% full power, and less negative than -3.0×10^{-4} $\Delta k/k/°F$ at rated thermal power.[1]

2. The fuel temperature Doppler coefficient of reactivity shall be more negative than -0.90×10^{-5} $\Delta k/k/°F$.[2]

4.5.1 Method

Reactivity coefficient measurements were made during the power escalation test program at 100.0% 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% full power and data recorded. From the measured temperature and power Doppler coefficients, the moderator and fuel temperature Doppler coefficients were calculated.

4.5.1.1 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 six seconds, followed immediately by a withdrawal for 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.

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[1] PT-120, Section 10.2.7, Rev. 11
[2] PT-120, Section 10.2.8, Rev. 11

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 corection factor accounts for the time delay involved in fuel temperature change during the measurement.

4.5.1.2 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 4.5-1.

$$\alpha_T = \alpha_M + \alpha_D \qquad \text{EQ. (4.5-1)}$$

Where: α_T = Temperature Coefficient of Reactivity ($\Delta k/k/°F$)
α_M = Moderator Coefficient of Reactivity ($\Delta k/k/°F$)
α_D = Doppler Coefficient of Reactivity ($\Delta k/k/°F$)

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 4.5-1 after the temperature and Doppler coefficients have been determined.

Temperature, moderator, and Doppler coefficients were theoretically predicted as shown in Table 4.5-1 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 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 fuel temperature Doppler coefficient relates the change in core reactivity to a corresponding change in fuel temperature. A theoretical prediction of the fuel temperature Doppler coefficient was made using the PDQ code with thermal feedback, and is presented in Table 4.5-1.

The measurement method used was to change the reactor power level 5% 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. This is then converted to a fuel temperature Doppler coefficient by multiplying by a theoretically derived factor, Δ%FP/Δ°F.

4.5.2 Results

The results of the measured temperature and Doppler coefficients, and calculated moderator coefficient at power are shown in Table 4.5-1 which also shows the predicted temperature, moderator, and Doppler coefficient results.

The calculated moderator coefficient is negative and therefore meets the acceptance criteria of being non-positive.

The results of the measured and predicted Doppler coefficient of reactivity are shown in Table 4.5-1. The acceptance criterion for the measured Doppler coefficient is that the coefficient must be more negative than -0.90×10^{-5} $\Delta k/k/°F$. Table 4.5-1 shows that the measured coefficient is below this value and that the acceptance criterion is adequately met.

4.6 Unit Heat Balance

Heat balance calculations were performed using the ModComp computer. The data was taken in accordance with Surveillance Procedure SP-312. No modifications were made during this shutdown which would require reverification of the heat balance calculation.

4.7 Core Power Distribution Test

The Core Power Distribution test was performed at each power plateau (75% FP and 95-100% FP) to measure the core flux and power distributions. These measured powers were then compared to the predicted values. The test data was also used to evaluate core performance and the departure from nucleate boiling ratio (DNBR). Additionally, test results were used in the Power Imbalance Detector Correlation test and the Incore Detector test.

The test, performed in accordance with PT-120, is subject to the following acceptance criteria:

> The core power distribution and thermal-hydraulic parameters must be measured, evaluated, and deemed reasonable.[1]

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[1]PT-120, Section 10.2.6, Rev. 11

The highest measured radial and total peaking factors shall not be greater than 5% and 7.5%, respectively, of predicted values at the 40-75% FP and 95-100% FP plateaus.[1]

The minimum DNBR is greater than 1.30 and the measured worst case maximum linear heat rate (LHR) is less than 19.2 kW/ft[2] when extrapolated to the next power plateau.

The extrapolated worst case minimum DNBR is greater than 1.30 and the extrapolated worst case maximum LHR is less than 19.2 kW/ft, or if the MLHR is greater than 19.2 kW/ft, the extrapolated imbalance falls outside the power imbalance trip envelope as shown in Figure 4.7-1.[3]

Continuous monitoring of the core power density at 364 core locations was accomplished using the incore monitoring system. This system is comprised of 52 detector strings each having 7 individual neutron detectors. These detectors are 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. The output of the incore detectors was connected to the unit computer and corrected for background, fuel depletion and the as-built dimensions to provide accurate outputs of relative neutron flux. The computer output of the corrected signals was used to develop core power distributions which provide power peaking information necessary to determine 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 4.7-1.

The out-of-core nuclear instrumentation provides the core power and core power imbalance signals to the reactor protection system, since the incore monitoring system does not immediately respond to prompt changes in the core conditions. The out-of-core nuclear instrumentation (NI) is shown in Figure 3.7-1 as NI 1 through 8.

4.7.1 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. For this test, the APSRs were maintained at a constant position and the axial incore imbalance was maintained within 2% FP of the imbalance identified in the Physics Test Manual.

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[1] PT-120, Section 10.2.5, Rev. 11
[2] PT-120, Sections 10.2.3 and 10.2.2, Rev. 11
[3] PT-120, Enclosure 2, Parts 5.0 and 6.2, Rev. 11

4.7.1.1 Normal Operating Core Power Distributions. Normal operating equilibrium xenon core power distributions were measured in this test. Data was taken at each power plateau; 72.9% FP, and 100% FP. Values were obtained over an eighth core for radial and maximum total power peaking factors. These results were then compared with the values predicted in the Physics Test Manual. Based on the maximum calculated and maximum measured peaks, percent deviations were determined. The percent deviation was calculated using the following equation:

$$\% \text{ Deviation} = \left[\frac{\text{Calculated - Measured}}{\text{Measured}} \right] \times 100 \quad \text{EQ.(4.3-1)}$$

The deviation was calculated for the radial and for the maximum total power peaking factors at each power plateau. These distributions are shown in Figures 4.7-2 through 4.7-5.

4.7.1.2 Worst Case Minimum DNBR. To maintain the integrity of the fuel cladding which prevents 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 19.2 kW/ft.

The upper boundary of the nucleate boiling region is called "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures 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.

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 empirically developed by Babcock & Wilcox 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 99% confidence level that DNB will not occur. This is considered a conservative margin to DNB for all operational conditions.

The worst case minimum DNBR values were calculated by the unit computer for each core power distribution that was taken during the power escalation test program.

Two normal operating equilibrium xenon core power distributions, as required by the Core Power Distribution Test, were obtained during the power escalation sequence. Each power distribution was subjected to the following analysis. From each core power distribution, the worst case measured minimum DNBR was selected. These DNBRs were then extrapolated to the overpower trip setpoint and corrected for axial peak location and magnitude. Verification of acceptable core conditions at the present and next power level of escalation was then performed based on the acceptance criterion that the extrapolated DNBR must be greater than 1.30. The extrapolated DNBR values were supplied by the plant computer's PDOs based on actual conditions at the plateau. Next, the measured worst case minimum DNBRs were extrapolated to the Loss of Coolant Accident and design overpower power level and corrected for axial peak location and magnitude.

4.7.1.3 Worst Case Maximum Linear Heat Rate Determination. Worst case maximum linear heat rate (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.

4.7.1.4 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 tilts are subject to the following acceptance criteria:

> The quadrant power tilts shall not exceed the limits specified in Technical Specifications Table 3.2-2.[1] Furthermore, the calculated incore power tilts must be less than the error adjusted tilt limit.[2]

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

$$\text{Quadrant Power} = \left[\frac{\text{Power in Any Core Quadrant}}{\text{Average Power of All Quadrants}} - 1 \right] \times 100 \qquad \text{EQ.(4.3-2)}$$

During the startup testing program, maximum quadrant power tilt was determined using the corrected signals from the 16 symmetric incore monitoring assemblies. Figures 4.7-2 through 4.7-5 include the maximum quadrant power tilt for each standard core power distribution taken as required by the Core Power Distribution Test.

4.7.2 <u>Results</u>

4.7.2.1 Normal Operating Core Power Distributions. The results of the core power distributions are shown in Figures 4.7-2 through 4.7-5. The percent deviation for each symmetric length core assembly is also given. The acceptance criteria for this test were met.

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[1] PT-120, Section 10.2.4, Rev. 11
[2] PT-120, Section 10.2.11, Rev. 11

4.7.2.2 Worst Case Minimum DNBR. The results of various worst case minimum DNBR values calculated at each test plateau under normal rod configurations are plotted in Figure 4.7-6. 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.

4.7.2.3 Worst Case Maximum Linear Heat Rate. In all cases, the acceptance criterion limit of 19.2 kW/ft was met during the power escalation test program. These results are summarized in Table 4.7-1.

4.7.2.4 Quadrant Power Tilt. Technical Specification tilt limits were not exceeded at any of the power plateaus.

4.7.2.5 Axial Power Imbalance. Results from standard core power distributions taken at 72.9% FP during the performance of the Power Imbalance Detector Correlation Test show that the imbalance trip envelope (Figure 4.7-1) of the reactor protective system is sufficient to protect the unit from exceeding the DNBR and the LHR limits under all core imbalance conditions. In addition, analyses indicate that the largest thermal margins (measured by DNBR and LHR) exist when a negative 5.0% to positive 10.0% incore axial offset is present as shown in Figure 4.12-3.

The core imbalances measured in conjunction with the core power distribution of this section are included in Figures 4.7-2 through 4.7-5.

4.8 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.

4.9 Pseudo Rod Ejection Test

In previous cycles, the Pseudo Rod Ejection test was not performed during the power escalation test program since this test had been performed during zero power physics testing. Since the maximum HZP ejected rod worth from the Physics Test Manual is less than 0.8 %Δk/k, it was unnecessary to perform these tests.[1]

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[1] "Reduced Physics Testing, Task Summary Report" document by Babcock and Wilcox, B&W document number 86-1164722-00, dated March 1987 for B&W Owners' Group Performance Committee.

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 of testing during the initial start up.

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

Power imbalance is defined as;

$$\text{Imbalance} = \%\text{ power in top of core} - \%\text{ power in bottom of core} \qquad \text{EQ. (4.5-1)}$$

The 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 allowed in the core so that the DNBR or LHR limits are not exceeded is set in the reactor protection system. The imbalance 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.

The Power Imbalance Detector Correlation (PIDC) test was performed in accordance with Performance Testing Procedure PT-120. The PIDC was done to determine the relationship between core offset as indicated by the out-of-core power range NI detectors and core offset as indicated by the full incore monitoring system. Offset is the imbalance divided by the total core power. The PIDC also verified that an acceptable relationship between the backup and full incore monitoring systems offset was observed. Finally, the PIDC verified that the measured worst case minimum DNBR and maximum LHR were acceptable. These values were verified to assure that the plant operates within the assumptions made in the safety analysis calculations.

Three acceptance criteria are specified for the Power Imbalance Detector Correlation Test:

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 between 0.96 to 1.10.[1]

The measured relationship between the full incore monitoring system offset and the backup incore recorder offset lies within the acceptable region shown in Figure 4.12-2.[2]

The measured worst case minimum DNBR is greater than 1.30 and the measured worst case maximum LHR is less than 19.2 kW/ft.[3]

4.12.1 Method

During the power escalation sequence at Crystal River 3, imbalance measurements were made to determine the acceptability of the out-of-core detectors' ability to measure 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 72.9% FP.

In performing the test, the APSRs were positioned to obtain the desired full incore imbalance with reactivity compensations made by control rod groups 6 and/or 7. At 72.9% FP, the offset indicated by the full incore system, out-of-core system, and backup recorder system was recorded, and is shown in Table 4.12-1.

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:

$$OCO = (M \times ICO) + B \qquad \text{EQ. (4.5-2)}$$

Where: OCO = Out-of-Core Offset (Percent)
ICO = Incore Offset (Percent)
M = Slope of Relationship
B = Intercept, when Incore Offset = 0

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 is outside the range of 0.96 to 1.10, then the gain of the out-of-core power range detectors would be adjusted. The relationship of measured slope to gain factor is as follows:

$$GF = (M2/M1) \times GF_o \qquad \text{EQ. (4.5.3)}$$

Where: GF = Desired Gain Factor
M2 = Desired Slope
M1 = Measured Slope
GF_o = Present Gain Factor

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[1] PT-120, Section 10.2.9, Rev. 11
[2] PT-120, Section 10.2.10, Rev. 11
[3] PT-120, Sections 10.2.3 and 10.2.2

Verification of the adequacy of the power imbalance system trip setpoint was performed in conjunction with the worst case analysis on each minimum DNBR and maximum LHR measured. Each measured point was extrapolated to the power/imbalance/flow envelope boundary limits given in Figure 4.7-1. In this way, the adequacy of the imbalance system trip setpoints to protect the unit from exceeding thermal-hydraulic limits could be verified.

4.12.2 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 APSRs and control rod group 6 and/or 7 at 72.9% 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 in the range of 0.96 to 1.10, which verified the utilization of the gain factors used 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, and show that the acceptance criterion was met.

During all phases of testing, worst case minimum DNBR and maximum LHR were recorded against incore offset. These results are given in Table 4.12-3 and Figure 4.12-3. The most limiting value observed for the maximum LHR and the worst case minimum DNBR was 10.63 kW/ft and 3.684, respectively, which is well within the procedural acceptance criteria.

4.13 Nuclear Instrumentation Calibration At Power Test

The Nuclear Instrumentation Calibration test was performed "to verify the ability to calibrate power range nuclear instrumentation to measured core conditions".[1] This test was performed in accordance with procedures PT-120 and SP-113, and in conjunction with Surveillance Procedure SP-312.

The acceptance criteria for this test are:

The power range nuclear instrumentation (NI) is calibrated to within ±2% of the power calculated in the heat balance.[2]

The high level bistable trip is set to trip within the specified limits.[3]

The absolute difference between the out-of-core detector axial power imbalance and the in-core detector axial power imbalance is less than 2.5%.[4]

Furthermore, the Technical Specifications require that the overlap between the intermediate and power range nuclear instrumentation be in excess of one decade.

4.13.1 Method

Reactor power was increased to the specified power level, while continuosly monitoring all the parameters that indicate power level change. A heat balance was then performed. Based on the results of the heat balance, the sensitivity of the linear amplifiers for each power range channel was adjusted, if necessary, and another heat balance was performed. This process continued until indicated power and heat balance power were within 2% of each other.

This test was performed at each of the major power plateaus; 75% FP and 95-100% FP.

4.13.2 Results

The results of the NI calibration met the acceptance criteria.

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[1] FSAR, Table 13-4, #13
[2] PT-120, Section 10.2.1, Rev. 11
[3] SP-113, Section 9.3, Rev. 43
[4] SP-113, Section 9.4, Rev. 43

4.14 Emergency Feedwater Flow Test

No Emergency Feedwater Flow test was performed since no modifications were made which would invalidate the results of the tests performed during Cycle 2 startup.

4.15 Turbine/Generator Operation

No Turbine/Generator operational testing was performed since no changes were made to the turbine or generator which would invalidate the results of the test 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

The Incore Detector Test was performed to verify the adequacy of the system to provide a description of core conditions. The test, performed in accordance with PT-120, is subject to the following acceptance criterion:

> All detector outputs must be consistent and reasonable.[1]

4.17.1 Method

The 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.

The values were taken from the Performance Data Output at 75% FP. From this, the average level current for each detector, including background, was found. Then, each current for each level was divided by the average current. This process was repeated for each detector string. The groupings in Table 4.17-1 were made based on symmetric or near-symmetric locations.

4.17.2 Results

The results of this comparison at 75% FP are shown in Table 4.17-1.

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[1] PT-120, Section 10.2.6, Rev. 11

4.18 Reactor Coolant System Hot Leakage Test

The Reactor Coolant System (RCS) Hot Leakage is monitored on a regular basis during plant operation as required by Technical Specifications. No additional RCS leakage testing was 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 and Effluent Monitoring testing was performed as these systems have been performing normally since initial startup. Therefore, no further testing was required.

TABLE 4.5-1 SUMMARY OF TEMPERATURE, MODERATOR, AND DOPPLER COEFFICIENTS OF REACTIVITY

Full Power Escalation Testing

COEFFICIENT	MEASURED	CALCULATED	ACCEPTANCE REQUIREMENT
Temperature (10^{-4} $\Delta k/k/°F$)	-0.50048[a]	-0.549[c]	--
Moderator (10^{-4} $\Delta k/k/°F$)	-0.34947[a]	-0.402[c]	Less than zero, and less negative than -3.0×10^{-4} $\Delta k/k/°F$
Fuel Temp. Doppler (10^{-5} $\Delta k/k/°F$)	-1.564[b]	-1.51[d]	More negative than -0.90×10^{-5} $\Delta k/k/°F$

Source: [a]PT-120, Encl. 10
[b]PT-120, Encl. 11
[c]Physics Test Manual for CR3 C7, Table 12
[d]Reload Report for CR3 C7, Table 5-1, BAW-1988

TABLE 4.7-2 MAXIMUM LINEAR HEAT RATE BY INCORE DETECTOR LEVEL

Full Power Escalation Testing

In-Core Detector Level	Maximum Linear Heat Rate (kW/ft)		LOCA Limit (kW/ft)
	72.9% FP	100.0% FP	
8	4.55	5.90	15.0
7	7.89	11.14	15.4
6	8.67	11.58	16.3
5	9.11	11.76	16.4
4	9.09	11.75	16.1
3	8.31	11.33	14.1
2	7.69	11.13	11.9
1	5.12	7.63	10.9

Source: SP-104, Encl. 1, Rev. 20

TABLE 4.12-1 SUMMARY OF POWER IMBALANCE DETECTOR CORRELATION TEST
75% FP
Full Power Escalation Testing

Power Level[a] (% FP)	Rod Position (% WD)[a]			Incore Offset (%)		Out-of-Core Offset (%)[b]				Worst Case MLHR (kW/ft)[d]	Worst Case Minimum DNBR[d]
	1-6	7	8	Full[b]	Backup[c]	NI-5	NI-6	NI-7	NI-8		
72.86	100	87.04	52.26	+0.64	+1.06	+0.29	+0.39	+0.21	+0.21	9.76	4.161
72.91	100	87.47	48.90	+1.22	+1.90	+0.88	+0.96	+0.80	+0.80	9.75	3.948
72.88	100	89.60	29.46	+6.63	+11.36	+6.08	+6.07	+5.73	+5.83	9.95	3.909
72.75	100	76.62	65.09	-5.27	-4.49	-6.50	-6.44	-6.34	-6.26	10.20	3.908
72.83	100	68.08	68.31	-13.09	-11.51	-14.47	-14.06	-14.15	-13.98	10.24	3.924
72.82	100	59.45	65.40	-21.63	-17.65	-22.33	-21.58	-21.73	-21.70	10.27	3.912
73.01[e]	-	-	-	+12.91	+11.43	+11.38	+11.10	+11.04	+11.23	10.63	3.684

Sources:
[a] PT-120, Rev 11
[b] PT-120, Encl. 6, Part 1, Rev. 11
[c] PT-120, Encl. 5, Rev. 11
[d] PT-120, Encl. 4, Part 2, Rev. 11
[e] Additional data point to meet the criterion on the sum of the squared differences between the average incore offset and the measured offset (≥612).

TABLE 4.12-2 SUMMARY OF LEAST SQUARES LINEAR REGRESSION ANALYSIS OF POWER RANGE CHANNELS AND BACKUP RECORDER DURING POWER IMBALANCE CORRELATION TEST

75% FP

Full Power Escalation Testing

Detector System	Data Sets	Intercept		Slope (m)	
		Calculated	Allowable	Calculated	Allowable
Out-of-Core[a]					
NI-5	7	-0.87	±2.5	1.00	0.96≤m≤1.10
NI-6	7	-0.78	±2.5	0.97	0.96≤m≤1.10
NI-7	7	-0.92	±2.5	0.97	0.96≤m≤1.10
NI-8	7	-0.83	±2.5	0.97	0.96≤m≤1.10
Backup[b]	7	1.31	—	0.92	—

Sources: [a] PT-120, Encl. 6, Part 4, Rev. 11
[b] PT-120, Encl. 7, Part 1, Rev. 11

TABLE 4.12-3 WORST CASE MINIMUM DNBR AND MAXIMUM LINEAR HEAT RATE
VS. FULL INCORE OFFSET
75% FP
Full Power Escalation Testing

Full Incore Offset (%)	Worst Case Maximum LHR		Worst Case Minimum DNBR	
	(kW/ft)	Location	Value	Location
-0.64	9.76	D-3	4.161	N-11
+1.22	9.75	D-3	3.948	D-3
+6.63	9.95	D-3	3.909	D-3
-5.27	10.20	D-3	3.908	D-3
-13.09	10.24	D-3	3.924	D-3
-21.63	10.27	D-3	3.912	D-4
+12.91	10.63	N-5	3.684	D-5

Source: PT-120, Encl. 4, Part 2, Rev. 11

TABLE 4.17-1 COMPARISON OF INCORE MONITORED ASSEMBLIES' FLUX SHAPES
75% FP
Full Power Escalation Testing

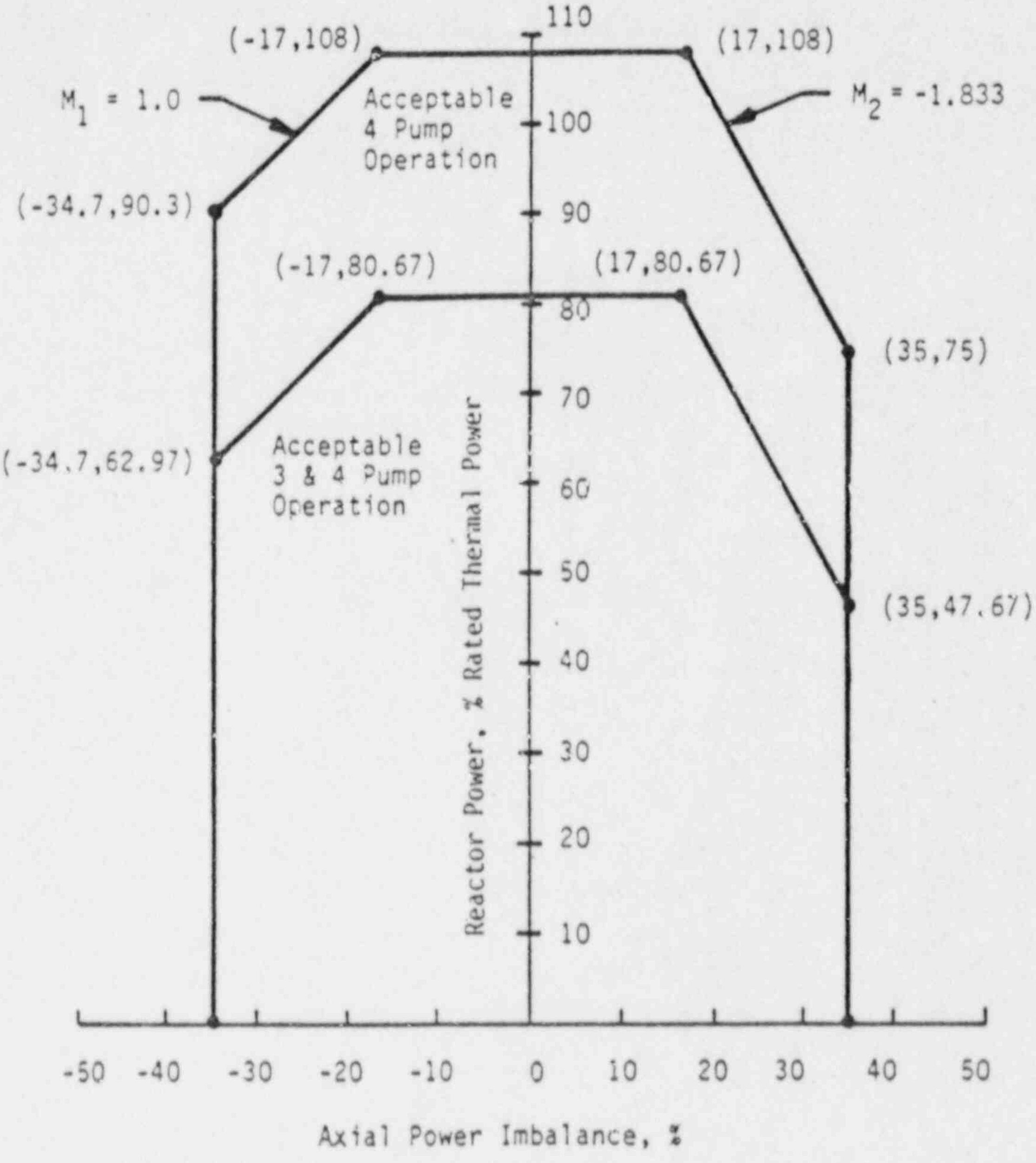
Group	Incore Detector No.	Detector Location	(Detector Level Current)/(Average Detector Current) Level 1	Level 2	Level 3	Level 4	Level 5	Level 6	Level 7	Back-ground (amps)
01	5	E-09	0.765	1.023	1.145	1.188	1.154	1.109	0.616	54
	7	E-07	0.740	1.034	1.156	1.218	1.147	1.104	0.600	9
	9	G-05	0.732	1.044	1.156	1.205	1.157	1.085	0.619	54
	11	K-05	0.741	1.063	1.152	1.184	1.167	1.100	0.593	6
	13	M-07	0.739	1.011	1.144	1.181	1.188	1.140	0.596	17
	16	M-09	0.740	1.011	1.157	1.194	1.169	1.131	0.598	8
	19	K-11	0.734	1.128	1.127	1.209	1.118	1.110	0.575	7
	25	G-11	0.765	1.025	1.144	1.177	1.156	1.114	0.618	0
02	23	F-13	0.701	1.077	1.136	1.225	1.148	1.136	0.578	8
	28	C-10	0.728	1.081	1.195	1.209	1.170	1.063	0.554	66
	32	C-06	0.728	1.101	1.179	1.202	1.167	1.056	0.568	65
	35	F-03	0.701	1.096	1.152	1.229	1.200	1.078	0.545	10
	39	L-03	0.715	1.061	1.168	1.213	1.169	1.096	0.577	67
	43	O-06	0.738	1.091	1.157	1.195	1.183	1.062	0.574	64
	47	O-10	0.727	1.079	1.176	1.195	1.179	1.059	0.585	65
	50	L-13	0.726	1.080	1.168	1.193	1.165	1.101	0.569	66
03	6	F-07	0.754	1.013	1.146	1.200	1.172	1.072	0.643	54
	8	G-06	0.730	1.009	1.142	1.224	1.169	1.097	0.630	7
	15	N-09	0.711	1.033	1.162	1.207	1.205	1.110	0.571	9
	17	M-10	0.750	1.047	1.159	1.178	1.156	1.051	0.658	61
	18	L-11	0.738	1.042	1.148	1.203	1.145	1.091	0.633	5
	20	K-12	0.754	1.060	1.163	1.222	1.187	1.054	0.560	19
	33	D-05	0.719	1.065	1.182	1.227	1.220	1.069	0.519	8
	34	E-04	0.715	1.069	1.180	1.255	1.194	1.056	0.531	6
04	24	F-12	0.724	1.104	1.223	1.130	1.174	1.040	0.605	12
	27	D-10	0.752	1.104	1.218	1.181	1.145	1.046	0.553	12
05	31	B-07	0.689	1.034	1.167	1.224	1.221	1.111	0.554	14
	36	G-02	0.689	1.059	1.159	1.221	1.190	1.106	0.576	66

TABLE 4.17-1 COMPARISON OF INCORE MONITORED ASSEMBLIES' FLUX SHAPES
75% FP
Full Power Escalation Testing (Continued)

Group	Incore Detector No.	Detector Location	(Detector Level Current)/(Average Detector Current)							Back-ground (amps)
			Level 1	Level 2	Level 3	Level 4	Level 5	Level 6	Level 7	
06	38	L-02	0.694	1.074	1.168	1.217	1.204	1.076	0.566	0
	44	P-06	0.681	1.034	1.160	1.220	1.208	1.082	0.614	11
07	22	G-13	0.733	1.037	1.151	1.183	1.204	1.118	0.575	69
	29	C-09	0.693	1.018	1.206	1.224	1.194	1.089	0.576	13
08	40	M-03	0.719	1.043	1.176	1.195	1.188	1.105	0.574	0
	42	O-05	0.718	1.044	1.179	1.198	1.187	1.098	0.575	8
09	1	H-08	0.623	1.053	1.142	1.253	1.165	1.115	0.648	15
	2	H-09	0.749	1.049	1.122	1.197	1.142	1.096	0.646	51
	3	G-09	0.730	1.004	1.162	1.211	1.151	1.108	0.634	10
10	4	F-08	0.711	1.026	1.167	1.196	1.175	1.079	0.645	8
	10	H-05	0.708	1.012	1.158	1.252	1.225	1.042	0.605	18
	12	L-06	0.738	1.032	1.140	1.197	1.141	1.079	0.674	45
	14	N-08	0.742	1.050	1.199	1.257	1.221	1.101	0.429	10
11	37	H-01	0.606	1.016	1.161	1.204	1.218	1.211	0.583	0
	45	R-07	0.653	1.039	1.142	1.147	1.227	1.204	0.588	3
	51	D-14	0.652	1.023	1.195	1.219	1.235	1.135	0.541	4
	52	C-13	0.627	1.068	1.222	1.240	1.269	1.093	0.479	8
12	26	E-11	0.821	1.054	1.175	195	1.166	1.041	0.549	9
	30	B-08	0.685	1.033	1.170	1.206	1.211	1.138	0.556	13
	41	N-04	0.756	1.099	1.213	1.223	1.221	1.114	0.377	66
13	21	H-13	0.734	1.027	1.168	1.206	1.198	1.108	0.561	10
	46	R-10	0.600	1.012	1.148	1.231	1.241	1.161	0.607	0
	48	O-12	0.646	1.054	1.177	1.232	1.244	1.123	0.526	3
	49	M-14	0.665	1.058	1.177	1.197	1.239	1.117	0.547	3

Source: Group 1, 75% FP
BKGD1, SPDNR1

Trip Setpoint for Nuclear Overpower Based on RCS Flow and Axial Power Imbalance
Full Power Escalation Testing



Source: TSCRN-152
Figure 2.2-1

Figure 4.7-1

COMPARISON OF MEASURED AND PREDICTED RADIAL POWER
PEAKING FACTORS WITH 3D EQUILIBRIUM XENON AT
75% FULL POWER
Full Power Escalation Testing

	Measured	Predicted
Control Rod Group Positions Groups 1-6 Group 7 Group 8	 100.0% WD 90.4% WD 47.1% WD	 100.0% WD 93.3% WD 41.0% WD
Core Power Level	72.9% FP	75% FP
Boron Concentration	1574 ppm	-
Core Burnup	3.3 EFPD	3.0 EFPD
Axial Imbalance	0.53% FP	-1.46% FP
Maximum Quadrant Tilt	1.61%	-

	H	K	L	M	N	O	P	R
8	0.86 0.92 6.97							
9	1.14 1.17 2.63	0.92 0.95 3.26						
10	0.92 0.95 3.26	1.19 1.18 -0.80	0.88 0.92 4.54					
11	1.26 1.20 -4.70	0.95 0.98 3.15	1.24 1.21 -2.40	1.12 1.17 4.46				
12	0.90 0.96 6.66	1.31 1.26 -3.80	0.93 0.97 4.30	1.34 1.28 -4.40	1.14 1.11 -2.60			
13	1.30 1.30 0.00	1.15 1.19 3.47	1.30 1.28 -1.50	1.10 1.10 0.00	1.13 1.10 -2.60	0.45 0.47 4.44		
14	1.09 1.08 -0.90	1.24 1.22 -1.60	0.91 0.94 3.29	0.90 0.89 -1.10	0.46 0.47 2.17			
15	0.79 0.77 -2.50	0.54 0.53 -1.80	0.35 0.37 5.71					

Source: PT-120, Encl. 2, Parts 1 & 2

x.xx	Measured Results
x.xxx	Predicted Results
x.xx	% Deviation

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

Figure 4.7-2

COMPARISON OF MEASURED AND PREDICTED MAXIMUM TOTAL POWER
PEAKING FACTORS WITH 3D EQUILIBRIUM XENON AT
75% FULL POWER
Full Power Escalation Testing

	Measured	Predicted
Control Rod Group Positions Groups 1-6 Group 7 Group 8	 100.0% WD 90.4% WD 47.1% WD	 100.0% WD 93.3% WD 41.0% WD
Core Power Level	72.9% FP	75% FP
Boron Concentration	1574 ppm	-
Core Burnup	3.3 EFPD	3.0 EFPD
Axial Imbalance	0.53% FP	-1.46% FP
Maximum Quadrant Tilt	1.61%	-

	H	K	L	M	N	O	P	R
8	1.05 1.09 3.81							
9	1.36 1.39 2.21	1.09 1.12 2.75						
10	1.08 1.13 4.63	1.44 1.40 -2.78	1.04 1.09 4.81					
11	1.56 1.44 -7.69	1.12 1.17 +4.46	1.48 1.45 -2.03	1.32 1.40 6.06				
12	1.11 1.15 3.60	1.58 1.52 -3.80	1.11 1.18 6.31	1.65 1.55 -6.06	1.38 1.36 -1.45			
13	1.56 1.58 1.28	1.37 1.43 4.38	1.56 1.55 -0.64	1.30 1.33 2.31	1.39 1.35 -2.88	0.56 0.58 3.57		
14	1.32 1.31 -0.76	1.50 1.49 -0.67	1.10 1.14 3.64	1.11 1.09 -1.80	0.56 0.57 1.79			
15	0.96 0.94 -2.08	0.65 0.65 0.00	0.43 0.45 4.65					

Source: PT-120, Encl. 2, Parts 1 & 3

x.xx	Measured Results
x.xx	Predicted Results
x.xx	% Deviation

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

Figure 4.7-3

COMPARISON OF MEASURED AND PREDICTED RADIAL POWER
PEAKING FACTORS WITH 3D EQUILIBRIUM XENON AT
100% FULL POWER
Full Power Escalation Testing

	Measured	Predicted
Control Rod Group Positions Groups 1-6 Group 7 Group 8	 100.0% WD 90.9% WD 34.9% WD	 100.0% WD 90.1% WD 27.9% WD
Core Power Level	100.0% FP	100% FP
Boron Concentration	1486 ppm	1541 ppm
Core Burnup	6.6 EFPD	4.0 EFPD
Axial Imbalance	-1.69% FP	-3.12% FP
Maximum Quadrant Tilt	1.25%	0.00%

	H	K	L	M	N	O	P	R
8	0.86 0.93 7.53							
9	1.15 1.17 1.74	0.92 0.96 4.35						
10	0.92 0.96 4.35	1.18 1.18 0.00	0.89 0.93 4.49					
11	1.24 1.19 -4.03	0.94 0.98 4.26	1.23 1.21 -1.63	1.13 1.17 3.54				
12	0.90 0.95 5.55	1.29 1.25 -3.10	0.94 0.98 4.26	1.34 1.27 -5.22	1.12 1.10 -1.79			
13	1.31 1.29 -1.53	1.16 1.19 2.59	1.28 1.27 -0.78	1.10 1.10 0.00	1.14 1.09 -4.39	0.46 0.48 4.35		
14	1.10 1.08 -1.82	1.23 1.22 -0.81	0.92 0.95 3.26	0.92 0.90 -2.17	0.48 0.48 0.00			
15	0.79 0.77 -2.53	0.55 0.54 -1.82	0.37 0.38 2.70					

Source: PT-120, Encl. 2, Parts 1 & 2

x.xx	Measured Results
x.xx	Predicted Results
x.xx	% Deviation

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

Figure 4.7-4

COMPARISON OF MEASURED AND PREDICTED MAXIMUM TOTAL POWER PEAKING FACTORS WITH 3D EQUILIBRIUM XENON AT 100% FULL POWER
Full Power Escalation Testing

	Measured	Predicted
Control Rod Group Positions Groups 1-6 Group 7 Group 8	 100.0% WD 90.9% WD 34.9% WD	 100.0% WD 90.1% WD 27.9% WD
Core Power Level	100.0% FP	100.0% FP
Boron Concentration	1486 ppm	1541 ppm
Core Burnup	6.6 EFPD	4.0 EFPD
Axial Imbalance	-1.69% FP	-3.12% FP
Maximum Quadrant Tilt	1.25%	0.00%

	H	K	L	M	N	O	P	R
8	1.03 1.10 6.80							
9	1.33 1.38 3.76	1.07 1.12 4.67						
10	1.06 1.13 6.60	1.39 1.40 0.72	1.02 1.09 6.86					
11	1.50 1.43 -4.67	1.07 1.16 8.41	1.41 1.42 0.71	1.29 1.39 7.75				
12	1.08 1.15 6.48	1.51 1.49 -1.32	1.09 1.11 1.83	1.59 1.53 -3.77	1.31 1.35 3.05			
13	1.53 1.56 1.96	1.33 1.42 6.77	1.49 1.52 2.01	1.26 1.32 4.76	1.35 1.34 -0.74	0.56 0.58 3.57		
14	1.28 1.30 1.56	1.45 1.49 2.76	1.07 1.14 6.54	1.08 1.09 0.93	0.56 0.58 3.57			
15	0.94 0.95 1.06	0.64 0.65 1.56	0.44 0.46 4.54					

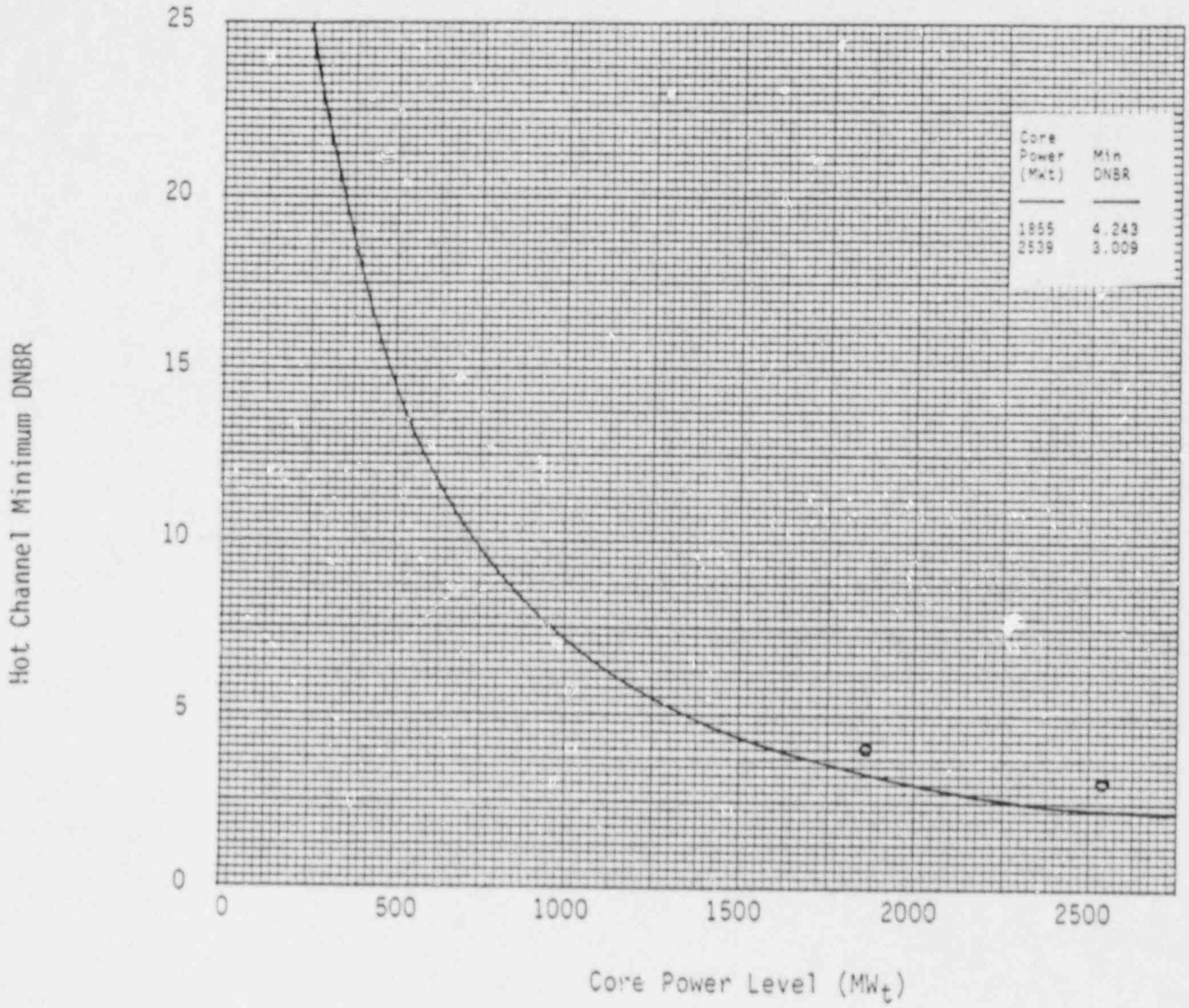
Source: PT-120, Encl. 2, Parts 1 & 3

x.xx	Measured Results
x.xx	Predicted Results
x.xx	% Deviation

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

Figure 4.7-5

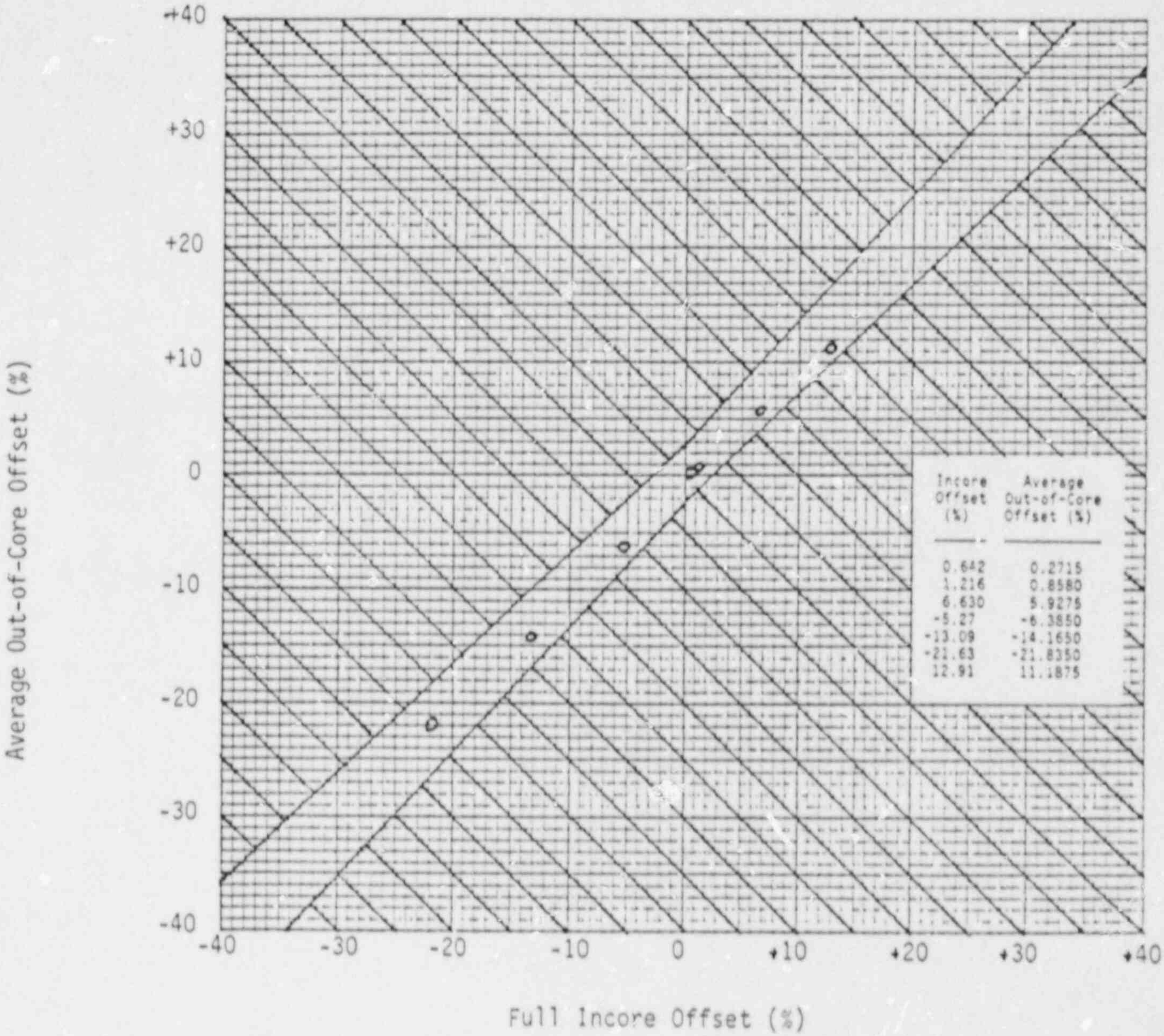
HOT CHANNEL MINIMUM DNBR VS. CORE POWER LEVEL
Full Power Escalation Testing



Source: Group 67

Figure 4.7-6

AVERAGE OUT-OF-CORE OFFSET VS. FULL INCORE OFFSET
75% FP
Full Power Escalation Testing

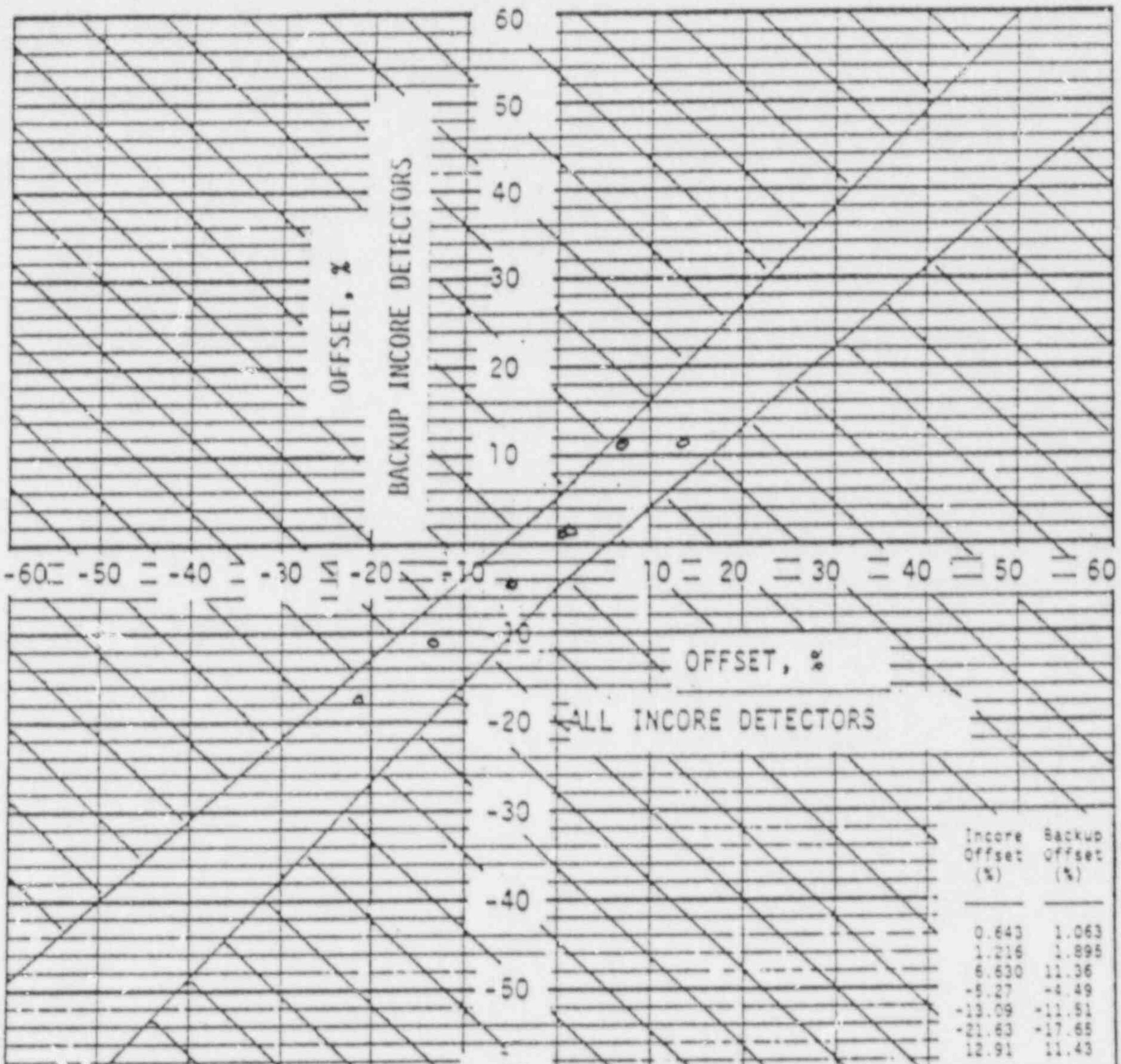


Source: PT-120
Encl. 6, Part 1

Figure 4.12-1

BACK UP INCORE OFFSET VS. FULL INCORE OFFSET
75% FP
Full Power Escalation Testing

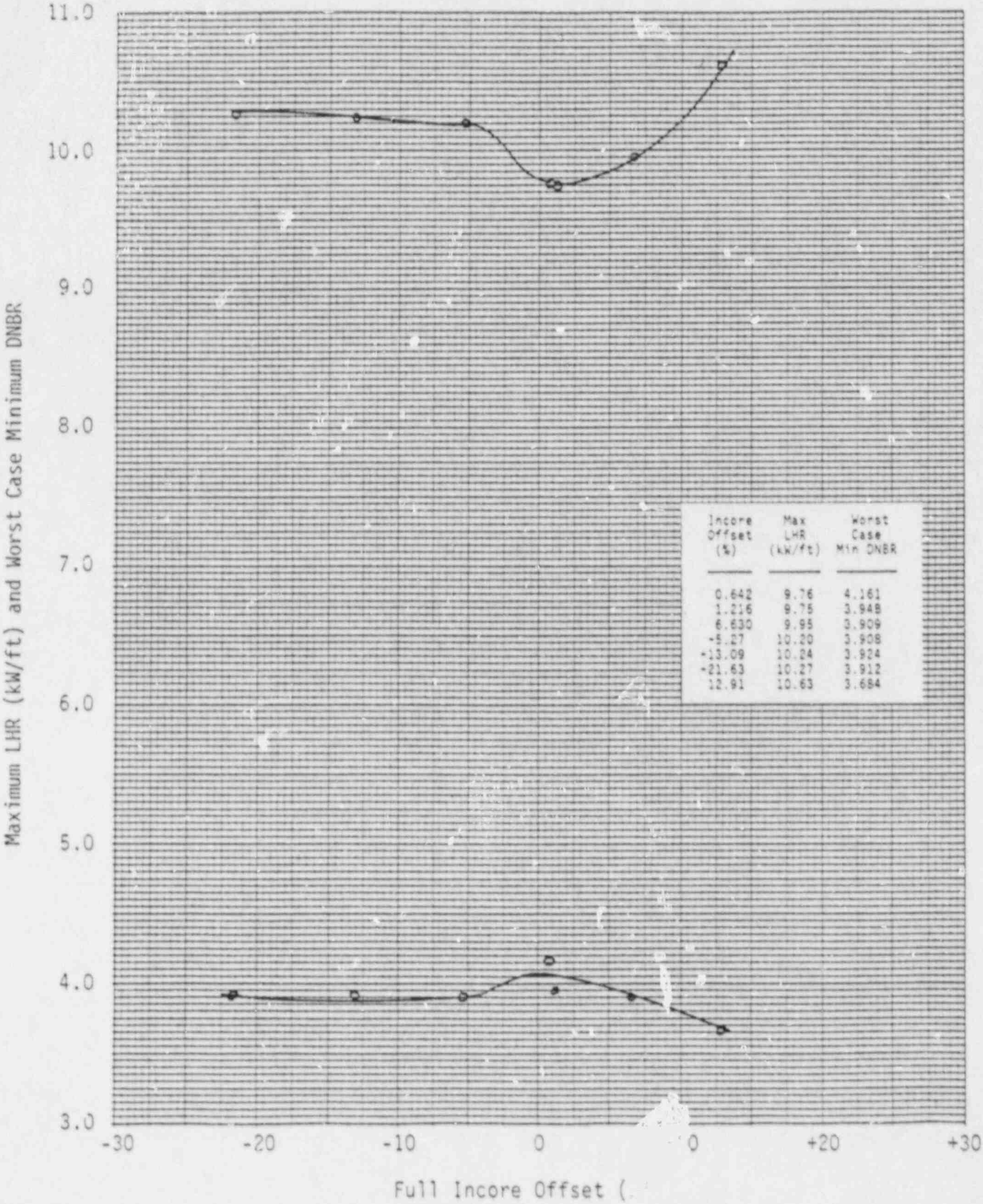
BACKUP INCORE OFFSET
VS
FULL INCORE OFFSET



Incore Offset (%)	Backup Offset (%)
0.643	1.063
1.216	1.895
6.630	11.36
-5.27	-4.49
-13.09	-11.51
-21.63	-17.65
12.91	11.43

Figure 4.12-2

MAXIMUM LHR AND WORST CASE MINIMUM DNBR VS FULL INCORE OFFSET
75% FP
Full Power Escalation Testing



Incore Offset (%)	Max LHR (kW/ft)	Worst Case Min DNBR
0.642	9.76	4.161
1.216	9.75	3.948
6.630	9.95	3.909
-5.27	10.20	3.908
-13.09	10.24	3.924
-21.63	10.27	3.912
12.91	10.63	3.684

Figure 4.12-3

Source: PT-120
Encl. 4, Part 2

Florida Power
CORPORATION

Walter S. Wilgus
Vice President
Nuclear Operations

May 3, 1988
3F0588-02

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Cycle Seven Startup Report

Dear Sir:

Florida Power Corporation (FPC) hereby submits the Crystal River Unit 3 Cycle Seven Startup Report. This report is submitted in accordance with Technical Specification 6.9.1.3 and Regulatory Guide 10.1, Item 170.

If you have any questions concerning this report, please contact this office.

Sincerely,

W S Wilgus

W.S. Wilgus, Vice President
Nuclear Operations

KRW/dhd
Attachment

xc: Dr. J. Nelson Grace
Regional Administrator, Region II

Mr. T.F. Stetka
Senior Resident Inspector

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