METROPOLITAN EDISON COMPANY SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

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July 31, 1978
GQL 1285

Director of Nuclear Reactor Regulation
Attn: R. W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

Enclosed please find a copy of the TMI-1 Cycle 4 Startup Physics Test Report, submitted in accordance with our letter of April 10, 1978 (GQL 0658).

Sincerely,

J. G. Herbein
Vice President-Generation

JGH:WSS:dkf

Enclosure

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## 1.0 Core Performance - Measurements at Zero Power - Summary

Core performance measurements were conducted during the Zero Power Test Program which began on April 28, 1978 and ended on May 1, 1978. This section presents a summary of the zero power measurements. In all cases, the applicable test and Technical Specifications limits were met. A data summary is included at the end of Section 1.

a. Initial Criticality

Initial criticality was achieved on April 28, 1978 at reactor conditions of 532°F and 2155 psig. Control Rod groups 1 through 6 were withdrawn to 100% and group 7 was positioned at 92% withdrawn. Control Rod group 8 was positioned at 37% withdrawn. Criticality was achieved by deborating the Reactor Coolant from 1808 ppm to 1227 ppm. Initial criticality was achieved in an orderly manner and good agreement was found between the predicted critical boron concentrations of 1200 ppm at 75% on group 7 and the measured critical boron concentration of 1227 at 92% on group 7.

b. Nuclear Instrumentation Overlap

At least two decades overlap was measured between the source and intermediate range neutron detectors as required by Technical Specifications.

c. Reactimeter Checkout

An on-line functional check of the Mod-Comp reactimeter (using NI-3) was performed after initial criticality. Reactivity calculated by the reactimeter was within 5% of the core reactivity determined from doubling time measurements.

d. All Rods Out Critical Boron Concentration

The measured all rods out critical boron concentration of 1231 ppmB was in good agreement with the calculated value of 1226 ppmB.

e. Temperature Coefficient Measurement

The measured temperature coefficients of reactivity at 532°F, zero power were within the acceptance criteria limits over the range of boron concentrations and rod positions that the measurements were made.

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f. Regulating Control Rod Worth Measurements

The measured results for control rod worths of groups 5, 6 and 7 conducted at zero power 532°F using the boron/rod swap methods were in good agreement with predicted values except for group 5. The maximum deviation between measured and predicted worths was +14.5% which was for group 5 worth. In retrospect, this is felt to be due to the higher power in the center of the core which was discovered during power escalation testing.

g. Soluble poison worth

The measured results for the soluble poison differential worth at 532°F were, on the average, 8% higher than the predicted values.

h. Shutdown Margin Verification

Minimum shutdown margin verification measurements were performed at zero power 532°F using the rod drop technique to determine the total worth of the safety groups. The shutdown margin with the most reactive rod stuck out of the core was calculated to be 1.07% $\Delta K/K$. This meets the acceptance criteria of 1% $\Delta K/K$ shutdown margin. The measured value was determined to be low due to calculational errors in the reactimeter program which only occurred with extremely large flux changes like those experienced when tripping the safety rods.

i. Ejected Control Rod Worth

To determine whether a real core tilt exists, four ejected rods from symmetric core locations were measured. The worth of the rod in location N-12 was measured using both the boron swap and rod swap methods. The worth of the three other rods symmetric to location N-12 (N-4, D-4, D-12) were measured using the rod swap method. From these measurements it was determined that a core tilt did exist and that the control rod in location D-4 had the highest ejected rod worth. In order to verify the worth of the rod in location D-4, the rod was deborated from 100% to 0% withdrawn with group 5 rods at 0% withdrawn.

The error adjusted maximum ejected rod worth for group 7 rod 12 was 0.98% $\Delta K/K$ which meets the Tech. Spec. requirement of being less than 1.0% $\Delta K/K$. The measured worths of the four ejected rods, without error adjustments, were in good agreement with the predicted value of 0.81% $\Delta K/K$.

## Zero Power Physics Test

## Cycle 4

## Data Summary

| Parameter | Calculated Value | Measured Value |
|---|---|---|
| Critical Boron | 1200 ± 100 ppm @ 75% gp. 7 | 1227 ppm @ 91% gp. 7 |
| Sensible Heat | N/A | $3.3 \times 10^{-7}$ amps |
| NI Overlap | >1 decade | >1 decade |
| All Rods Out Boron Concentration | 1226 ± 100 ppm | 1231 ppm |
| Temperature Coefficient (1234 ppm) | $-8.5 \times 10^{-4}$% ΔK/K/°F ± 4 pcm/°F | $+ 2.6 \times 10^{-4}$% ΔK/K/°F |
| Moderator Coefficient | $< + 5 \times 10^{-3}$% ΔK/K/°F | $+ 2.26 \times 10^{-3}$% ΔK/K/°F |
| Integral Rod Worths (532°F) GP5-7 | 3.61 ± 0.36% ΔK/K | 3.97% ΔK/K |
| Group 7 | 1.37 ± 0.21% ΔK/K | 1.48% ΔK/K |
| Group 6 | 1.00 ± 0.15% ΔK/K | 1.07% ΔK/K |
| Group 5 | 1.24 ± 0.19% ΔK/K | 1.42% ΔK/K |
| Temperature Coefficient (880 ppm) | $-11.9 \times 10^{-3}$% ΔK/K/°F ± 4 pcm/°F | $-10.8 \times 10^{-3}$% ΔK/K/°F |
| Moderator Coefficient | $-9.9 \times 10^{-3}$% ΔK/K/°F ± 4 pcm/°F | $-8.8 \times 10^{-3}$% ΔK/K/°F |
| Error Adjusted Maximum of Symmetric Ejected Rod Worths | <1.0% ΔK/K | 0.98% ΔK/K |

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## 2.0 Core Performance - Measurements at Power - Summary

This section summarizes the physics tests conducted with the reactor at power. Testing is performed at power plateaus of 40, 75 and 100% thermal core power. Operation in the power range began on May 2, 1978.

Periodic measurements and calibrations were performed on the plant nuclear instrumentation during the escalation to power. The four power range detector channels were calibrated based upon primary and secondary plant heat balance measurements. Testing of the incore nuclear instrumentation was performed to ensure that all detectors were functioning properly and that the detector outputs were processed correctly by the plant computer. Core axial imbalance determined from the incore instrumentation system was used to calibrate the out of core detector imbalance indication.

The major physics measurements performed during power escalation consist of determining the moderator and power doppler coefficients of reactivity, determining the associated power distributions effected by the simulated dropped control rod, and obtaining detailed radial and axial imbalances. Values of minimum DNBR and maximum linear heat rate are monitored throughout the test program to ensure that core thermal limits would not be exceeded.

### a. Nuclear Instrumentation Calibration at Power

The power range channels were calibrated as required during the startup program to indicate within two percent of the total core power as determined by a primary or secondary plant heat balance. These calibrations were required due to power level, boron and/or control rod configuration changes during testing. While at indicated 40% FP, it was determined that power was higher than 40% from the MW net electrical output. The heat balance calculation was checked and an error was found and corrected. The actual power had been approximately 46% and was returned to 40% FP.

### b. Incore Detector Testing

Tests conducted on the incore detector system demonstrated that all detectors were functioning as expected, that symmetrical detector readings agreed within acceptable limits and that the plant computer applied the correct background, length and depletion correction factors. While at 40% FP, string 5 of the SPND's was found to have bad indication from 1 of its 7 levels. The bad level was deleted from computer scan.

c. Core Power Distributions

Core power distribution measurements were conducted at 40%, 75% and 100% full power under steady state equilibrium xenon conditions for specified control rod configurations. The results at 40% and 75% full power indicated radial and total peaks higher than predicted and outside the acceptable range. At 75% full power the radial peak was 9.6% higher than predicted and the total peak was 11.3% higher than predicted. The acceptance criteria are 5% and 7.5% respectively. Reactor power was limited to 91% full power until Babcock and Wilcox analyzed the core power distribution data and a Technical Specification amendment was issued based on new error bands of 11% and 13% for radial and total peaks respectively. The core power distribution data for 100% full power was obtained at approximately 25 EFPD and the radial and total peaks were measured to be 8.9% and 6.9%, respectively, higher than the predicted values.

The results of the minimum DNBR and maximum LHR analyses are given in Table 4.5.1. All quadrant power tilts and axial core imbalances measured during the power distribution tests were within the Technical Specification and normal operational limits.

d. Dropped Rod Power Distribution Verification

The dropped control rod test performed at 40% met all required acceptance criteria. The core power distributions and thermal conditions that developed during dropped rod operation at power showed adequate margins to minimum DNBR and maximum linear heat rate limits.

e. Power Imbalance Detector Correlation

The results of the Axial Power Shaping Rod (APSR) scans performed at 40% and 75% F.P. show that an acceptable incore versus out-of-core offset slope of 1.15 is obtained by using a gain factor of 4.904 in the power range scaled difference amplifiers. The measured values of minimum DNBR and maximum linear heat rate for various axial core imbalances indicate that the Reactor Protection Trip Setpoints provide adequate protection to the core. Imbalance calculations using the backup recorder provide a reliable alternate to computer calculated values.

f. Rod Reactivity Worth Measurements

Differential control rod reactivity worth measurements were performed in conjunction with the reactivity coefficients, the measured rod worths agreed with the design values.

g. Reactivity Coefficients at Power

The temperature coefficient measured at 100% FP was -9.03 pcm/$^{o}$F. The measured power doppler coefficient at 100% FP was -10.68 pcm/% FP.

## 3.0 Core Performance - Measurements at Zero Power

This section presents the detailed results and evaluations of the zero power physics testing. The zero power testing included initial criticality, nuclear instrumentation overlap, reactimeter checkout, all rods out critical boron concentration, temperature coefficient measurement, control rod worths, boron worth, shutdown margin verification and ejected control rod worth.

## 3.1 Initial Criticality

Initial criticality for Cycle 4 was achieved on April 28, 1978 at reactor conditions of 532$^{o}$F and 2155 psig. Control rod groups 1 through 4 were previously withdrawn during the heatup to 532$^{o}$F. The initial reactor coolant system (RCS) boron concentration was 1806 ppm.

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Inverse Count Rate Versus Rod Withdrawal

NI-1 and NI-2

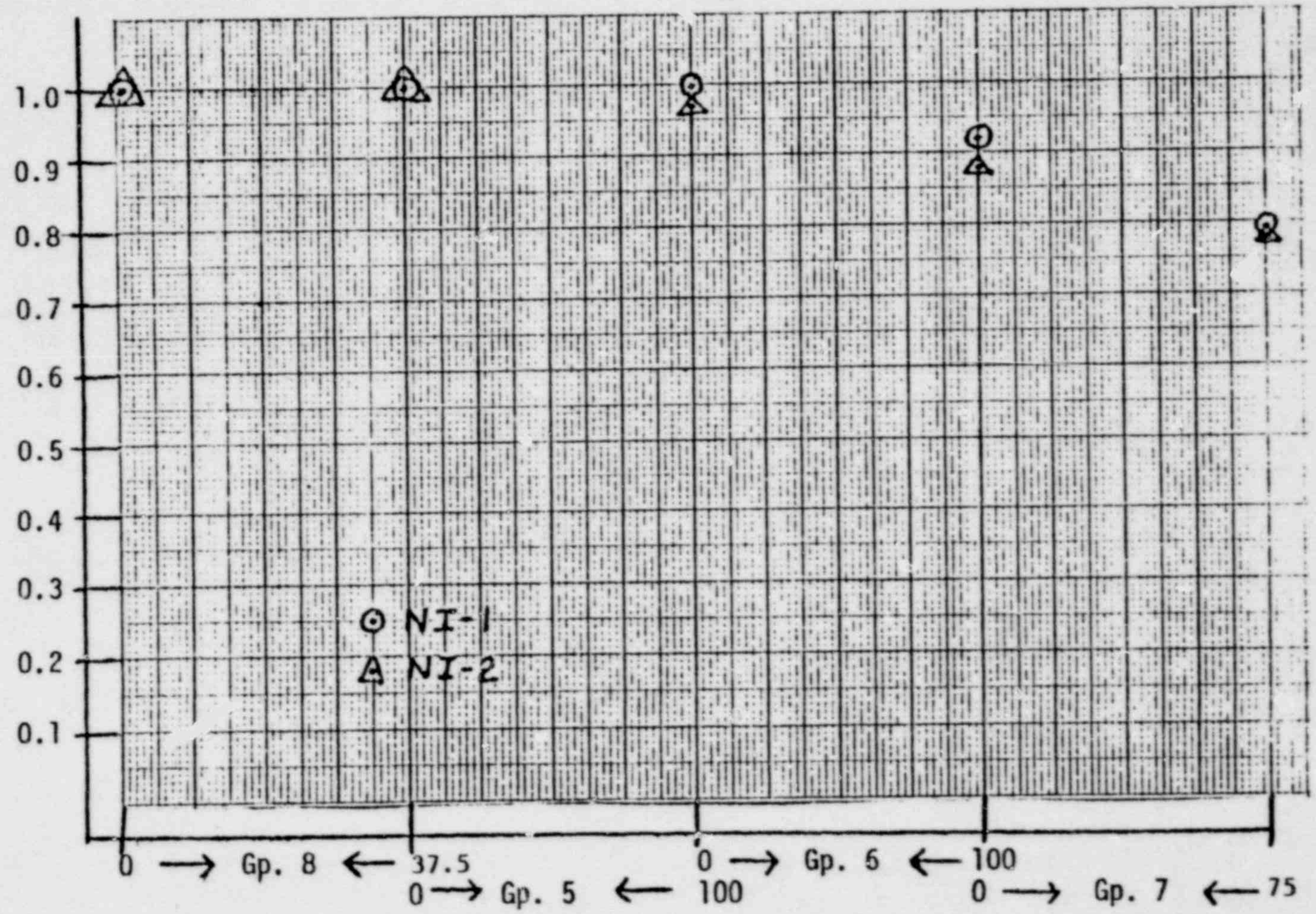




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Figure 3.1-1

Inverse Count Rate vs. Boron Concentration

NI-1

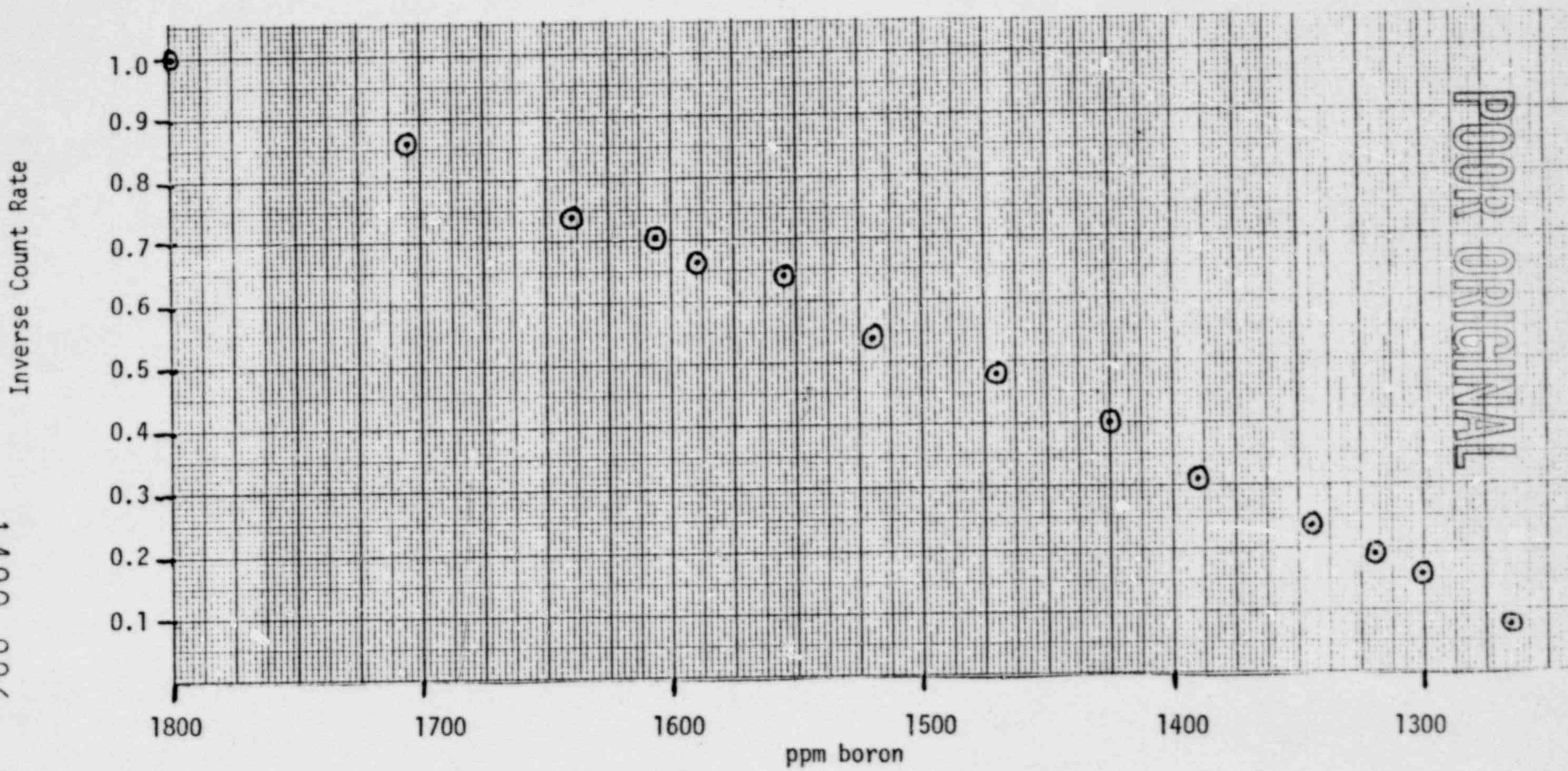


Figure 3.1-2

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Inverse Count Rate vs Boron Concentration

NI-2

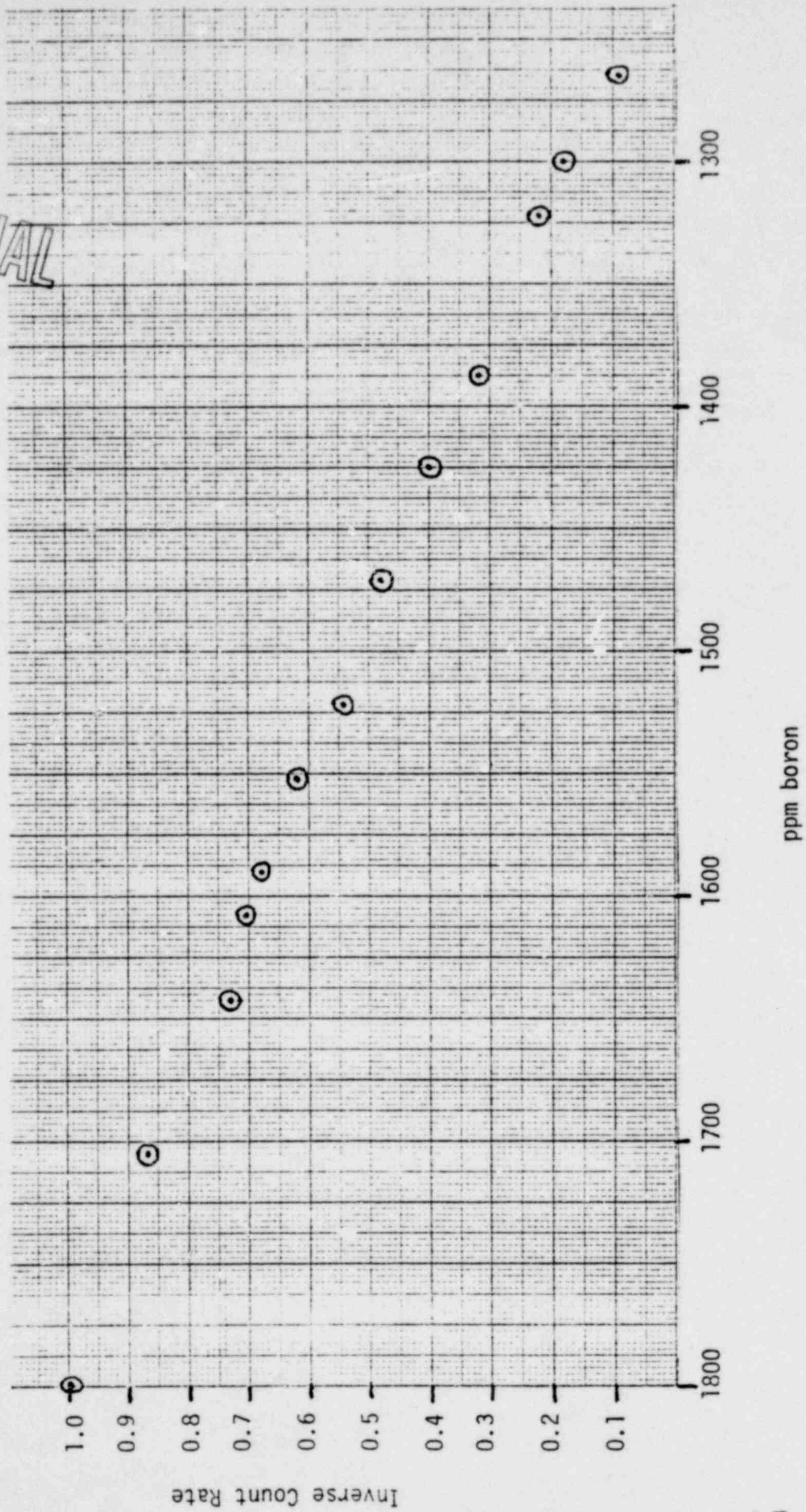


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Figure 3.1-3

Inverse Count Rate vs Deboration

NI-1

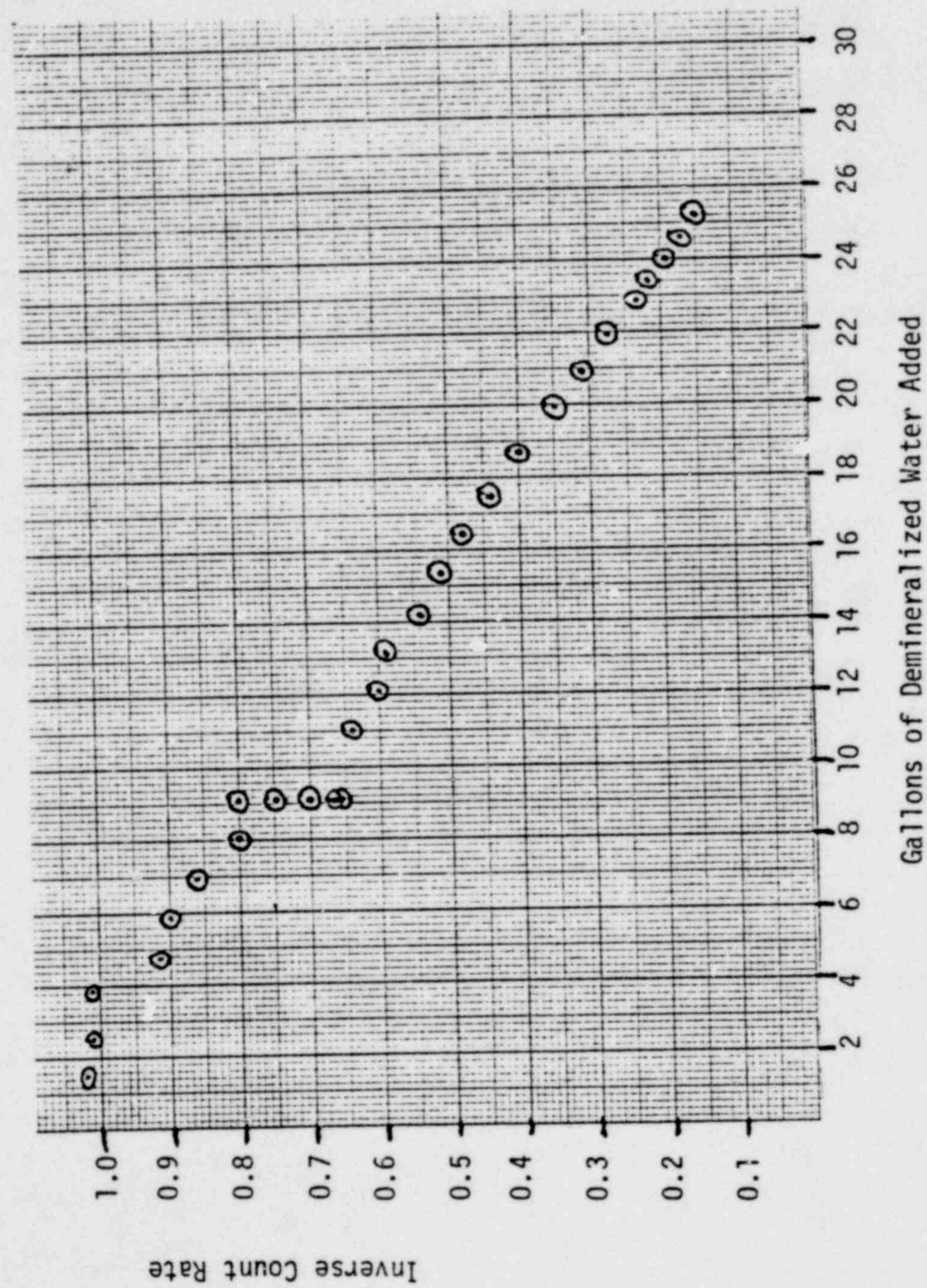


Figure 3.1-4

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Inverse Count Rate vs Deboration

NI-2

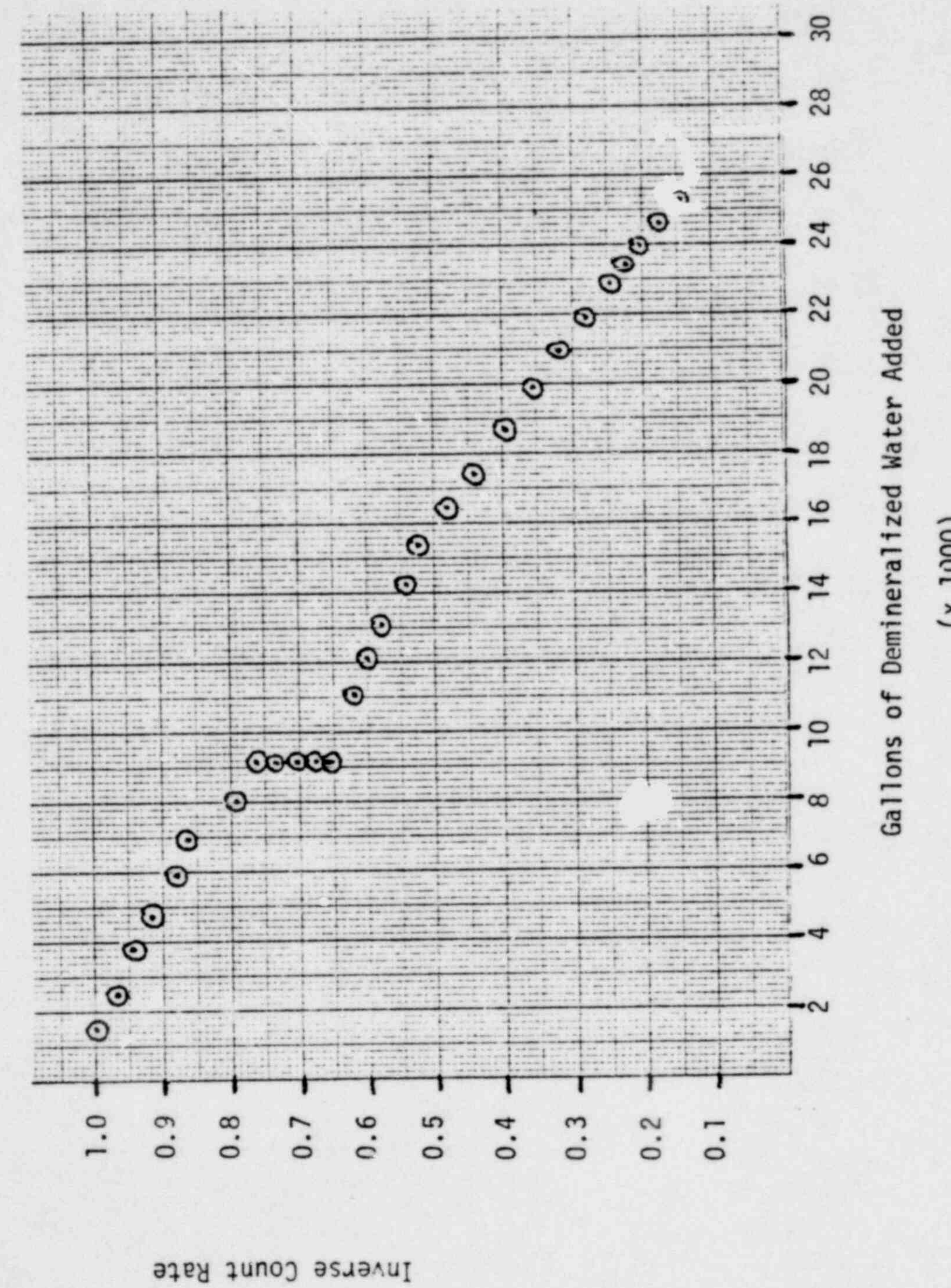


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Figure 3.1-5

The approach to critical began with withdrawing control rod groups 5 and 6 to 100% and positioning group 7 at 75% withdrawn. Control rod group 8 was positioned at 37.5% withdrawn. Criticality was subsequently achieved by deborating the reactor coolant system to a boron concentration of 1227 ppm. The procedure used in the approach to critical is outlined below in three basic steps:

Step 1 Control Rod Withdrawal
Group 8 37.5% withdrawn
Group 5 100% withdrawn
Group 6 100% withdrawn
Group 7 75% withdrawn

Step 2 Deborate using a feed and bleed flow rate of 70 gpm until the inverse count rate is between 0.2 and 0.1.

Step 3 Stop deboration and increase letdown flow to maximum (140 gpm) to enhance mixing between the makeup tank and the reactor coolant system. Achieve initial criticality and position control rod group 7 to control neutron flux as the reactor coolant system boron concentration reaches equilibrium.

Throughout the approach to criticality, plots of inverse multiplication were maintained by two independent persons. Two plots of inverse count rate (ICR) versus control rod position were maintained during control rod withdrawal. Two plots of ICR versus RCS boron concentration and two plots of ICR versus gallons of demineralized water added were maintained during the dilution sequence. At the end of each reactivity addition (boron dilution or control rod withdrawal), count rates were obtained from each startup range neutron detector channel. The ratio of the initial average count rate to the count rate at the end of each reactivity addition is the value plotted.

During control rod withdrawal (step 1) ICR plots versus control rod group position were maintained from the outputs of source range channels NI 1 and 2. The withdrawal interval for each control rod group was limited to no more than half the remaining predicted distance to criticality as determined from the ICR plots.

Deboration of the reactor coolant system was accomplished in two steps as indicated above. First, deboration from 1808 ppm was commenced using a feed and bleed flow rate of 70 gpm (step 2). RC boron samples were taken every 30 minutes and samples from the makeup tank and the pressurizer were taken hourly. Two ICR plots were maintained vs. gallons of demineralized water added, and two plots were maintained vs. RC letdown concentration every 30 minutes. Deboration at a letdown rate of 70 gpm was continued until one of the ICR plots indicated 0.15. At this time, demineralized water additions were stopped and the letdown flow rate was increased to 140 gpm to expedite mixing in the RCS (step 3).



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As the final boron mixing took place it became evident that the reactor was still slightly subcritical. Therefore, group 7 control rods were withdrawn and criticality was achieved at 91% withdrawn.

The inverse count rate plots maintained during the approach are presented in Figures 3.1-1 through 3.1-5. As can be seem from the plots, the response of the source range channels during reactivity additions was very good. Figure 3.1-1 is the plot of ICR versus control rod group withdrawal for data taken from NI channels 1 and 2. Figures 3.1-2 and 3.1-3 are the ICR plots versus RCS boron concentration and Figures 3.1-4 and 3.1-5 are the ICR plots versus gallons of demineralized water added to the RCS, for source range channels NI-1 and NI-2, respectively.

In summary, initial criticality was achieved in an orderly manner. There was good agreement between the measured and the predicted critical boron concentration.

## 3.2 Nuclear Instrumentation Overlap

### a. Purpose

Technical Specification 3.5.1 states tht prior to operation in the intermediate nuclear instrumentation (NI) range, at least one decade of overlap between the source range NIs and the intermediate range must be observed. This means that before the source range count rate equals $10^5$ cps the intermediate range NI must be on scale.

### b. Test Method

To satisfy the above overlap requirements after initial criticality was achieved, core power was increased until the intermediate range channels came on scale. Detector signal response was then recorded for both the source range and intermediate range channels. This was repeated for two more decades until the source range channels approached $10^6$ cps.

### c. Test Results

The results of the initial NI overlap data at 532°F and 2155 psig have shown two decades overlap between the source and intermediate ranges.

### d. Conclusions

The linearity, overlap and absolute output of the intermediate and source range detectors are within specifications and performing satisfactorily. There is at least 2 decades overlap between the source and intermediate ranges, thus satisfying T.S. 3.5.1.

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## 3.3 Reactivity Calculations

### a. Purpose

Reactivity calculations during Cycle 4 test program were performed using the Mod-Comp Reactimeter. After initial criticality and prior to the first physics measurement, an on-line functional check of the reactimeter was performed to verify its accuracy for use in the test program.

### b. Test Method

After initial criticality and nuclear instrumentation overlap were established, intermediate range channel NI-3 was input to the reactimeter and the reactivity calculations were started. After steady state conditions with a constant neutron flux were established, a small amount of positive reactivity was inserted in the core by withdrawing control rod group 7. Stop watches were used to measure the doubling time of the neutron flux and the reactivity inserted was determined from the period-reactivity curves. The measurement was repeated for several values of reactivity inserted by rod group 7, from $\pm 0.025\% \Delta K/K$ to $\pm 0.075\% \Delta K/K$. The reactivities determined from doubling time measurements were compared with the reactivity calculated by the reactimeter.

### c. Test Results

The results of the reactimeter verification measurements are summarized in Table 3.3-1. All the measured values were determined to be satisfactory and showed that the reactimeter was ready for startup testing.

### d. Conclusions

An on-line functional check of the reactimeter was performed after initial criticality. The measured data shows that the core reactivity measured by the reactimeter was in good agreement with the values obtained from neutron flux doubling times.

## COMPARISON OF REACTIMETER AND DOUBLING TIME (DT) REACTIVITY MEASUREMENTS

from NI-3

| Case No. | Measured DT (Sec.) | Measured Reactivity ($10^{-3}$ %$\Delta K/K$) | Calculated Reactivity ($10^{3}$ %$\Delta K/K$) | Percent Difference % |
|---|---|---|---|---|
| 1 | 135 | +33.2 | +33.0 | -0.60 |
| 2 | 194 | 31.4 | 31.8 | +1.27 |
| 3 | 57.3 | +68.0 | +67.8 | 0.29 |
| 4 | 92.4 | 83.0 | 80.0 | 3.61 |

TABLE 3.3-1

### 3.4 All Rods Out Critical Boron Concentration

a. Purpose

The all rods out critical boron concentration measurement is performed to obtain an accurate value for the excess reactivity loaded in the TMI Unit I core and to provide a basis for the verification of calculated reactivity worths. This measurement was performed at system conditions of 532°F and 2155 psig.

b. Test Method

The Reactor Coolant System was borated to an almost all rods out condition with control rod groups 1-6 at 100% withdrawn and with group 7 maintaining criticality at approximately 92% withdrawn. Once steady state conditions were established, control rod group 7 was withdrawn to 100% and the resultant reactivity change was measured. The measured boron concentration with group 7 partially inserted was then adjusted to the all rods out configuration using the result of the rod worth measurement to determine the reactivity worth, in terms of ppm boron, of the inserted control rods.

c. Test Results

The results of the measurement at 532°F are tabulated below.

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ALL RODS OUT CRITICAL BORON CONCENTRATION

ppm boron

| Moderator Temperature | Calculated Result | Measured Result |
|---|---|---|
| 532$^{o}$F | 1226 ppm | 1231 ppm |

The measured boron concentration with group 7 positioned at 92% was 1227 ppm. An additional 4 ppm was added to this value that is derived from 0.041%$\Delta$K/K due to group 7 withdrawal to 100%, using a differential boron worth of 1.021%$\Delta$K/K per 100 ppm boron.

d. Conclusions

The above results show that the measured boron concentrations are in excellent agreement with predicted results of 1226 $\pm$ 100 ppm.

## 3.5 Temperature Coefficient Measurements

a. Purpose

The moderator temperature coefficient of reactivity can be positive, depending upon the soluble boron concentration in the reactor coolant. Because of this possibility, the Technical Specifications state that the moderator temperature coefficient shall not be positive at full power conditions. The moderator temperature coefficient cannot be measured directly, but it can be derived from the core temperature coefficient and a known fuel temperature (isothermal Doppler) coefficient. The temperature coefficient of reactivity was measured for several different boron concentrations at the zero power conditions of 532$^{o}$F and 2155 psig to provide comparison of the moderator temperature coefficient with the design calculations prior to operation in the power range.

b. Test Method

Steady state conditions were established by maintaining reactor flux, reactor coolant pressure, turbine header pressure and core average temperature constant, with the reactor critical at approximately $3x10^{9}$ amps in the intermediate range. (The measurement began with the reactor critical at a slightly higher flux level if a negative feedback effect was expected from a temperature increase or at a lower flux level if a positive feedback effect was expected from a temperature increase). 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 strip chart recorder.

Once steady state conditions were established, a positive heatup rate was started by closing the turbine bypass valves. After the core average temperature increased by about 10°F, core temperature and flux were stabilized and the process was reversed by decreasing the core average temperature to the initial value by opening the turbine bypass valves. This procedure was completed once with control rod group 7 at 89% wd. and then once again at a rod index of zero. Calculation of the temperature coefficient from the measured data was then performed by dividing the change in core reactivity by the corresponding change in core temperature over a specific time period.

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Summary of Isothermal Temperature Coefficient

Measurements at the Zero Power

Conditions of 532°F and 2155 psig

| RC Boron Concentration (ppm) | Control Rod Positions (% withdrawn) | Temperature Coefficient ($x10^{-3}$% ΔK/K°F) Measured Value | Calculated Results |
|---|---|---|---|
| 1234 | Gps 1-6 @ 100<br>Gp 7 @ 89<br>Gp 8 @ 37.5 | +0.26 | -0.85 |
| 880 | Gps 1-4 @ 100<br>Gp 5 @ 19<br>Gps 6-7 @ 0<br>Gp 8 @ 37.5 | -10.8 | -12.0 |

TABLE 3.5-1

c. Test Results

Isothermal temperature coefficient measurements were conducted at two different reactor coolant boron concentrations during the zero power test program. The results of the measurements are summarized in Table 3.5-1. The calculated values are included for comparison.

In all cases the measured results compare favorably with the calculated values.

d. Conclusions

The measured values of the temperature coefficient of reactivity at 532$^{o}$F, zero reactor power are within the acceptance criteria of $\pm 4.0 \times 10^{-3}$ %ΔK/K/$^{o}$F of the predicted value. Calculation of the moderator coefficient indicates that it is well within the limits of Technical Specifications 3.1.7.

## 3.6 Soluble Poison Worth

a. Purpose

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. The differential reactivity worth of the boric acid in terms of ppm boron was measured during the zero power test.

b. Measurements of the differential boron worths at 532$^{O}$F were performed in conjunction with the control rod worth measurements. The control rods worths were measured by the boron swap technique in which a boration deboration rate was established and the control rods were withdrawn/inserted to compensate for the changing core reactivity. The reactimeter was used to provide a continuous reactivity calculation throughout the measurement. The differential boron worth was then determined by summing the incremental reactivity values measured during the rod worth measurements over a known boron concentration range. The average differential boron worth is the measured change in reactivity divided by the change in boron concentration.

c. Test Results

Measurements of the soluble boron differential worth were completed at the zero power condition of 532$^{o}$F. The measured boron worth was 11.02 pcm/ppmB at an average boron concentration 1065 ppmB. The predicted value was 10.2 pcm/ppmB.

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d. <u>Conclusions</u>

The measured results for the soluble poison differential worth at 532°F were in good agreement with the predicted differential worth.

## 3.7 Control Rod Group Worth Measurements

a. <u>Purpose</u>

The total amount of excess reactivity controlled at beginning-of-life (BOL), hot (532°F), clean conditions is 12.95%$\Delta K/K$. During reactor operations, nearly all of the excess reactivity is controlled by the soluble poison systems. Additional control is provided by moveable control rods. This section provides comparison between the calculated and measured results for the control rod group worths.

The layout of the core according to the standard alphabet-numeric mesh showing the inital location of the control rod groups is shown in Figure 3.7-1. The number of control rods and the reactivity control function of each group is given in the Table 3.7-1. The grouping of the control rods shown in Figure 3.7-1 will be used until the end of Cycle 4. Calculated and measured control rod group reactivity worths for the normal withdrawal sequence were determined at reactor conditions of zero power, 532°F and 2155 psi. The measured results were obtained using results of reactivity and group position from the strip chart recorders.

b. <u>Test Method</u>

Control rod group reactivity worth measurements were performed at zero power, 532°F using the rod drop and boron/rod swap methods. The boron/rod swap method was used to measure the differential and integral reactivity worths of control rod groups 5, 6, and 7.

The boron swap method consisted of establishing a deboration rate in the reactor coolant system and compensating for the reactivity changes by inserting the control rod groups in incremented steps. In the rod swap technique (similar to the boron swap method), the reactivity changes caused by moving the rod group being measured are compensated for by moving another rod group. The reactivity changes that occurred during the measurements were calculated by the reactimeter and differential rod worths were obtained from the known reactivity worth versus the change in rod group position. The differential rod worths of each group were then summed to obtain the integral rod group worths.

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c. Test Results

Control rod group reactivity worths were measured at the zero power, 532°F conditions. The boron/rod swap method was used to determine differential and integral rod worths for control rod group 5 - 7 from 100% to 0% withdrawn.

The integral reactivity worths for control rod groups 5 through 7 are presented in Figures 3.7-2 through 3.7-4.

These curves were obtained by integrating the measured differential worth curves. The integral worth of group 8 rods was not measured. The calculated worth for 532°F, zero power is 0.45%$\Delta$K/K. Figure 3.7-5 is a plot of the total reactivity worth of groups 5 through 7 for the normal withdrawal sequence.

Table 3.7-2 provides a comparison between the predicted and measured results for the rod worth measurements. Except for group 5, the results show good agreement between the measured and predicted rod group worths. The maximum deviation between measured and predicted was +14.5%. Table 3.7-2 also provides the predicted rod worths at hot full power conditions with APSR's centered.

d. Conclusions

Differential and integral control rod group reactivity worths are measured using the boron/rod swap and rod drop methods. The measured results at zero power, 532°F indicate satisfactory agreement with the predicted group worths.

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CONTROL ROD GROUP LOCATIONS
CYCLE 4

X

| W | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 | 14 | 15 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A | | | | | | | | | | | | | | | |
| B | | | | | | (3) | | (7) | | (3) | | | | | |
| C | | | | | (1) | | (6) | | (6) | | (1) | | | | |
| D | | | | (7) | | (8) | | (5) | | (8) | | (7) | | | |
| E | | | (1) | | (5) | | (2) | | (2) | | (5) | | (1) | | |
| F | | (3) | | (8) | | (7) | | (5) | | (7) | | (8) | | (3) | |
| G | | | (6) | | (2) | | (4) | | (4) | | (2) | | (6) | | |
| H | | (7) | | (5) | | (5) | | (3) | | (5) | | (5) | | (7) | |
| K | | | (6) | | (2) | | (4) | | (4) | | (2) | | (6) | | |
| L | | (3) | | (8) | | (7) | | (5) | | (7) | | (8) | | (3) | |
| M | | | (1) | | (5) | | (2) | | (2) | | (5) | | (1) | | |
| N | | | | (7) | | (8) | | (5) | | (8) | | (7) | | | |
| O | | | | | (1) | | (6) | | (6) | | (1) | | | | |
| P | | | | | | (3) | | ) | | (3) | | | | | |
| R | | | | | | | | | | | | | | | |

Z

(X) ← Control Rod Group Number

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Figure 3.7-1

CONTROL ROD GROUP 5

Integral Worth at Zero Power, 532$^{o}$F, OEFPD



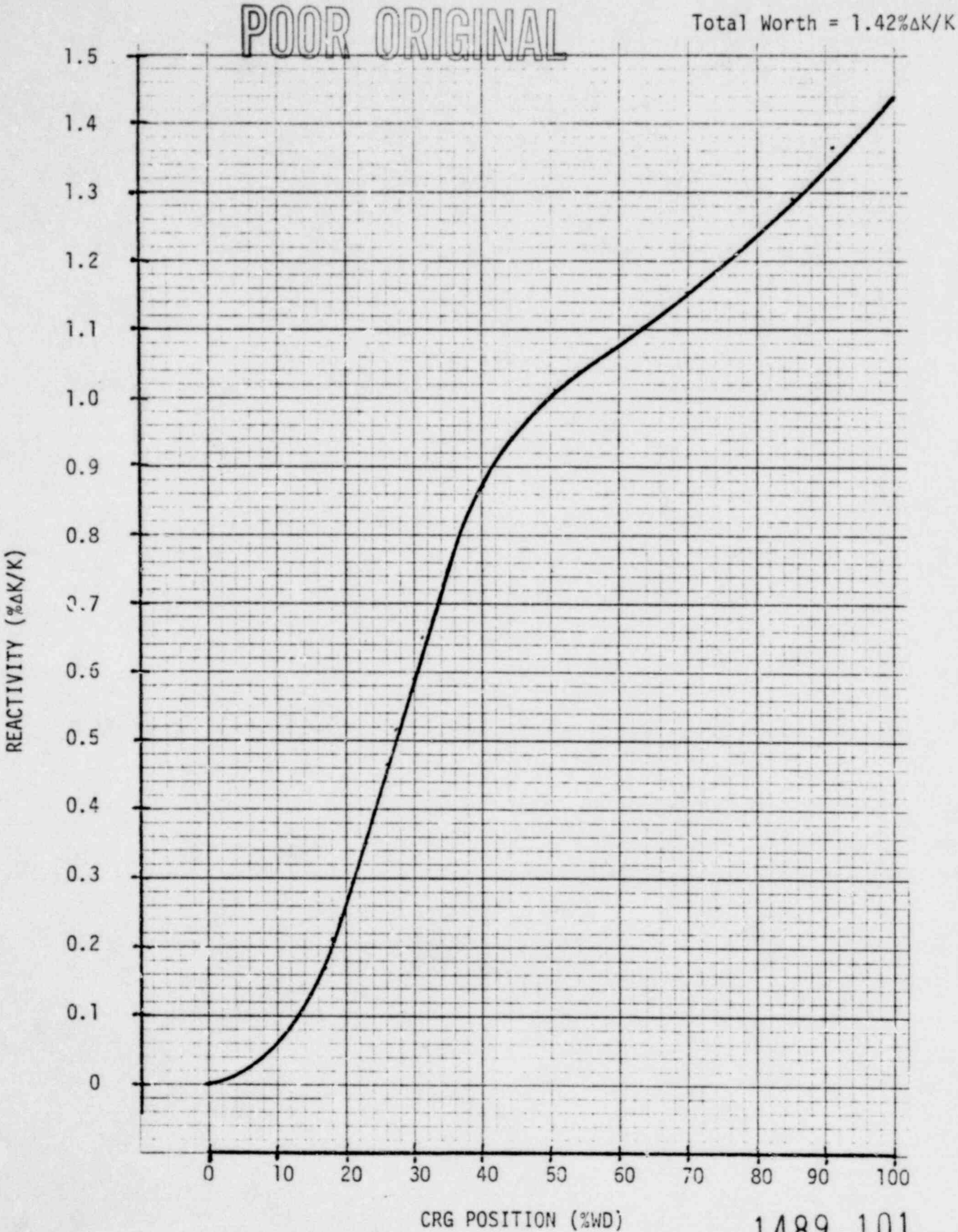


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FIGURE 3.7-2

CONTROL ROD GROUP 6

Integral Worth at Zero Power, 532°F, OEFPD

POOR ORIGINAL

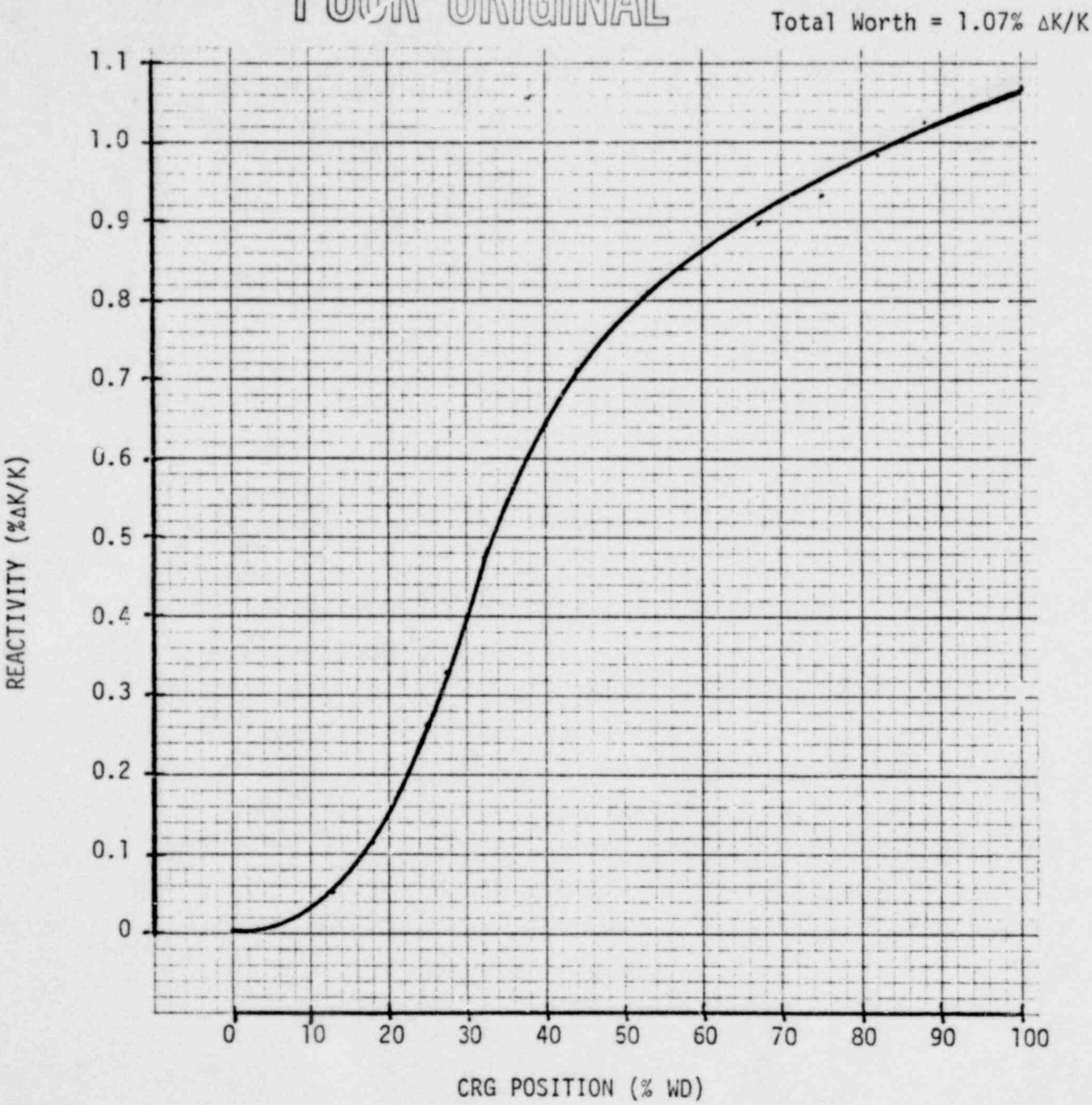


FIGURE 3.7-3

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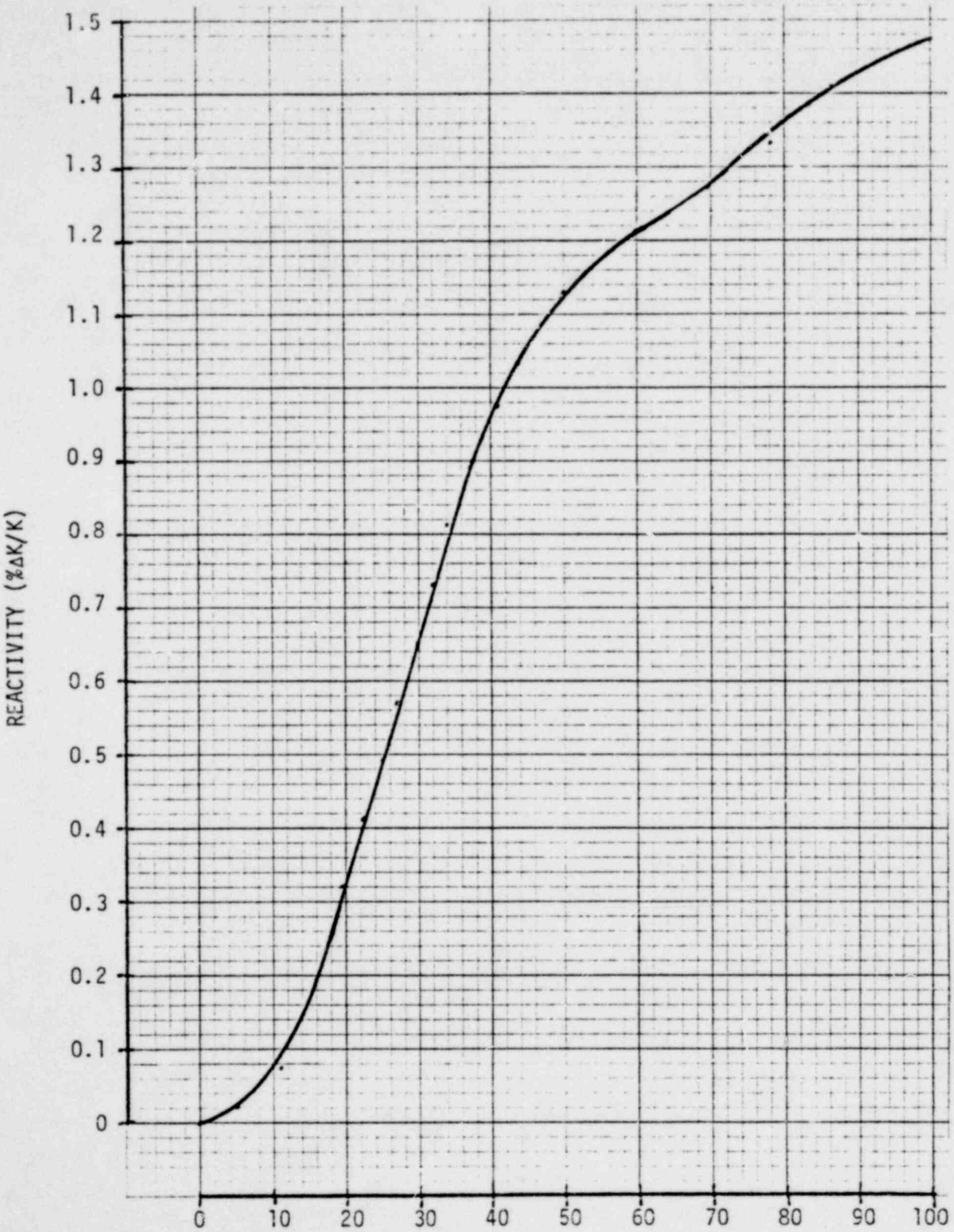
CONTROL ROD GROUP 7

Integral Worth at Zero Power, 532°F, OEFPD

Total Worth = 1.48%ΔK/K






CRG POSITION (% WD)

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FIGURE 3.7-4

TOTAL REACTIVITY WORTH VERSUS ROD WITHDRAWAL
At Zero Power, 532°F, 0EFPD

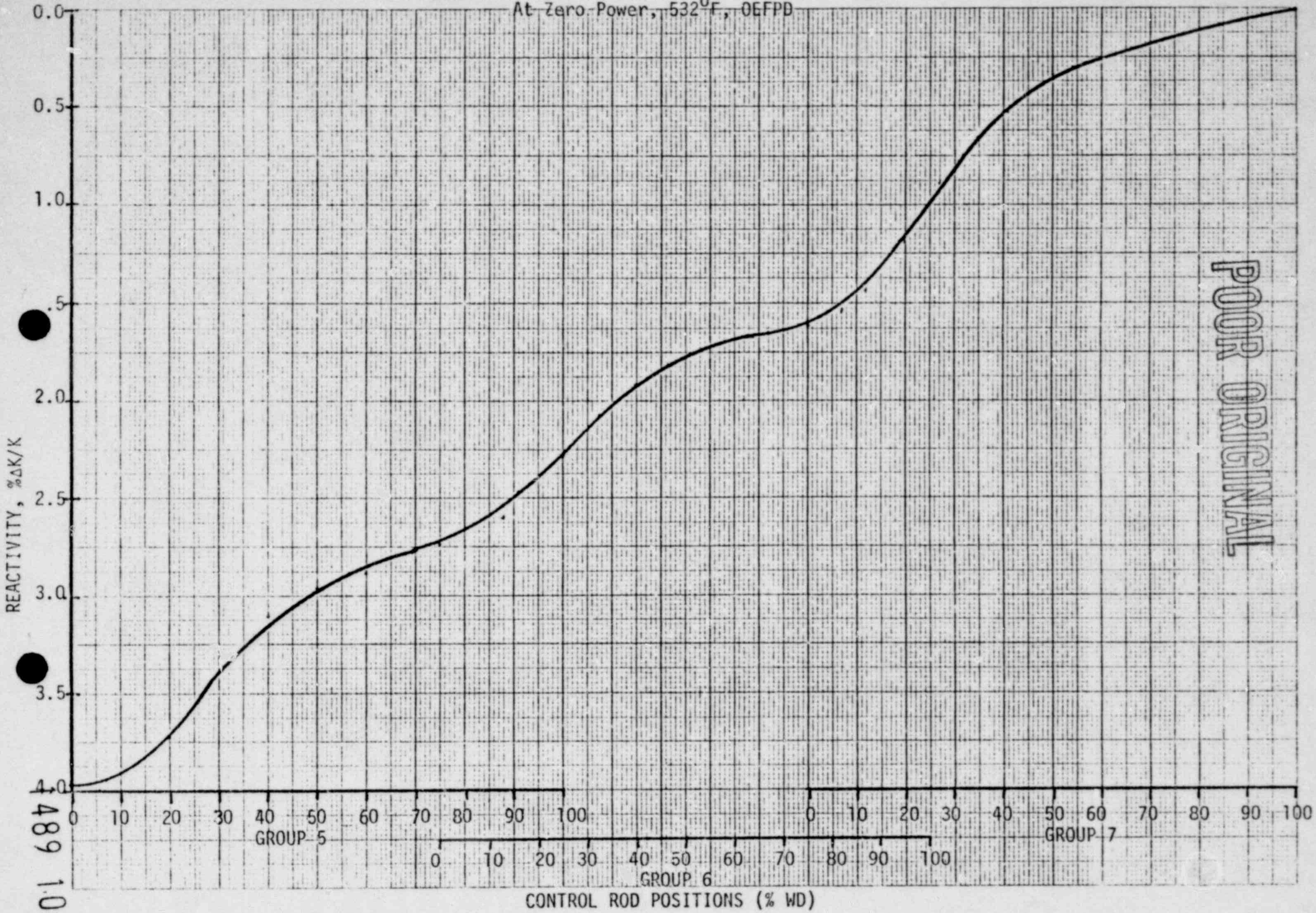




FIGURE 3.7-5

CYCLE 4

REACTIVITY CONTROL

FUNCTION OF CONTROL ROD GROUPS

| Rod Group Number | Number of Rods | Control Function |
|---|---|---|
| 1 | 8 | Safety |
| 2 | 8 | Safety |
| 3 | 9 | Safety |
| 4 | 4 | Safety |
| 5 | 12 | Power Doppler |
| 6 | 8 | Power Doppler |
| 7 | 12 | Transient Xenon |
| 8 | 8 | Axial Power Shaping |
| | 69 | |

## COMPARISON OF CALCULATED AND MEASURED CONTROL ROD GROUP REACTIVITY WORTH

Table 3.7-2

A. Moderator Temperature at 532$^{O}$F, APSRs at 37.5% Withdrawn

| Rod Group | Number Of Rods | Predicted Worth (%ΔK/K) | Measured Worth (%ΔK/K) | Percent Deviation (%) (1) |
|---|---|---|---|---|
| 5 | 12 | 1.24 | 1.42 | +14.5 |
| 6 | 8 | 1.00 | 1.07 | + 7.0 |
| 7 | 12 | 1.37 | 1.48 | + 8.0 |

B. Moderator Temperature at 579$^{O}$F, APSRs Centered

| Rod Group | Number Of Rods | Predicted Worth (%ΔK/K) |
|---|---|---|
| 5 | 12 | 1.27 |
| 6 | 8 | 1.02 |
| 7 | 12 | 1.44 |

(1) Percent deviation is calculated assuming predicted value is correct.



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## 3.8 Ejected Control Rod Worth

a. Purpose

Technical Specification 3.5.2 states that the maximum worth of a single inserted control rod at zero power condition of $532^{\circ}F$, 2155 psig shall not exceed 1.0%ΔK/K. A pseudo ejected control rod worth measurement was performed during the zero power test program to verify the safety analysis calculations relating to the hypothetical ejection of the most reactive control rod.

b. Test Method

Pseudo ejected control rod worths were measured for the rod in core location N-12 at zero power using two different techniques. The first technique was the boron-swap method during which the boron concentration of the reactor coolant system was slowly and continuously increased. The pseudo ejected rod was withdrawn in quick steps to compensate for the reactivity inserted by the boration and the reactivity change was measured by the reactimeter. The sum of the incremental reactivity changes gives the total worth of the ejected rod. In the second technique (rod swap method), critical equilibrium conditions were established with the pseudo ejected rod withdrawn to 100%. The ejected rod was then inserted into the core by swapping reactivity with group 5. The measured instantaneous worth of the dropped rod is taken as the worth of the ejected rod.

The worth of the three other rods symmetric to location N-12 (N-4, D-4, D-12) were measured using the rod swap method.

c. Test Results

From the measured worths of these four rods it was determined that a real core tilt existed and that the control rod in location D-4 had the highest ejected rod worth. In order to verify the worth of the rod in location D-4, the rod was deborated from 100% to 0% withdrawn with group 5 rods at 0% withdra[illegible] See Figure 3.8-1 for core locations.

The error adjusted maximum ejected rod worths for the four rods are listed below:

| Core Location | Ejected Rod Worth | |
|---|---|---|
| | Boron Swap (%ΔK/K) | Rod Swap (%ΔK/K) |
| N-12 | 0.83 | 0.76 |
| N-4 | | 0.77 |
| D-4 | 0.98 | 0.83 |
| D-12 | | 0.77 |

d. Conclusions

The measured ejected rod worths before error adjustments are in good agreement with the predicted value of 0.81%ΔK/K. Location D-4 has the highest error adjusted ejected rod worth at 0.98%ΔK/K which meets the Technical Specification requirement that the value not exceed 1.0%ΔK/K.

TROL ROD LOCATIONS FOR EJE[illegible] ROD
MEASUREMENT AT 532 F, 2155 PSI, 0 EFPD

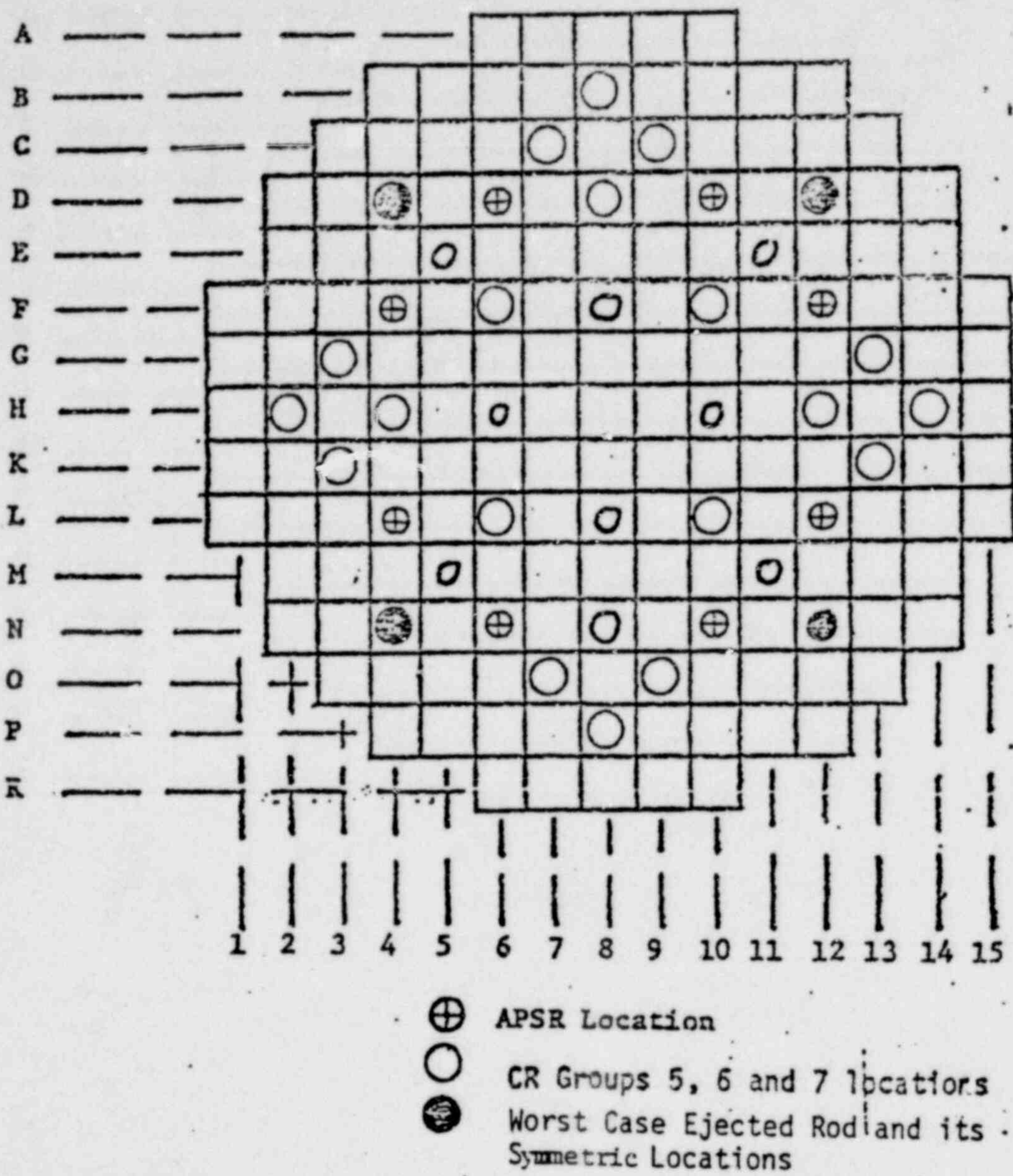


FIGURE 3.8-1



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3.9 Shutdown Margin

a. Purpose

Technical Specification 3.5.2 states that the available shutdown margin shall not be less than 1%ΔK/K with the most reactive control rod stuck out of the core. The purpose of the safety rod drop worth measurement at zero power, 532$^{o}$F was to verify that sufficient shutdown margin exists with the most reactive control rod stuck out of the core.

b. Test Method

Critical equilibrium conditions were established with groups 1-4 control rods at 100% WD and groups 5-7 at 0% WD. The reactimeter started logging data at 0.2 second intervals. The reactor was then tripped manually and the negative reactivity inserted into the core by control rod groups 1-4 was measured. It was assumed that the calculated worst case stuck rod had not dropped into the core and the available shutdown margin was calculated.

c. Test Results

The most reactive control rod at zero power, 532$^{o}$F was calculated to be in location L-14 (EOC-4 is most restrictive case) and those locations symmetric to it. Table 3.9-1 shows the results of the test. The boron concentration for the test was 880 ppmB. Correction factors were applied to the measured reactivity value from the reactimeter to correct for changes in the spatial flux distribution immediately after the rods drop. The minimum available shutdown margin with the most reactive control rod stuck out of the core was measured to be 1.07%ΔK/K. The measured value was determined to be low due to calculational errors in the reactimeter program which only occurred with extremely large flux changes like those experienced when tripping the safety rods.

d. Conclusions

Minimum shutdown margain verification was completed for the zero power condition at 532$^{o}$F. The calculated worst case stuck rod worth of 2.06%ΔK/K in location L-14 at EOC-4 was assumed for the stuck rod condition. The shutdown margin available under this condition was 1.07%ΔK/K (measured) which guarantees that the Technical Specification limit of 1.0%ΔK/K is satisfied.

Groups Tripped Safeties 1-4

Initial Position Gp 8 @ 37.5

| Complete Table Value (4 Sec. Interval) | Indicated Reactivity Reactivity (pcm) | |
|---|---|---|
| 1 | 4578 | Rods Start In |
| 2 | 2876 | |
| 3 | 3756 | |
| 4 | 3732 | |
| 5 | 3231 | |
| 6 | 3194 | |
| 7 | 2749 | |
| 8 | 2698 | |
| 9 | 2711 | Average 2608 pcm<br>gp 5<br>-142 worth<br>2466 |
| 10 | 2508 | |
| 11 | 2256 | |
| 12 | 2141 | |

Measured Value = Average x 1.41 = 3477 pcm

Shutdown Margin Verification

(Rod Worths of Gps. 1-4 by rod drop) x (0.9)= 3.13%ΔK/K

Worth of Worst Case Stuck Rod = 2.06%ΔK/K

Shutdown Margin = 1.07%ΔK/K

TABLE 3.9-1



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## 4.0 Core Performance - Measurements at Power

This section presents the results of the physics measurements that were conducted with the reactor at power. Testing was conducted at three major power plateaus, 40%, 75% and 100% of 2535 megawatts thermal core power, as determined from primary and secondary calorimetric measurements. Operation in the power range began on May 2, 1978. Power escalations occurred as the required testing at each plateau was successfully completed.

Periodic measurements and calibrations were performed on the plant nuclear instrumentation during the escalation to full power. The four power range detector channels were calibrated based upon primary and secondary plant heat balance measurements. Testing of the incore nuclear instrumentation was performed to ensure that all detectors were functioning properly and that the detector inputs were processed correctly by the plant computer. Core axial imbalance determined from the incore instrumentation system was used to calibrate the out of core detector imbalance indication.

The major physics measurements performed during power escalation consisted of determining the moderator and power Doppler coefficients of reactivity, analysis of the power distribution with the most reactive control rod dropped into the core at 40% FP, and obtaining detailed radial and axial core power distribution measurements for several core axial imbalances. Values of minimum DNBR and maximum linear heat rate were monitored throughout the test program to ensure that core thermal limits would not be exceeded.

## 4.1 Nuclear Instrumentation Calibration at Power

### a. Purpose

The purpose of Nuclear Instrumentation Calibration at Power was to calibrate the power range nuclear instrumentation indication to within ±2% FP of the reactor thermal power as determined by a heat balance and to within ±3.5% incore axial offset as determined by the incore monitoring system. Additional purposes during the power escalation program were as follows:

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

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

Two acceptance criteria are specified for nuclear instrumentation calibration at power as listed below.

(1) The power range nuclear instrumentation indicates power level within ±2% FP of the power level determined by heat balance and within ±3.5% of the incore axial offset as determined by the incore detectors.

(2) The high power level trip bistable is set to trip at the desired value, ±0.5% FP.

b. Test Method

As required during power escalation, the top and bottom linear amplifier gains were adjusted to maintain power range nuclear instrumentation channel power indication with ±2% of the power calculated by a heat balance. During top and bottom linear amplifier gain adjustment the ratio of their gains was maintained constant as long as the indicated axial offset was within ±3.5% of incore offset; if not, their gains were adjusted to correct imbalance and heat balance mismatch at the same time. While at 40% FP it was determined that the heat balance calculations were incorrect and that actual power was approximately 46%. The error was corrected and power was returned to 40% FP.

Data was also taken to verify overlap between the intermediate and power range channels. The required overlap was a minimum of one decade between these two nuclear instrumentation ranges.

When directed by the controlling procedure for physics testing (1550-01) the high flux trip bistable setpoint was adjusted. The major settings during power escalation are given below:

| Test Plateau %FP | Bistable Setpoint % FP |
|---|---|
| 40 | 50 |
| 75 | 85 |
| 100 | 105.5 |

c. Test Results

An analysis of test results indicated that changes in Reactor Coolant System boron and xenon buildup or burnout affected the power as observed by the nuclear instrumentation. This was as expected since the power range nuclear instrumentation measures reactor neutron leakage which is directly related to the above changes in system conditions. Each time that it was necessary to calibrate the power range nuclear instrumentation, the acceptance criteria of calibration to within ±2.0% FP of the heat balance power was met without any difficulty. Also, each time it was necessary to calibrate the power range nuclear instrumentation, the ±3.5% axial offset criteria as determined by the incore monitoring system was also met.

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The high flux trip bistable was adjusted to 50, 85 and 105.5% FP prior to excalation of power to 40, 75 and 100% FP, respectively.

d. Conclusions

The power range channels were calibrated to within two percent of heat balance power several times during the startup program. These calibrations were required due to power level, boron, and/or control rod configuration changes during the program. Acceptance criteria for nuclear instrumentation calibration at power were met in all instances.

The power range channels were calibrated to within two percent of heat balance power several times during the startup program. These calibrations were required due to power level, boron, and/or control rod configuration changes during the program. Acceptance criteria for nuclear instrumentation calibration at power were met in all instances.

## 4.2 Incore Detector Testing

a. Purpose

Self-powered-neutron-detectors (incore detector system) monitor the core power density within the core and their outputs are monitored and processed by the plant computer to provide accurate readings of relative neutron flux.

Tests conducted on the incore detector system were performed to:

(1) Verify that the output from each detector and its response to increasing reactor power was as expected.

(2) Verify that the background, length and depletion corrections applied by the plant computer are correct.

b. Incore Detector Tests

The response of the incore detectors versus power level was determined and a comparison of the symmetrical detector outputs made at steady state reactor power of 40 and 75% FP.

Using the corrected SPND maps, calculations were performed to determine the detector current to average detector current values per assembly for each incore detector versus axial positions. Any detector levels which were determined to have failed were deleted from scan.

At 75% FP, SP1301-5.3, Incore Neutron Detectors-Monthly check, was performed to calibrate the back-up recorders to the current P/$\overline{P}$ value.

c. Conclusions

Incore detector testing during power escalation demonstrated that detectors were functioning as expected, that symmetrical detector readings agreed within acceptable limits and that the computer applied correction factors are accurate. The backup incore recorders were calibrated at 75% FP and operational above 80% FP as required by the Technical Specifications.

## 4.3 Dropped Control Rod Power Verification

a. Purpose

The purpose of the Dropped Control Rod Power Distribution Verification Test at 40% FP was to verify satisfactory values of minimum DNBR and maximum linear heat rate with the control rod which will produce the most adverse thermal conditions inserted to 0% withdrawn.

b. Test Method

The results of core thermal calculations show that control rod 2 in group 6 location G-13 and those rods symmetric to it, would produce the most adverse thermal effects if it were dropped into the core during operation at power. The dropped rod test was conducted at 40% FP by inserting group 6 rod 2 to 0% withdrawn while compensating for reactivity changes with group 7 as necessary. As the rod was inserted it was verified that asymmetric alarm and fault indications occur at 7 and 9 inches respectively. With the rod at 0% withdrawn, core power distributions and thermal conditions were recorded. The rod was then returned to 100% withdrawn, again compensating for reactivity by inserting group 7.

c. Test Results

With group 6 rod 2 inserted in location G-13, the worst case maximum linear heat rate when extrapolated to 100% is 15.4 kw/ft which is well below the limit of 19.6 kw/ft. The extrapolated value of minimum DNBR to 100% FP is 2.81 which is well above the limit of 1.30.

d. Conclusions

The dropped control rod power verification test performed at 40% FP met all required acceptance criteria. The core power

distribution and thermal conditions that developed from the dropped rod showed adequate margins to minimum DNBR and maximum LHR limits.

## 4.4 Power Imbalance Detector Correlation Test

### a. Purpose

The Power Imbalance Detector Correlation Test has four objectives:

1. To determine the relationship between the indicated outcore power distribution and the actual incore power distribution.

2. To demonstrate axial Xenon control using the Axial Power Shaping Rods (APSR's).

3. To verify the adequacy and accuracy of backup imbalance calculations as done in AP 1203-7, "Hand Calculation for Quadrant Power Tilt and Core Power Imbalance."

4. To determine the core maximum linear heat rate and minimum DNBR at various power imbalances.

### b. Test Method

This test was conducted at 40% and 75% FP to determine the relationship between the core axial imbalance as indicated by the incore detectors and the out-of-core detectors. Based upon this correlation, it could be verified that the minimum DNBR and maximum linear heat rate limits would not be exceeded by operating within the flux/delta flux/flow envelope set in the Reactor Protection System. The test at 75% FP was performed after approximately 14 EFPD of operation at 90% FP.

The method employed to conduct the test is outlined below:

1. Steady State conditions were established at the required power level with core xenon concentrations at equilibrium.

2. The Incore Monitoring System was verified as operational and the backup recorders were checked as having been calibrated in accordance with SP 1301-5.3, "Incore Neutron Detectors-Monthly Check" (at 75% FP only).

3. The unit computer was verified as operational with applicable Nuclear Steam System (NSS) programs functioning properly.

4. Baseline data consisting of the following was collected per RP1550-08:

   a. PDO data segments 1-6.
   b. Computer Group 55, Imbalance/Tilt/Insertion
   c. Computer Group 34, 3-D Power Map
   d. Computer Group 48, SPND Map Corrected all Levels
   e. Computer Group 36, Fuel Assembly to Average Fuel Assembly Power Ratios
   f. Core burnup and RCS boron concentration
   g. Computer Group 20, Worst Case Thermal Conditions.

5. Once the baseline data was acquired, an imbalance was established using the group 8 control rods (APSRs). During group 8 movement, the integrated control system automatically compensated for the reactivity changes by repositioning group 6 to maintain constant power level.

6. The imbalance previously established was observed for a minimum period of twenty minutes prior to obtaining the following data:

   a. Specified Operator Trend Group
   b. Backup Incore Detector Recorder Data
   c. Computer Group 20, Worst Case Thermal Conditions
   d. PDO Segments 1-6 (at maximum positive and negative imbalances only)
   e. Computer Group 55, Imbalance/Tilt/Insertion
   f. Specified Operator Trend Group (Second Printout)

7. A new imbalance was then established and the same data was recorded once again; this procedure was repeated until the maximum positive and negative imbalances had been established and the required data recorded.

   As each imbalance condition was established, core power distribution and worst case thermal information was obtained from the plant computer to ensure safe conduct of the test. A plot was maintained of incore offset versus out-of-core offset.

c. Test Results

The relationship between the ICD and OCD offsets was determined at 40% FP and 75% FP by performing an imbalance scan with the APSR's. The average slope measured on the four out-of-core detectors was 1.59 at 40% and 1.35 at 75% full power. The lowest slope was 1.5 at 40% FP and 1.27 at 75% FP for NI-7. The scaled difference amplifier gain was 6.5 at 40% FP and 5.416 at 75% FP.

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A comparison of the incore detector (ICD) offset versus the out-of-core (OCD) detector offset obtained for each NI channel is shown in Table 4.4-1.

Core power distribution measurements were taken in conjunction with the most positive and most negative imbalances at 40% FP and the values of minimum DNBR and worst case MLHR and compared to the acceptance criteria.

The worst case values of minimum DNBR and maximum linear heat rate determined at 40% FP are listed in Table 4.4-2.

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TABLE 4.4-2

| Nominal Power (%) | Measured DNBR | Worst Case MLHR (kw/ft) | EXTRAPOLATIONS Power (%) | DNBR | MLHR |
|---|---|---|---|---|---|
| 40 | 8.26 | 5.52 | 85 | 2.59 | 12.25 |

The worst case DNBR ratio was greater than the minimum limit of 1.3 and the maximum value of linear heat rate was less than the fuel melt limit of 19.6 kw/ft after extrapolation to 85% FP. These results show that Technical Specification limits have been met and that adequate protection is provided by the Reactor Protection System trip setpoints for the allowed axial imbalances during power operation.

Backup imbalance calculations using AP 1203/7 agreed with computer calculated imbalances. Table 4.4-3 lists the computer calculated imbalances as well as imbalances obtained using the incore detector backup recorders.

TABLE 4.4-3

| Nominal Power (%) | Computer Calculated Imbalance (%) | Backup Recorder Imbalance (%) |
|---|---|---|
| 75 | + 4.42 | + 4.29 |
| 75 | + 2.34 | + 2.22 |
| 75 | -12.13 | - 8.97 |
| 75 | - 9.68 | - 7.20 |
| 75 | - 4.21 | - 2.76 |
| 75 | - 0.23 | + 0.49 |

d. Conclusions

Backup imbalance calculations performed in accordance with AP 1203/7 provide an acceptable alternate method to computer calculated values of imbalance. Using a difference amplifier K factor of 4.904 will provide a slope greater than 1.15 when OCD offset is plotted versus ICD offset.

Minimum DNBR and Maximum Linear Heat Rate parameters were well within Technical Specifications limitations.

TABLE 4.4-1

| Nominal Power % | % ICD Offset | % OCD OFFSET NI-5 | NI-6 | NI-7 | NI-8 |
|---|---|---|---|---|---|
| 40 | +14.7 | 22.5 | 25.3 | 23.0 | 21.2 |
| 40 | +15.1 | 23.7 | 26.1 | 23.6 | 22.3 |
| 40 | + 7.8 | 12.2 | 13.9 | 13.7 | 11.2 |
| 40 | -32.4 | -56.1 | -54.2 | -48.7 | -52.3 |
| 40 | -18.9 | -31.25 | -30.0 | -26.0 | -29.4 |
| 40 | -12.7 | -19.8 | -18.5 | -15.7 | -19.0 |
| 40 | - 0.12 | + 0.26 | + 1.04 | + 2.1 | - 0.78 |
| 75 | + 6.21 | 8.04 | 8.52 | 7.17 | 7.61 |
| 75 | + 3.33 | 4.24 | 4.40 | 3.68 | 3.89 |
| 75 | +17.07 | -23.90 | -24.13 | -21.87 | -22.57 |
| 75 | -13.67 | -19.06 | -19.37 | -17.26 | -18.04 |
| 75 | - 5.95 | - 8.42 | - 8.51 | - 7.78 | - 7.99 |
| 75 | - 0.33 | - 0.70 | - 0.70 | - 0.92 | - 0.81 |

$$\text{ICD OFFSET} = \frac{\text{POWUP - POWLW}}{\text{powup + POWLW}} \times 100\%$$

$$\text{OCD OFFSET} = \frac{\text{CHANNEL IMBALANCE}}{\text{CHANNEL POWER}} \times 100\%$$

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## 4.5 Core Power Distribution Verification

### a. Purpose

To measure the core power distributions at 40, 75 and 100 percent full power to verify that the core axial imbalance, quadrant power tilt, maximum linear heat rate and minimum DNBR do not exceed their specified limits. Also, to compare the measured and calculated power distributions.

### b. Test Method

Core power distribution measurements were performed at each of the major power plateaus of the test program (40%, 75% and 100% full power) under steady state conditions for specified control rod configurations. To provide the best comparison between measured and predicted results, three-dimensional equilibrium xenon conditions were established for all the measurements. Data collected for the measurements consisted of detailed power distribution information at 364 core locations from the incore detector system and the worst case core thermal conditions were calculated using this data. The measured data was compared to calculated results.

### c. Test Results

A summary of the three cases studied in this report is given in Table 4.5-1 which gives the core power level core burnup, control rod pattern, boron concentration, xenon conditions, axial imbalance, maximum quadrant tilt, minimum DNBR, maximum LHR and power peaking data for each measurement. The highest Worst Case MLHR was 12.75 at 100% FP which is well below the limit of 19.6 kw/ft. The lowest minimum DNBR value was 3.52 at 100% FP which is well above the limit of 1.30.

The quadrant power tilt and axial imbalance values measured were all within the allowable limits. Table 4.5-2 gives a comparison between the maximum calculated and predicted radial and total peaks for an 1/8 core power distribution.

The results of the core power distribution comparisons at 40% and 75% full power indicated radial and total peaks higher than predicted and outside the acceptable range. At 75% full power the radial peak was 9.6% higher than predicted and the total peak was 11.3% higher than predicted. The acceptance criteria were 5% and 7.5% respectively. Reactor power was limited to 91% full power until Babcock and Wilcox analyzed the core power distribution data and a Technical Specification amendment was issued based on new error bands of 11% and 13% for radial and total peaks respectively. The core power

distribution data for 100% full power was then obtained at approximately 25 EFPD and the radial and total peaks were measured to be 8.9% and 6.9%, respectively higher than the predicted values. These results at 100% FP were therefore within the new acceptance criteria.

d. Conclusions

Core power distribution measurements were conducted at 40%, 75% and 100% full power. Calculated results were compared to measured radial and total peaks. The measured peaks at 40% and 75% full power differed from predicted peaks by amounts outside the acceptable deviation. A reanalysis of core power distribution data was performed to amend Technical Specifications to allow 11% and 13% deviations in radial and total peaks, respectively. The radial and total peaks measured for 100% full power at approximately 25 EFPD were higher than the predicted values but the difference was within the revised allowable deviations.

The measured values of DNBR and MLHR were all within the allowable limits. All quadrant power tilts and axial core imbalances measured during the power distribution test were within the Technical Specifications and normal operational limits.

## 4.6 Reactivity Coefficients at Power

a. Purpose

The purpose of this test was to measure the temperature and power doppler coefficients of reactivity at power. This information is then used to assure that Tech. Spec. 3.1.7.1, which states that the moderator temperature coefficient shall not be positive at power levels above 95% of rated power, is satisfied.

b. Test Method

For measuring the temperature coefficient of reactivity, the average RC temperature was decreased and then increased by about 5 Degree F. The reactivity associated with each temperature change was obtained from the change in controlling rod group position, and the values for the coefficient were calculated.

For measuring the power doppler coefficient of reactivity, reactor power was decreased and then increased by about 5 percent FP. The reactivity change was obtained from the change in controlling rod group position, and the values for the coefficient were calculated.

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In conjunction with both reactivity coefficient measurements, controlling rod group worth measurements using the fast insert/withdrawal method were performed.

c. Test Result

At 100% FP, temperature and power doppler coefficient measurements were performed. The temperature coefficient measured at 100% FP was-9.03 pcm/$^{o}$F which verified that the moderator temperature coefficient is negative above 95% FP.

The measured power doppler coefficient at 100% FP was -10.63 pcm/%FP which meets the acceptance criteria stating that it shall be more negative than -5.5 pcm/%FP. The measured value agrees well with the calculated value of -13.6 pcm/%FP.

d. Conclusions

The measured results indicate that the moderator temperature coefficient will be negative during power operation above 95% FP. The results of the temperature and power doppler coefficients of reactivity are in agreement with the expected results.

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Measured Core Power Distributions and Core Thermal Conditions
For Various Control Rod Patterns and Core Power Levels
Of 40, 75 and 100% FP

| | 40% | 75% | 100% |
|---|---|---|---|
| Date | 05/04/78 | 05/06/78 | 05/31/78 |
| Time | 1030 | 2110 | 1000 |
| Power Level, % FP | 40.0 | 74.6 | 100 |
| Xenon Equilibrium | 3-D | 3-D | 3-D |
| Rod Positions, % wd | | | |
| 1-6 | 100 | 100 | 100 |
| 7 | 85.8 | 85 | 93 |
| 8 | 33 | 25 | 29 |
| Core Burnup, EFPD | 0.77 | 2.37 | 25.05 |
| Boron Concentration, ppmb | 951 | 863 | 734 |
| Axial Imbalance, % FP | -0.75 | -1.75 | -0.47 |
| Maximum Quadrant Tilt, % | +2.80 | +1.70 | +0.92 |
| Minimum DNBR | 9.69 | 4.80 | 3.52 |
| Worst Case MLHR, kw/ft | 5.44 | 10.13 | 12.75 |
| Maximum Peaks (Measured) | | | |
| Radial | 1.385 | 1.377 | 1.347 |
| Total | 1.725 | 1.719 | 1.615 |
| Maximum Peaks (Predicted) | | | |
| Radial | 1.273 | 1.245 | 1.227 |
| Total | 1.552 | 1.525 | 1.503 |

TABLE 4.5-1

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TABLE 4.5-2

| Power Level | Maximum Measured Radial Peaking Factor | Maximum Predicted Radial Peaking Factor | Maximum Measured Total Peaking Factor | Maximum Predicted Total Peaking Factor |
|---|---|---|---|---|
| 40% | 1.385 | 1.273 | 1.725 | 1.552 |
| 75% | 1.377 | 1.245 | 1.719 | 1.525 |
| 100% | 1.347 | 1.227 | 1.615 | 1.503 |



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