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February 4, 1992
C311-92-2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Cycle 9 Startup Report

Enclosed is the GPU Nuclear Startup Report for TMI-1 Cycle 9 operation. Initial criticality for Cycle 9 was achieved at 1222 hours on November 14, 1991. Testing addressed by this report was completed and approved as of November 20, 1991. This report is being submitted in accordance with TMI-1 Technical Specification 6.9.1.A. No NRC response to this letter is necessary or requested.

Sincerely,

T G Broughton

T. G. Broughton
Vice President and Director, TMI-1

MRK

Enclosure

cc: Region I Administrator
TMI-1 Senior Project Manager
TMI Senior Resident Inspector

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GPU Nuclear

TMI-1

CYCLE 9

STARTUP REPORT

TMI-1 Nuclear Engineering

December, 1991

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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 November 14, 1991 and ended on November 15, 1991. This section presents a summary of the zero power measurements. In all cases, the applicable test and Technical Specifications limits were met. A summary of zero power physics test results appear as Table 1-1.

a. Initial Criticality

Initial criticality was achieved at 1222 on November 14, 1991. Reactor conditions were 532°F and 2155 psig. Control rod groups 1 through 6 were withdrawn to 100%; group 7 was positioned at 91% withdrawn; group 8 was positioned at 25% withdrawn. Criticality was achieved by deborating the Reactor Coolant from 2428 ppm to 2162 ppm. Initial criticality was achieved in an orderly manner and the acceptance criteria of 2217 ± 100 PPM was met.

b. Nuclear Instrumentation Overlap

At least one decade overlap was measured between the source and intermediate range detectors as required by Technical Specifications.

c. Reactimeter Checkout

An on-line functional check of the 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 2157 ppmB was within the acceptance criteria of 2221 ± 100 ppmB.

e. Temperature Coefficient Measurements

The measured temperature coefficient of reactivity at 532°F, zero power was within the acceptance criteria limit.

f. Control Rod Group 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 method were in good agreement with predicted values. The maximum deviation between measured and predicted worths was -2.4% which was for CRG-7 worth.

g. Differential Boron Worth

The measured differential boron worth at 532°F was 9.0% more than the predicted value. This is within the bounds of the FSAR and B&W supplied limits of ±15%.

TABLE 1-1

Summary of Zero Power Physics Test Results

Cycle 9

Parameter	Acceptance Criteria	Measured Value	Deviation
Critical Boron	2217 ± 100 ppm	2162 ppm	-55 ppm
NI Overlap	>1 decade	>1.57 decade	---
Sensible Heat	N/A	8.1×10^{-8} amps	---
All Rods Out Boron Concentration	2221 ± 100 ppm	2157 ppm	-64 ppm
Temperature Coefficient (2149 ppm)	2.17 pcm/°F ± 4 pcm/°F	3.11 pcm/°F	+0.94 pcm/°F
Moderator Coefficient	<9.0 pcm/°F	4.82 pcm/°F	---
Integral Rod Worths (532°F) GP5-7	2791 pcm ± 10%	2839.8 pcm	-1.72%
Group 7	868 pcm ± 15%	889.3 pcm	-2.4%
Group 6	808 pcm ± 15%	816.5 pcm	-1.04%
Group 5	1115 pcm ± 15%	1134 pcm	-1.67%
Diff Boron Worth (1951 ppm)	7.098 pcm/ppm ± 15%	7.804 pcm/ppm	-9.0%

2.0 CORE PERFORMANCE - MEASUREMENTS AT POWER - SUMMARY

This section summarizes the physics tests conducted with the reactor at power. Testing was performed at power plateaus of approximately 10, 30, 75, and 100% core thermal power. Operation in the power range began on November 15, 1991.

Four Westinghouse lead test fuel assembles (LTA) were monitored during the startup program to ensure that they were not the limiting (hottest) assemblies in the core with respect to radial power distribution power peaking. Analyses predict that the LTA will maintain at least a 3% hot pin relative power peaking margin.

a. Nuclear Instrumentation Calibration at Power

The power range channels were calibrated as required during the startup program based on power as determined by primary and secondary plant heat balance. These calibrations were required due to power level, boron and/or control rod configuration changes during testing.

b. Incore Detector Testing

Tests conducted on the incore detector system demonstrated that all detectors were functioning acceptably. Symmetrical detector readings agreed within acceptable limits and the plant computer applied the correct background, length and depletion correction factors. The backup incore recorders were operational above 80% FP as required by Technical Specifications.

c. Power Imbalance Detector Correlation Test

The results of the Axial Power Shaping Rod (APSR) movements performed at 75% FP show that an acceptable incore versus out-of-core offset slope of >0.96 is obtained by using a gain factor of 3.684 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 alternative to computer calculated values.

d. Core Power Distribution Verification

Core power distribution measurements were conducted at approximately 75% and 100% full power under steady state equilibrium xenon conditions for specified control rod configurations. The maximum measured and maximum predicted radial and total peaking factors are all in good agreement. The largest difference between the maximum measured and maximum predicted value was +2.8% for radial peaking at 99.98% FP. This met acceptance criteria of <5.0%.

The results of the core power distribution measurements are given in Table 4.4-1. All quadrant power tilts and axial core imbalances measured during the power distribution tests were within the Technical Specification and normal operational limits.

e. Reactivity Coefficients at Power

The isothermal temperature coefficient measured at approximately 99% FP was -5.91 pcm/°F. The measured power doppler coefficient at approximately 99% FP was -8.78 pcm/% FP. All Technical Specification and Safety Analysis requirements were met.

3.0 CORE PERFORMANCE - MEASUREMENTS AT ZERO POWER

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

3.1 Initial Criticality

Initial criticality for Cycle 9 was achieved at 1222 on November 14, 1991. Reactor conditions were 532°F and 2155 psig. Control rod groups 1 through 4 were withdrawn during the heatup to 532°F. The initial reactor coolant system (RCS) boron concentration was 2428 ppm.

The approach to criticality began by withdrawing control rod group 8 to 25% withdrawn, control rod groups 5 and 6 to 100% withdrawn, and positioning group 7 at 85% withdrawn. Criticality was subsequently achieved by deborating the reactor coolant system to a boron concentration of 2162 ppm. The procedure used in the approach to criticality is outlined below in two basic steps:

Step 1	Control Rod Withdrawal	
	Group 8	25% withdrawn
	Group 5	100% withdrawn
	Group 6	100% withdrawn
	Group 7	85% withdrawn

Step 2 Deborate using a feed and bleed flow rate of 50 gpm until the inverse count rate is at approximately 0.3. At this point, stop deboration and increase letdown flow to maximum (120 gpm). This enhances 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.

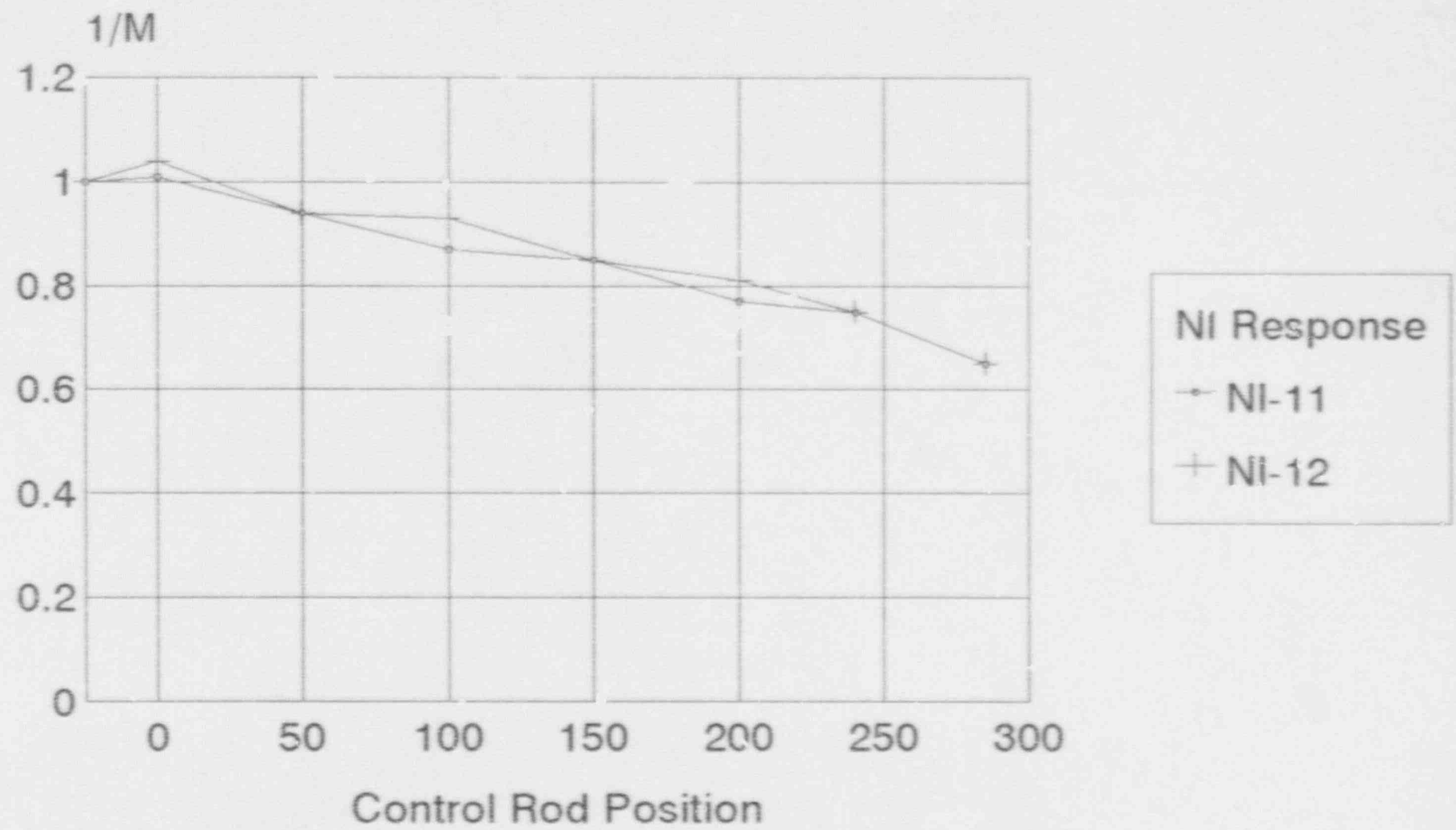
Throughout the approach to criticality, plots of inverse multiplication were maintained by two independent persons. Count rates were obtained from each source range neutron detector channel. One person used NI-1 and 11, the other used NI-2 and 12. Four plots of inverse count rate (ICR) versus control rod position were maintained during control rod withdrawal. Four plots of ICR versus RCS boron concentration and four plots of ICR versus gallons of demineralized water added were maintained during the dilution sequence.

The inverse count rate plots maintained during the approach to criticality are presented in Figures 3.1-1 through 3.1-3. As can be seen 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. Figure 3.1-2 is the ICR plots versus RCS boron concentration and Figure 3.1-3 is the ICR plots versus gallons of demineralized water added to the RCS.

In summary, initial criticality was achieved in an orderly manner. The measured critical boron concentration was within the acceptance criteria of 2217 ± 100 PPM.

Figure 3.1-1

1/M vs. CRG Position



Zero corresponds to 25% WD on Group 8, 0% on Group 5

Figure 3.1-2

1/M vs RCS Boron Concentration

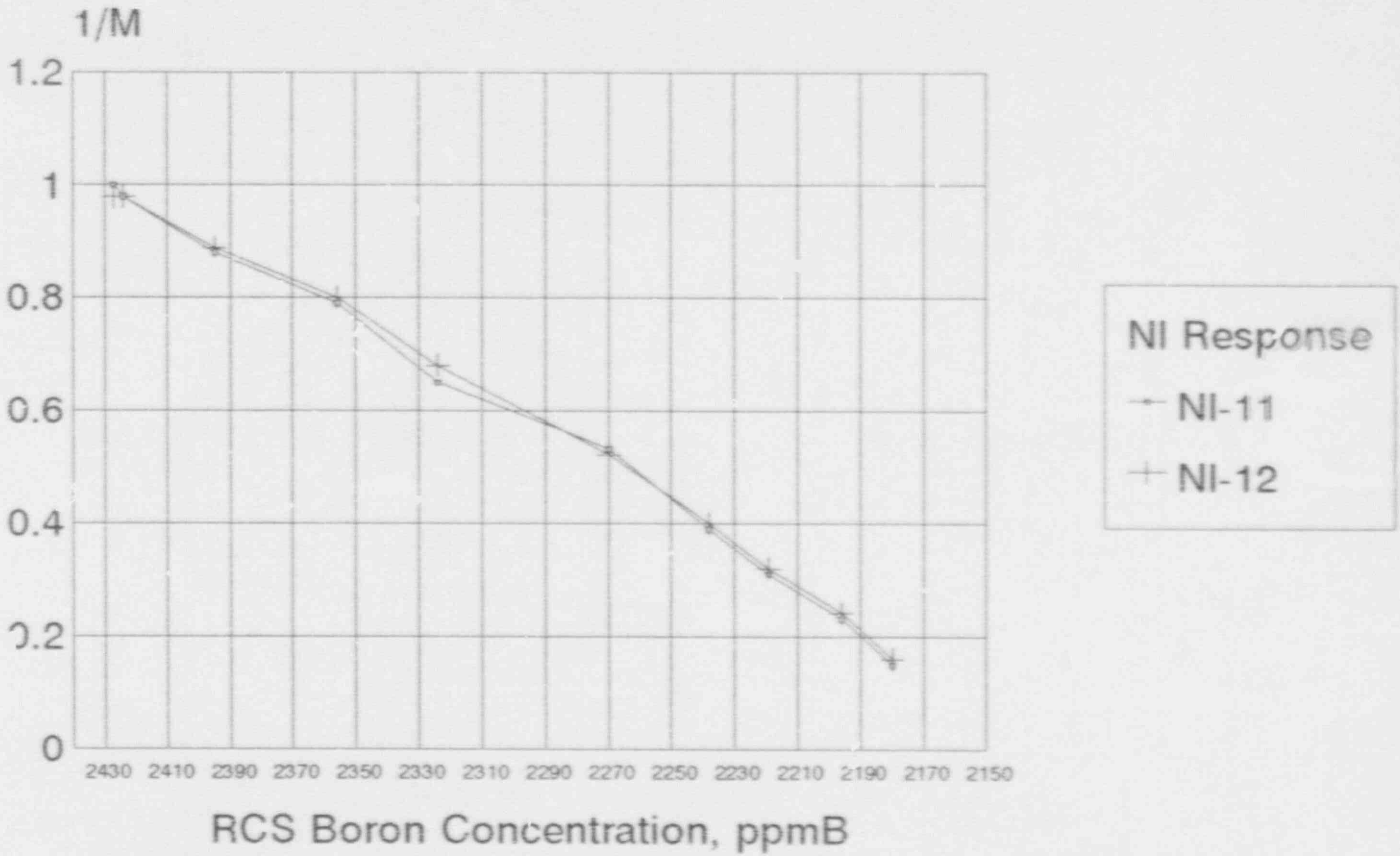
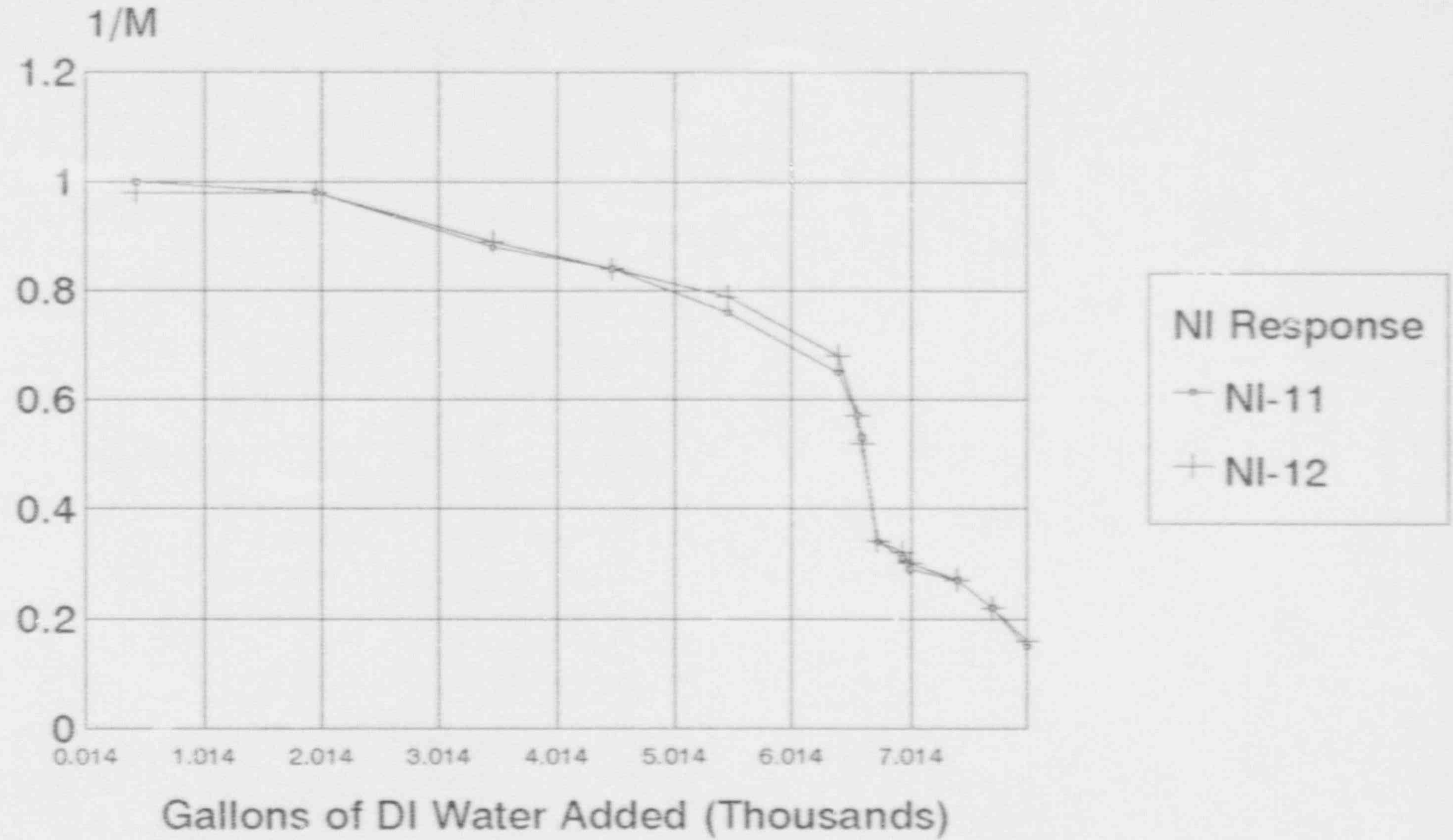


Figure 3.1-3

1/M vs Gallons of Water Added



Steep drop at end due to waiting and mixing

3.2 Nuclear Instrumentation Overlap

a. Purpose

Technical Specification 3.5.1.5 states that prior to operation in the intermediate nuclear instrumentation (NI) range, at least one decade of overlap between the source range NI's and the intermediate range NI's must be observed.

b. Test Method

To satisfy the above overlap requirements, 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 until the maximum source range value was reached.

c. Test Results

The results of the initial NI overlap data at 532°F and 2155 psig have shown a >1.57 decade 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 a one decade overlap between the source and intermediate ranges, thus satisfying T.S. 3.5.1.5.

3.3 Reactimeter Checkout

a. Purpose

Reactivity calculations during the Cycle 9 test program were performed using the reactimeter. After initial criticality and prior to the first physics measurement, an online 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 was established, intermediate range channel NI-3 was connected to the reactimeter and the reactivity calculations were started. After steady state conditions 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 was determined from the doubling time reactivity curves. The measurements were taken at +64.6 and -38.2 pcm. The reactivities determined from doubling time measurements were compared with the reactivity calculated by the reactimeter.

c. Test Results

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.

3.4 All Rods Out Critical Boron Concentration

a. Purpose

The all rods out critical boron concentration measurement was performed to obtain an accurate value for the excess reactivity loaded in the TMI Unit 1 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 all rods out condition and steady state conditions were established.

c. Test Results

The measured boron concentration with group 7 positioned at 100%WD was 2157 ppm.

d. Conclusions

The above results show that the measured boron concentration of 2157 ppm is within the acceptance criteria of 2221 ± 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 while greater than 95% FP. The moderator temperature coefficient cannot be measured directly, but it can be derived from the isothermal temperature coefficient and a known fuel temperature (Doppler) coefficient.

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 10^{-8} amps on the intermediate range. 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 recorders were connected with the reactivity value and the RCS average temperature displayed on a two channel strip chart recorder.

Once steady state conditions were established, a heatup rate was started by closing the turbine bypass valves. After the core average temperature increased by about 5°F core temperature and flux were stabilized and the process was reversed by decreasing the core average temperature by about 10°F. After core temperature and flux were stabilized, core temperature was returned to its initial value. Calculation of the temperature coefficient from the measured data was performed by dividing the change in core reactivity by the corresponding change in RCS temperature.

c. Test Results

The results of the isothermal temperature coefficient measurements are provided below. The predicted values are included for comparison.

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

RCS BORON (PPM)	MEASURED ITC (PCM/DEG F)	PREDICTED ITC (PCM/DEG F)	MEASURED MTC (PCM/DEG F)	REQUIRED MTC (PCM/DEG F)
2149	3.11	2.17	4.82	<9.0

d. Conclusions

The measured values of the temperature coefficient of reactivity at 532°F, zero reactor power are within the acceptance criteria of ± 4.0 pcm/°F of the predicted value. An extrapolation of the moderator coefficient to 100%FP indicated that it was well within the limits of Technical Specifications 3.1.7.2.

3.6 Control Rod Group Worth Measurements

a. Purpose

This section provides comparison between the calculated and measured results for the control rod group worths. The location and function of each control rod group is shown in Figure 3.6-1. The grouping of the control rods shown in Figure 3.6-1 will be used throughout Cycle 9. 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. Test Method

Control rod group reactivity worth measurements were performed at zero power, 532°F using the boron/rod swap method. Both the differential and integral reactivity worths of control rod groups 5, 6, and 7 were determined.

The boron/rod swap method consists of establishing a deboration rate in the reactor coolant system, then compensating for the reactivity changes by inserting the control rod groups in incremental steps.

The reactivity changes that occurred during the measurements were calculated by the reactimeter. Differential rod worths were obtained from the measured 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.

c. Test Results

Control rod group reactivity worths were measured at 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 3.6-2 through 3.6-4.

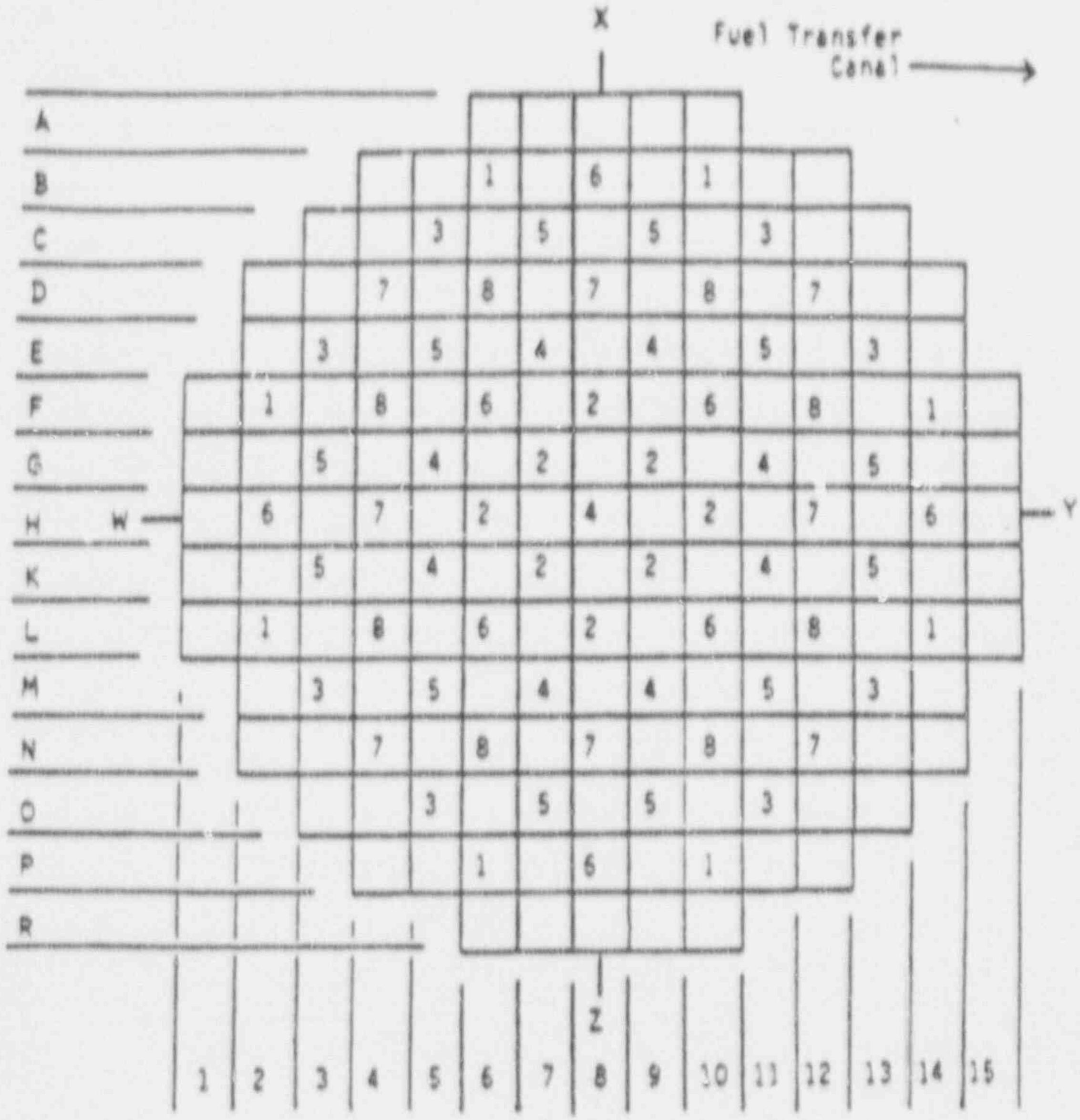
These curves were obtained by integrating the measured differential worth curves.

Table 3.6-1 provides a comparison between the predicted and measured results for the rod worth measurements. The results show good agreement between the measured and predicted rod group worths. The maximum deviation between measured and predicted worths for a group was -2.4%.

d. Conclusions

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

Figure 3.6-1 Control Rod Locations and Group Designations for TMI-1 Cycle 9

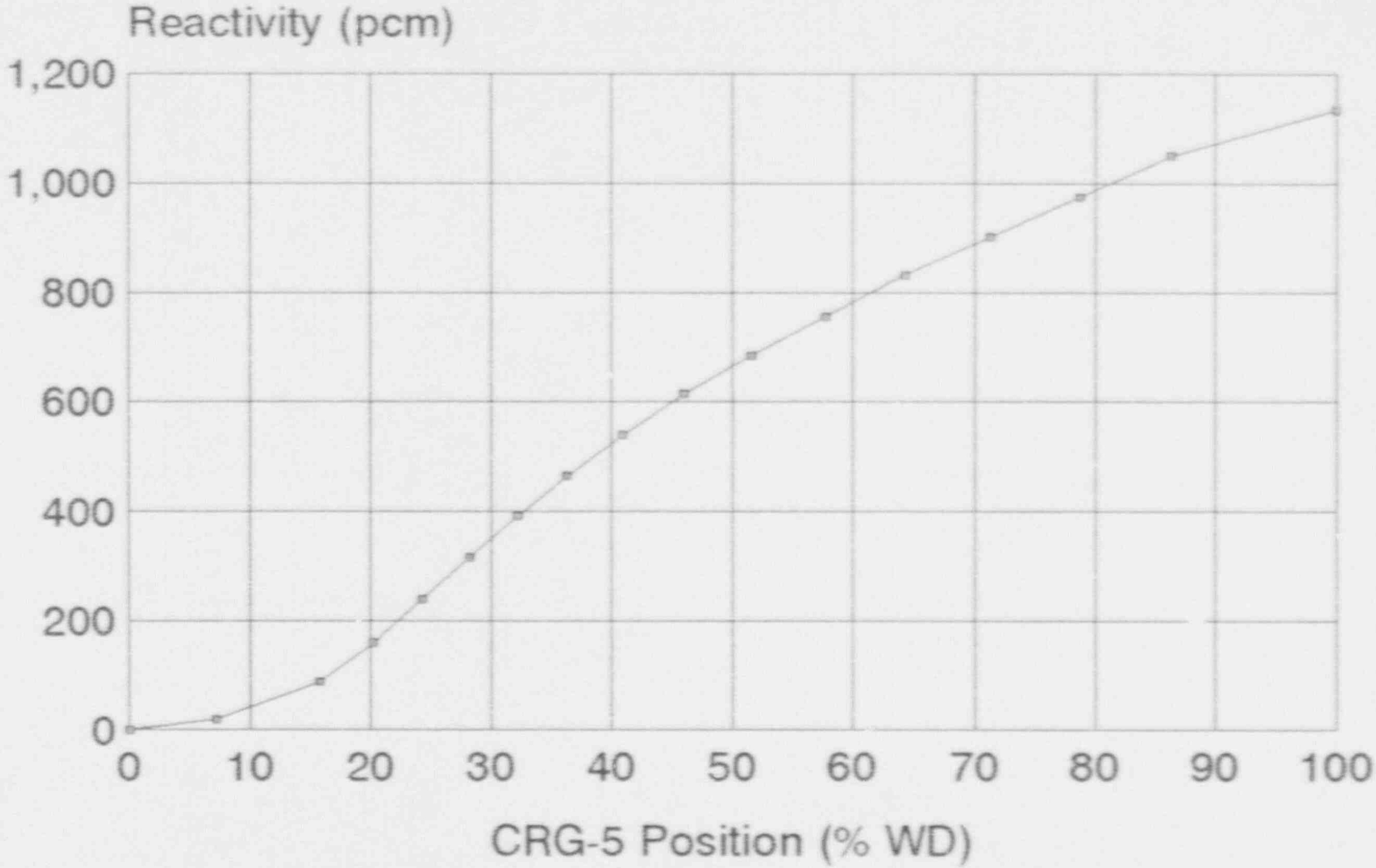


$\boxed{X}$ Group Number

Group	No. of Rods	Function
1	8	Safety
2	8	Safety
3	8	Safety
4	9	Safety
5	12	Control
6	8	Control
7	8	Control
8	8	APSRs

Figure 3.6-2

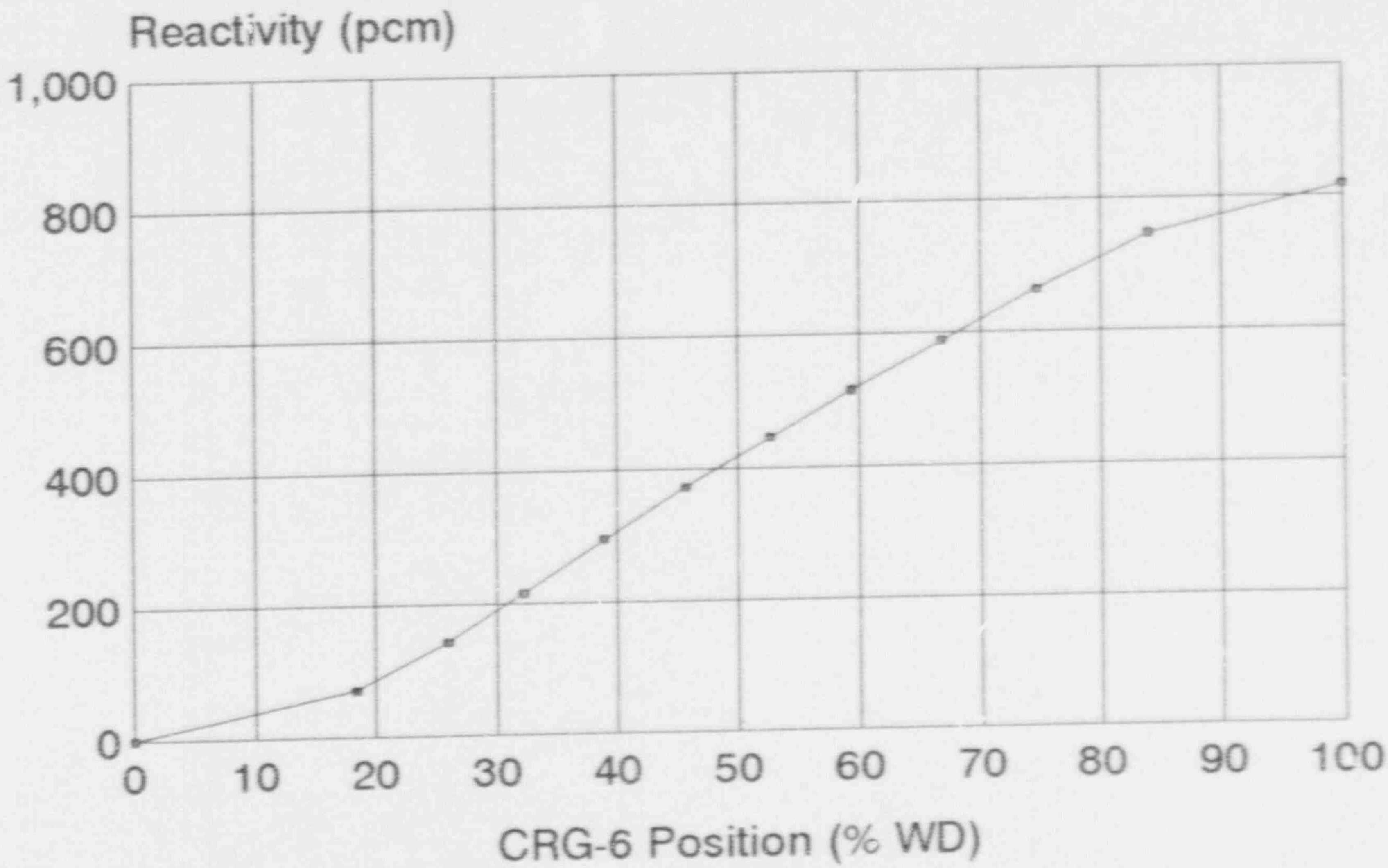
Integral Worth for CRG-5



Total Worth = 1134.0 pcm

Figure 3.6-3

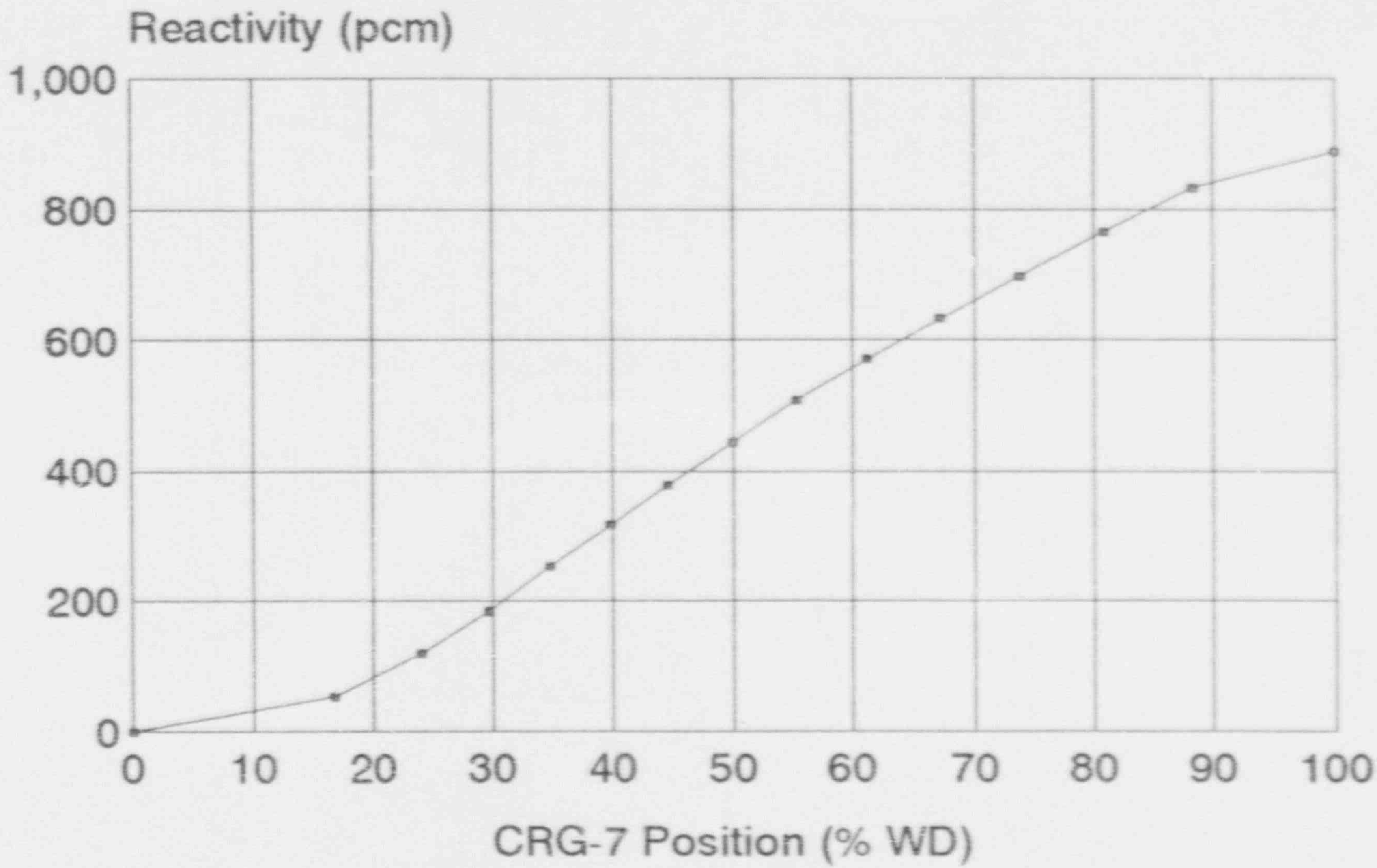
Integral Worth for CRG-6



Total Worth = 816.5 pcm

Figure 3.6-4

Integral Worth for CRG-7



Total Worth = 889.3 pcm

TABLE 3.6-1

COMPARISON OF PREDICTED VS MEASURED ROD WORTHS

CRG. NO.	MEASURED WORTH (PCM)	PREDICTED WORTH (PCM)	PERCENT DIFFERENCE %
5	1134	1115 ± 15%	-1.67
6	816.5	808 ± 15%	-1.04
7	889.3	868 ± 15%	-2.4
5-7	2839.8	2791 ± 10%	-1.72

3.7 Differential Boron 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 was measured during the zero power test.

b. Test Method

Measurements of the differential boron worth at 532°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 deboration rate was established and the control rods were 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°F. The measured boron worth was 7.804 pcm/ppmB at an average boron concentration of 1951 ppmB. The predicted value was 7.098 pcm/ppmB ± 15%.

d. Conclusions

The measured results for the soluble poison differential worth at 532°F was within 15% of the predicted differential worth.

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 power plateaus of approximately 10%, 30%, 75%, and 100% of 2568 megawatts core thermal power, as determined from primary and secondary heat balance measurements. Operation in the power range began on November 15, 1991.

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 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 the Nuclear Instrumentation Calibration at Power was to calibrate the power range nuclear instrumentation indication to be no less than 2% FP of the reactor thermal power as determined by a heat balance and to within ± 2.5% incore axial offset as determined by the incore monitoring system.

b. Test Method

As required during power escalation, the top and bottom linear amplifier gains were adjusted to maintain power range nuclear instrumentation indication to be not less than 2% of the power calculated by a heat balance.

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

Test Plateau % FP	Bistable Setpoint % FP
30	50
75	85
100	105.1

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 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 be no less than 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 ± 2.5% axial offset criteria as determined by the incore monitoring system was also met.

The high flux trip bistable was adjusted to 50, 85 and 105.1% FP prior to escalation of power to 30, 75 and 100% FP, respectively.

d. Conclusions

The power range channels were calibrated based on 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.

(3) To measure the degree of azimuthal symmetry of the neutron flux.

b. Test Method

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 approximately 10, 30, 75, and 100%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.

At 75% FP, SP-1301-5.3, Incore Neutron Detectors-Monthly Check, was performed to calibrate the backup recorder detectors to their incore depletion value.

c. Conclusions

Incore detector testing during power escalation demonstrated that all detectors were functioning as expected. Symmetrical detector readings agreed within acceptable limits and 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 Power Imbalance Detector Correlation Test

a. Purpose

The Power Imbalance Detector Correlation Test has four objectives:

1. To determine the relationship between the core power distrubtion as measured by the out-of-core detectors and the incore instruments.

2. To demonstrate axial power shaping 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 about 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.

CRG-8 was moved to establish the various imbalances. The integrated control system automatically compensated for reactivity changes by repositioning CRG-7 to maintain a constant power level.

c. Test Results

The relationship between the ICD and OCD offset was determined at about 75% FP by changing axial imbalance with the APSR's. The average slope measured on the four out-of-core detectors was 1.066. The lowest slope was 1.008 for NI-7. The scaled difference amplifier gain was left at 3.684.

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 4.3-1.

Core power distribution measurements were taken at the most positive and negative imbalances at 75% FP. The values of minimum DNBR and worst case MLHR were compared to the acceptance criteria.

The worst case values of minimum DNBR and maximum linear heat rate determined at 75% FP are listed in Table 4.3-2.

The worst case DNBR ratio was greater than the minimum limit and the maximum value of linear heat rate was less than the fuel melt limit of 20.5 kw/ft after extrapolation to 105.1 FP. These results show that Technical Specification limits have been met.

Backup offset calculations using AP 1203-7 agree with the computer calculated offset. Table 4.3-3 lists the computer calculated offset as well as offsets obtained using the incore detector backup recorders.

d. Conclusions

Backup imbalance calculations performed in accordance with AP 1203-7 provide an acceptable alternate method to computer calculated values of imbalance. A difference amplifier K factor of 3.684 will provide a slope greater than or equal to 0.96 when OCD offset is plotted versus ICD offset.

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

TABLE 4.3-1

INCORE OFFSET VS OUT-OF-CORE OFFSET

INCORE OFFSET (%)	OUT-OF-CORE OFFSET (%)			
	NI-5	NI-6	NI-7	NI-8
3.03	3.79	3.98	3.51	3.74
9.78	10.11	10.32	9.08	9.64
9.62	10.16	[illegible]	9.19	9.59
6.19	6.98	[illegible]	6.40	6.74
-2.03	-2.11	-1.93	1.96	-1.75
-8.06	-9.21	-9.16	-8.45	-8.41
-11.87	-13.48	-13.48	-12.32	-12.41
-13.46	-15.18	-15.22	-13.91	-14.04

TABLE 4.3-2

WORST CASE DNBR AND LHR

IMBALANCE %	MINIMUM DNBR	EXTRAPOLATE MDNBR	WORST CASE LHR (KW/FT)	EXTRAP. MAX. LHR (KW/FT)
7.33	3.961	2.55	9.67	12.71
-10.05	4.102	2.76	9.80	13.23

TABLE 4.3-3

FULL INCORE OFFSET VS BACKUP RECORDER OFFSET

FULL INCORE OFFSET (%)	BACKUP RECORDER OFFSET (%)
3.037	-0.065
9.786	6.286
-13.46	-14.06

4.4 Core Power Distribution Verification

a. Purpose

To measure the core power distributions at approximately 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 predicted power distributions.

b. Test Method

Core power distribution measurements were perfc ed at approximately 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. Data collected for the measurements consisted of power distribution information at 364 core locations from the incore detector system. The worst case core thermal conditions were calculated using this data. The measured data was compared with calculated results.

c. Test Results

A summary of the cases studied in this report is given in Table 4.4-1 which gives the core power level, control rod pattern, cycle burnup, boron concentration, axial imbalance, maximum quadrant tilt, minimum DNBR, maximum LHR and power peaking data for each measurement. The highest Worst Case MLHR was 12.01 kw/ft at 100% FP which is well below the limit of 20.5 kw/ft. The lowest minimum DNBR value was 3.081 at 100% FP which is well above the limit.

The quadrant power tilt and axial imbalance values measured were all within the allowable limits. Table 4.4-1 also gives a comparison between the maximum calculated and predicted radial and total peaks for an eighth core power distribution.

d. Conclusions

Core power distribution measurements were conducted at approximately 75% and 100% full power. Comparison of measured and predicted results show good agreement. The largest difference between the maximum measured and maximum predicted peak value was 2.8% for radial peaking at 100% FP. This met the acceptance criteria of <5.0%.

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.

TABLE 4.4-1

CORE POWER DISTRIBUTION RESULTS

POWER PLATEAU		75%FP	100%FP
DATE		11-16-91	11-18-91
Actual Power	(%FP)	74.71	99.98
CRG 1-6	(%WD)	100	100
CRG 7	(%WD)	92.0	87.7
CRG 8	(%WD)	31.4	29.7
Cycle Burnup	(EFPD)	0.83	2.25
Boron Conc.	(PPM)	1835	1663
Imbalance	(%)	2.88	-1.94
Maximum Tilt	(%)	0.74	0.65
MDNBR		4.196	3.081
Worst Case MLHR (KW/FT)		9.00	12.01
Maximum Radial Peak			
Measured		1.327	1.337
Predicted		1.31	1.30
Difference	(%)	1.28	2.8
Acceptance Criteria (%)		≤5%	≤5%
Maximum Total Peak			
Measured		1.550	1.559
Predicted		1.53	1.52
Difference	(%)	1.29	2.5
Acceptance Criteria (%)		≤7.5%	≤7.5%

4.5 Reactivity Coefficients at Power

a. Purpose

The purpose of this test is to measure the temperature coefficient of reactivity and power Doppler coefficient 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 RCS temperature is decreased and then increased by about 5 degrees F. The reactivity associated with each temperature change is obtained from the change in controlling rod group position, and the values for the coefficient are calculated.

For measuring the power Doppler coefficient of reactivity, data is extracted from the fast insert/withdrawal sequences. Differential controlling rod worth measurements are also determined using the fast insert/withdrawal method.

c. Test Results

Temperature and power Doppler coefficient measurements were performed. At about 98% FP the measured moderator temperature coefficient was -4.27 pcm/°F. This verifies that the moderator temperature coefficient is negative above 95% FP.

The measured power Doppler coefficient at 98% FP was -8.78 pcm/%FP and the measured fuel Doppler coefficient was -1.34 pcm/°F. This meets the acceptance criteria of being more negative than -0.9 pcm/°F.

d. Conclusions

The measured moderator temperature coefficient (MTC) results indicate that the MTC is negative above 95% F.P.

The measured fuel Doppler coefficient (FDC) results meet the requirement that the FDC be more negative than -0.9 pcm/°F.