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April 23, 2010
TMI-10-040

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1)
RENEWED FACILITY OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289

SUBJECT: CYCLE 18 STARTUP REPORT

Enclosed is the Startup Report for TMI Unit 1 Cycle 18 operation. Resumption of commercial operation for Cycle 18 was achieved at 1:39 AM on January 24, 2010. This report is being submitted in accordance with TMI Unit 1 TS 6.9.1.A. No NRC response to this letter is necessary or requested.

Sincerely,

Rich Libra

Richard W. Libra
Plant Manager

RWL/dbn

Enclosure: TMI-1 Cycle 18 Startup Report

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

IE26
NRR

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TMI-1

CYCLE 18

STARTUP REPORT

TMI REACTOR ENGINEERING

January 2010

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1.0 INTRODUCTION

Per TMI-1 Tech Spec 6.9.1.A, a Startup Report shall be submitted to NRC following:
(1) receipt of an operating license,
(2) amendment to the license involving a planned increase in power level,
(3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and
(4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

This report is submitted because modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant were installed for TMI-1 Cycle 18, namely replacement once-through steam generators (OTSGs). The replacement OTSGs were installed under the 50.59 process and are not expected to have significantly altered the nuclear, thermal, or hydraulic performance of the plant. A Functional Test Procedure was implemented to monitor key parameters during plant heatup and startup to verify that the performance of the replacement OTSGs is similar to the previous OTSGs.

2.0 MEASUREMENTS AT ZERO POWER – SUMMARY

2.1 CORE PERFORMANCE

Core performance measurements were collected during the Zero Power Test Program, which began on January 23, 2010. Tests were conducted in accordance with AREVA guidance, which in turn is based on ANSI Standard 19.6.1. This section presents a summary of the zero power measurements. In all cases, the applicable test and Technical Specifications (TS) limits were met. A summary of zero power physics test results appears as Table 2-1.

Throughout this report, deviations expressed as a percent are calculated as follows:

Deviation = (Predicted - Measured)/Predicted

All boron values in this report are corrected for Boron-10 depletion unless otherwise noted.

a. Initial Criticality

Initial criticality was achieved at 5:14 AM on January 23, 2010. Reactor conditions were 532 F and 2157 psig. Critical conditions were achieved with rod groups 1 through 6 withdrawn to 100%; group 7 at 86.6% WD; group 8 at 100% WD (fully withdrawn), and boron concentration at 2110 ppmB. Initial criticality was achieved in an orderly manner and within the acceptance criteria of 2100 ± 50 ppmB (neither value corrected for B-10 depletion).

b. Nuclear Instrumentation Overlap

The overlap between the source and intermediate range detectors was 1 decade, meeting the 1-decade requirement of the Technical Specifications.

c. Reactimeter Checkout

An on-line functional check of the reactimeter using the average of the two intermediate range nuclear instruments (NIs), NI-3 and NI-4 was performed after initial criticality. Reactivity calculated by the reactimeter was within the acceptance criterion of ±5% of the core reactivity determined from doubling and halving time measurements.

d. <u>All Rods Out Critical Boron Concentration</u>

The measured all rods out critical boron concentration of 2066 ppmB was within the acceptance criteria of 2087 ± 50 ppmB.

e. <u>Moderator Temperature Coefficient Measurements</u>

The measured moderator temperature coefficient of reactivity at 532 F, zero power was +0.018 pcm/F, within the acceptance criteria limit of <+9.0 pcm/F.

f. <u>Control Rod Group Worth Measurements</u>

The measured results for control rod worths of groups 6 and 7 conducted at zero power (nominal 532 F) using the boron/rod swap method were in good agreement with predicted values. The maximum deviation between measured and predicted worths was -4.26%, which was for CRG-6 worth. This was within the acceptance criterion for group worth of ±15%. The deviation for the combined Group 6 and 7 worth was approximately -1.66%, well within the ±10% acceptance criterion.

2.2 REPLACEMENT OTSG PERFORMANCE

During plant heatup, hot shutdown, and zero power operation the replacement OTSGs were stable (no oscillations observed) and their thermal and hydraulic performance was as expected. The operators were able to control plant temperature and heatup rate the same as with the previous OTSGs while using the same procedures. All monitored parameters were within the expected range based on previous plant heatups, hot shutdown and zero power operation.

TABLE 2-1

SUMMARY OF ZERO POWER PHYSICS TEST RESULTS

CYCLE 17

Parameter	Predicted Value	Measured Value	Deviation
Critical Boron (not B-10 corrected)	2100 ± 50 ppmB	2110 ppmB	-10 ppmB
NI Overlap	≥ 1 decade	1 decade	N/A
Sensible Heat	N/A	7.2 E-8 amps	N/A
All Rods Out Boron Concentration	2087 ± 50 ppmB	2066 ppmB	21 ppmB
Temperature Coefficient (2046 ppmB)	-1.437 ± 2 pcm/F	-1.574 pcm/F	+0.137 pcm/F
Moderator Temperature Coefficient	+0.156 pcm/F (< +9 pcm/F)	+0.018 pcm/F	+0.138 pcm/F
Integral Rod Worths (532 F) GP 6 & 7	1588 pcm ± 10%	1562 pcm	1.6 %
Group 7	903 pcm ± 15%	906 pcm	-0.33 %
Group 6	685 pcm ± 15%	656 pcm	4.2 %

3.0 MEASUREMENTS AT POWER – SUMMARY

3.1 CORE PERFORMANCE

This section summarizes the physics tests conducted with the reactor at power. Testing was performed at power plateaus of approximately 17, 41, 74, and 100% core thermal power. Operation in the power range began on January 23, 2010.

Gadolinia is again present in the TMI-1 core as an integral burnable poison. Of the 88 assemblies reloaded from Cycle 17 and earlier, all 88 contain gadolinia. All 68 of the 89 of the fresh assemblies loaded for Cycle 18 contain gadolinia. These assemblies require no special monitoring.

There are no Lead Test Assemblies (LTAs) or Lead Use Assemblies (LUAs) in Cycle 18. However, all 89 of the fresh assemblies are of the Mark-B-HTP-1 design. The Mark-B-HTP-1 design differs from the Mark-B-HTP design in that it is slightly shorter to allow for additional fuel assembly growth. Generally, in the Mark-B-HTP design the fuel pin spacer grids are welded to the guide tubes, which creates a stiffer fuel assembly frame. The spacer grid design improves the thermal performance of the assembly and uses "line contact" to hold the fuel pins in place to reduce the likelihood of grid-to-rod fretting. The assembly uses the AREVA FUELGUARD™ lower end fitting for debris resistance. The fuel pin design is the same as that used in previous fuel assembly designs.

a. Nuclear Instrumentation Calibration at Power

The power range channels were calibrated as required during power escalation based on the primary and secondary plant heat balance. Power range calibration is affected by changes in core design, plant power level, boron concentration and/or control rod configuration changes during testing.

b. Incore Detector Testing

Tests conducted on the incore detector system demonstrated that the incore detector system was functioning acceptably. Symmetrical detector readings agreed within acceptable limits. The plant computer applied background, length and depletion correction factors to the incore detector signals. The backup incore recorders were operational above 75% full power (FP).

c. Power Imbalance Detector Correlation Test

The results of the Control Rod Group 7 movements performed at approximately 74 %FP show that an acceptable incore versus out-of-core offset slope could be obtained by using gain factors ranging from 3.451 to 3.871 for the power range scaled difference amplifiers. These were the same gain factors used in Cycles 16 and 17. The measured values of $F_{\Delta H}^{N}$ and F_{q}^{N} 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 41 %FP under non-equilibrium xenon conditions and at approximately 100 %FP at equilibrium xenon conditions. The maximum measured and maximum predicted radial and total peaking factors are all in good agreement. The largest percent difference between measured and predicted values for new fuel was -2.93% for total peaking at 100%FP. This met its acceptance criterion of >-6.5%. All assemblies were within their limits for radial and total peak.

The results of the core power distribution measurements are given in Table 5.4-1. All quadrant power tilts and axial core imbalances measured during the power distribution tests were within the applicable Technical Specification, Core Operating Limits Report (COLR), and normal operational limits.

3.2 REPLACEMENT OTSG PERFORMANCE

From zero to 100%RP the replacement OTSGs continued to show stable performance (no oscillations observed) and their thermal and hydraulic performance was as expected. The operators were able to control plant power the same as with the previous OTSGs while using their normal procedures. All monitored parameters were within the expected range based on previous power operating cycles and design features of the new OTSGs.

4.0 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, and measurements of all rods out critical boron concentration, temperature coefficient, and control rod worths.

4.1 INITIAL CRITICALITY

Initial criticality for Cycle 18 was achieved at 0514 hours on January 23, 2010. Reactor conditions were 532 F and 2157 psig. Control rod groups 1 through 4 and 8 were withdrawn prior to the approach to criticality. Deboration from the refueling concentration to the target concentration for criticality also occurred prior to the approach to criticality. The critical boron concentration was greater than the 1% $\Delta k/k$ shutdown concentration so the required shutdown margin was maintained.

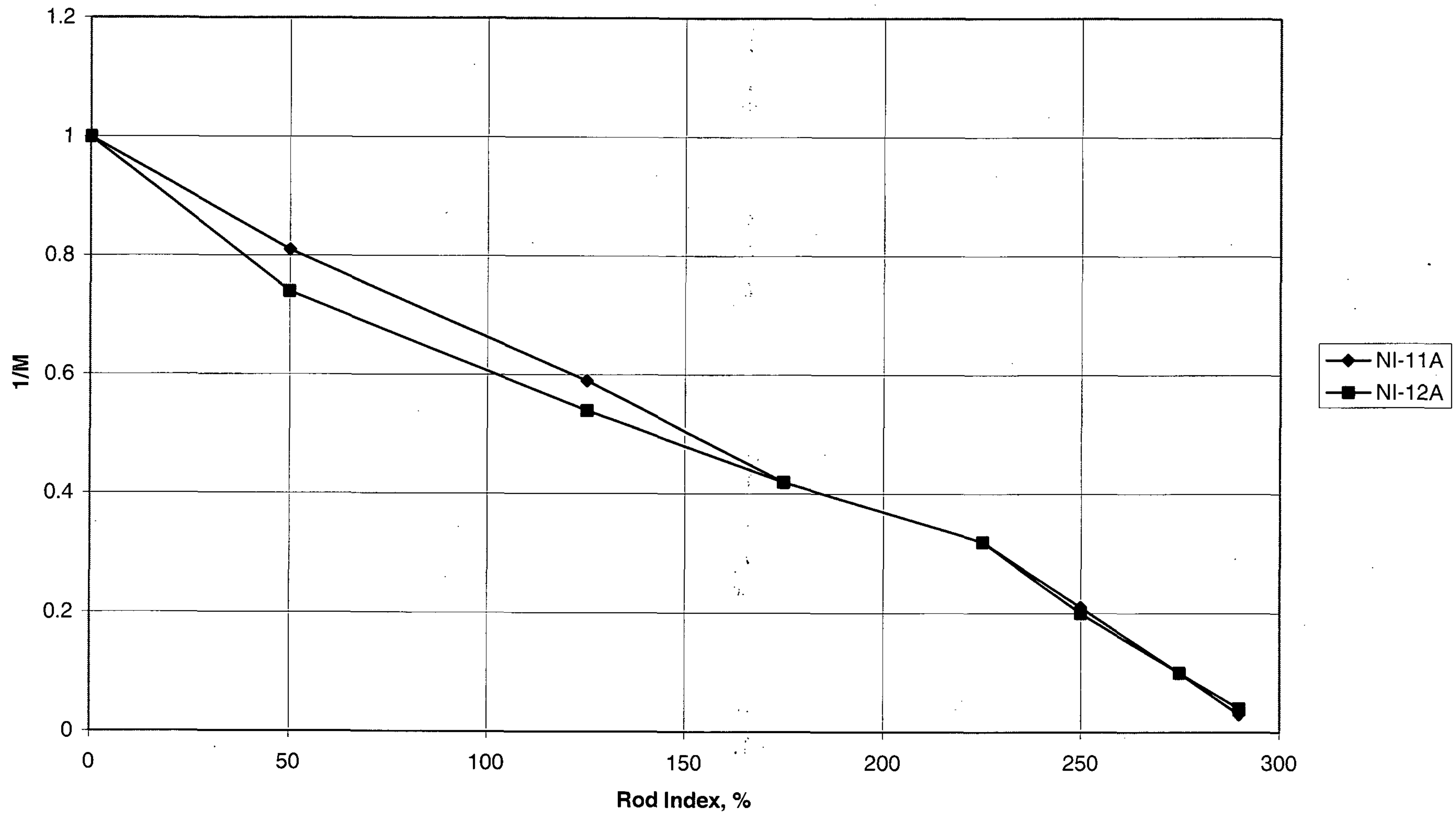
Criticality was achieved by withdrawing control rod groups 5 and 6 to 100% and control rod group 7 to 90%. As criticality was not achieved, control rod group 7 was inserted to 86.6%. The RCS boron concentration was diluted to achieve criticality in accordance with procedures.

Throughout the approach to criticality, independent persons maintained plots of inverse subcritical multiplication. Count rates were obtained from the source range neutron detector channels. Plots of inverse count rate (ICR) versus control rod position were maintained during control rod withdrawal.

The inverse count rate plots maintained during the regulating control rod withdrawal phase of the approach to criticality are presented in Figure 4.1-1. As can be seen from the plot, the response of the source range channels during reactivity additions was very good.

In summary, initial criticality was achieved in an orderly manner. The RCS ^{10}B depletion correction factor was 0.98164. The measured critical boron concentration at the critical rod position was 2100 ppmB, within the acceptance criteria of 2110 ± 50 ppmB (2061 and 2071 ± 50 ppmB, respectively, when corrected for B-10 depletion).

Figure 4.1-1
TMI-1 Cycle 18
1/M vs. Rod Position for Initial Cycle Startup



4.2 NUCLEAR INSTRUMENTATION OVERLAP

a. Purpose

Technical Specification 3.5.1.5 states that prior to operation in the intermediate NI range, at least one decade of overlap between the source range NIs and the intermediate range NIs must be observed.

b. Test Method

To satisfy the above overlap requirements, core power was increased while both the source and intermediate range channels were on scale. Detector signal response was then recorded for both the source and intermediate range channels.

c. Test Results

The results of the initial NI overlap data at 532 F and 2155 psig (nominal pressure and temperature) have shown a ≥ 1 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.

4.3 REACTIMETER CHECKOUT

a. Purpose

Reactivity calculations during the Cycle 18 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 was established, the reactimeter and the reactivity calculations were started. Steady state conditions were established and a small amount of positive reactivity was inserted in the core by withdrawing control rod group 7. Then a small amount of negative reactivity was inserted in the core by inserting control rod group 7 to slightly below the critical position.

Reactivity Measurement and Analysis System (RMAS) software compared the reactivity calculated from the doubling and halving times to the values calculated by the reactimeter. Measurements were taken at approximately +98 and -30 pcm.

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 within 1% of the values obtained from neutron flux doubling times compared to the acceptance criterion of <5%.

4.4 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 approximately 532 F and 2155 psig.

b. Test Method

Starting from the critical condition, the Group 7 control rods were withdrawn to 92.4% withdrawn. The resulting reactivity change was measured with the reactimeter. The small additional reactivity between 92.4% withdrawn and fully withdrawn was determined from the predicted control rod worth tables in accordance with procedures. The boron equivalent of the total reactivity difference between the critical condition and "all rods out" condition was calculated and added to the lab-measured RCS boron concentration. This value is the measured all rods out (ARO) boron concentration

c. Test Results

The measured ARO boron concentration, corrected for B-10 depletion, with group 7 positioned at 100% withdrawn (WD) was 2066 ppmB.

d. Conclusions

The above results show that the measured boron concentration of 2066 ppmB is within the acceptance criterion of 2087 ± 50 ppmB.

4.5 TEMPERATURE COEFFICIENT MEASUREMENT

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, Technical Specification 3.1.7.2 states that the moderator temperature coefficient shall be less than or equal to +9.0 pcm/F at power levels less than or equal to 95% FP. In addition, Technical Specification 3.1.7.1 states that the moderator temperature coefficient shall not be positive while greater than 95 %FP. COLR Figure 9 is more restrictive, requiring a negative coefficient above 80 %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 neutron flux, reactor coolant pressure, turbine header pressure and core average temperature constant, with the reactor critical at approximately 5 E-9 amps on the intermediate range. Equilibrium boron concentration was established in the Reactor Coolant System (RCS), make-up tank and pressurizer to eliminate reactivity effects due to boron changes during the subsequent temperature swings. The reactivity value and the RCS average temperature were displayed on the RMAS monitor.

Once steady state conditions were established, a heatup was initiated by closing the turbine bypass valves. After the core average temperature increased by about 3°F, core temperature and flux were stabilized. The process was reversed by decreasing the core average temperature by about 3°F, returning the RCS temperature to nearly 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 ppmB	Measured ITC pcm/F	Predicted ITC pcm/F	Calculated MTC pcm/F	Required MTC pcm/F
2046	-1.574	-1.437	+0.018	<+9.0

d. Conclusions

The measured values of the moderator temperature coefficient of reactivity at 532 F, zero reactor power are within the acceptance criteria of ± 2.0 pcm/F of the predicted value, and less than the Technical Specification 3.1.7.2 criterion of +9.0 pcm/F. An extrapolation of the moderator temperature coefficient to 80%FP indicated that it was well within the limits of TS 3.1.7.1 and the more restrictive limits of the COLR.

4.6 CONTROL ROD GROUP WORTH MEASUREMENT

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 4.6-1. The grouping of the control rods shown in Figure 4.6-1 will be used throughout Cycle 18. Calculated and measured control rod group reactivity worths for the normal withdrawal sequence were determined at nominal reactor conditions of zero power, 532 F and 2155 psig. The measured results were obtained using results of reactivity and group position from the RMAS system.

Figure 4.6-1
Control Rod Locations and Group Descriptions for TMI-1 Cycle 18

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

| X | Group Number

Group Number	Number of Control Rods in the Group	Control Rod Function
1	8	Safety
2	8	Safety
3	8	Safety
4	8	Safety
5	12	Control
6	8	Control
7	9	Control
8	8	APSR*

* For Cycle 18, the APSRs must be positioned more than 98% withdrawn prior to criticality. Except for rod motion surveillance testing and shutdowns, they must be maintained fully withdrawn during power operation for the rest of the cycle.

b. Test Method

Control rod group reactivity worth measurements were performed at zero power, 532 F and 2155 psig using the boron/rod swap method as described in Topical Report BAW-10242P-A. Both the differential and integral reactivity worths of control rod groups 6 and 7 were determined.

The boron/rod swap method consists of establishing a deboration rate in the RCS, then compensating for the reactivity changes by manually 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

The integral reactivity worths for control rod groups 6 through 7 are presented in Figures 4.6-2 and 4.6-3.

These curves were obtained by integrating the measured differential worth curves. Table 4.6-1 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 -4.3%.

d. Conclusions

Differential and integral control rod group reactivity worths were measured using the boron/rod swap method. Measurements met the criteria for performing the test on only groups 6 and 7. The measured results at zero power, 532 F and 2155 psig indicate good agreement with the predicted group worths. All individual group worths and the combined worth met their acceptance criteria.

Figure 4.6-2
Integral Worth for CRG-6
Cycle 18

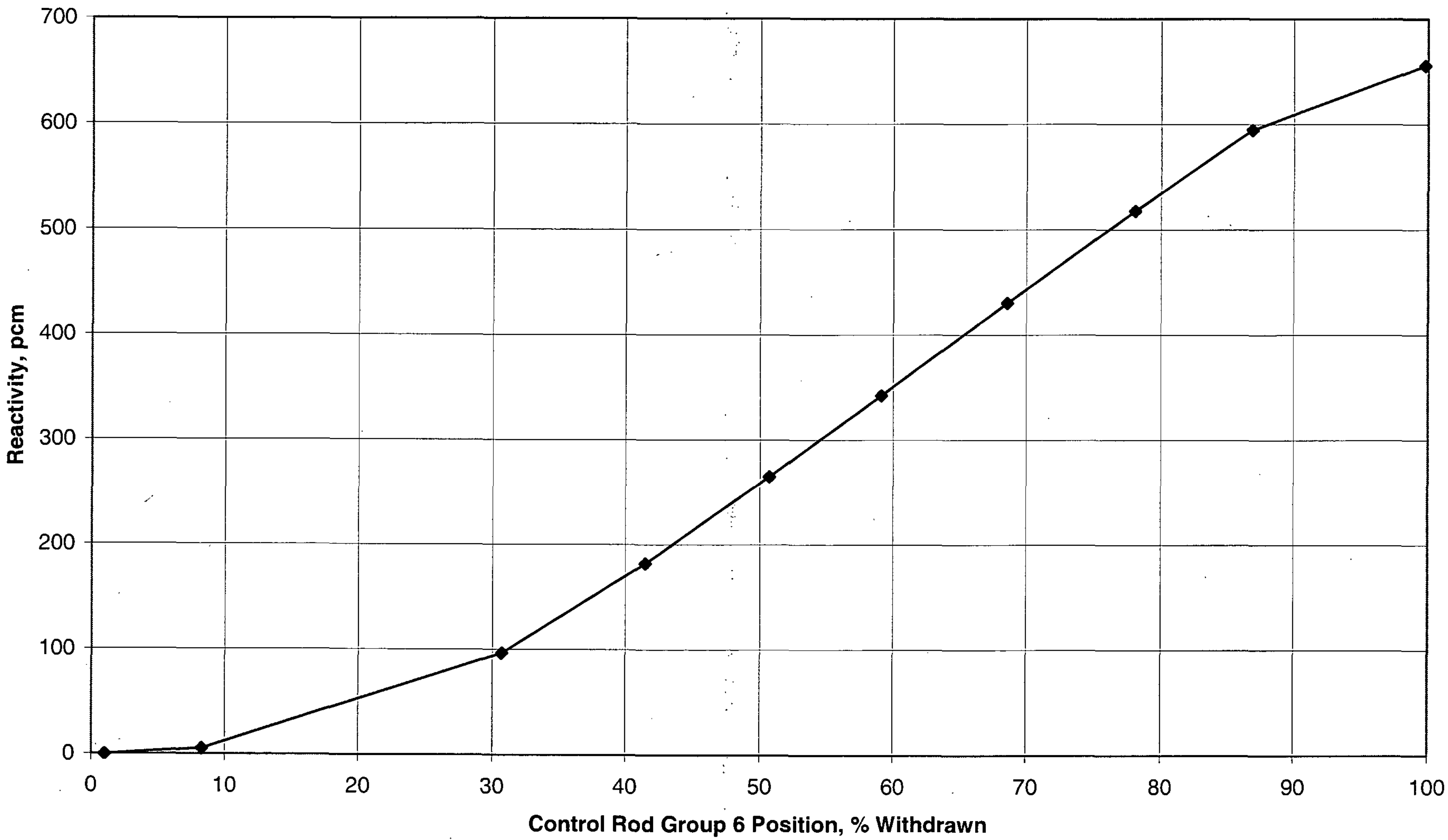


Figure 4.6.3
Integral Worth for CRG-7
Cycle 18

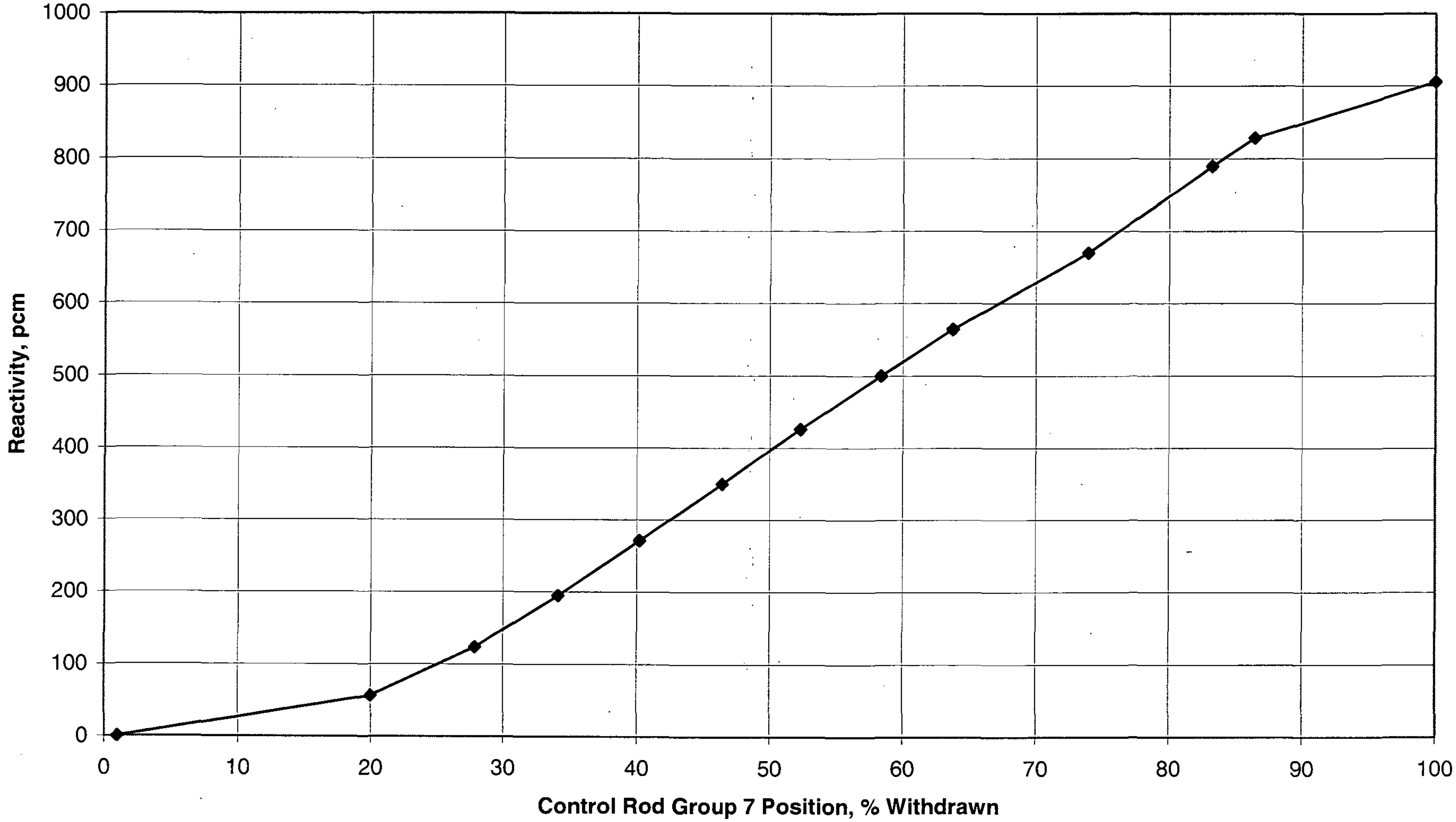


TABLE 4.6-1

COMPARISON OF PREDICTED VS MEASURED ROD WORTHS

Control Rod Group	Predicted Worth, pcm	Measured Worth, pcm	Percent Difference
6	685.0 ± 15%	655.8	-4.26%
7	903.0 ± 15%	905.9	+0.32
6 & 7	1588 ± 10%	1561.7	-1.66%

5.0 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 17%, 41%, 74%, and 100% of 2568 megawatts core thermal power, as determined from primary and secondary heat balance measurements. Operation in the power range began on January 23, 2010.

Periodic measurements and calibrations were performed on the plant's power range NIs during the escalation to full power. The four power range 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 plant computer processed the detector inputs correctly. Core axial imbalance determined from the incore instrumentation system was used to calibrate the out-of-core detector (power range) imbalance indication.

The major physics measurements performed during power escalation and at full power consisted of obtaining detailed radial and axial core power distribution measurements. Also, during power escalation, nuclear instrument response was determined for several core axial imbalances. Values of $F^N_{\Delta H}$ and F^N_q were monitored throughout the test program to ensure that core thermal limits would not be exceeded.

5.1 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 within 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 within 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:

Nominal Test Plateau, %FP	Nominal Bistable Setpoint, %FP
≤40	50
≤80	90
100	105.1*

*Normal full power setpoint

c. Test Results

An analysis of test results obtained at less than hot full power equilibrium conditions indicated that changes in core design, RCS boron, and xenon buildup or burnout affected the power as indicated by the NIs. This was expected since the power range NIs measure 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 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 ± 2.5% axial offset criteria as determined by the incore monitoring system was also met when required.

The high flux trip bistable was adjusted to a nominal setpoint of 50, 90 and 105.1 %FP prior to escalation of power to nominal physics testing plateaus of ≤40, ≤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 core design, power level, RCS boron, and/or control rod configuration changes during the program. Acceptance criteria for nuclear instrumentation calibration at power were met in all instances.

5.2 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) 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 powers of approximately 17, 41, and 100 %FP.

At approximately 74 %FP, Surveillance Test Procedure 1301-5.3, "Incore Neutron Detectors-Monthly Check," was performed to calibrate the backup recorder detectors to their incore depletion value. This was also a prerequisite for the Power Imbalance Detector Correlation Test described in the next section (5.3).

Five detectors were identified as failed. However, the number and location of the failed detectors were within limits defined in the COLR Full Incore System (FIS) Operability Requirements. The incore monitoring software made appropriate substitutions for the failed detectors.

c. Conclusions

Incore detector testing during power escalation demonstrated that the detectors were functioning acceptably. Symmetrical detector readings agreed within acceptable limits and the computer applied correction factors are accurate. The backup incore recorders were calibrated and were operational above 75 %FP.

5.3 POWER IMBALANCE DETECTOR CORRELATION TEST

a. Purpose

The Power Imbalance Detector Correlation Test (PIDC) has three objectives:

(1) To determine the relationship between the core power distribution as measured by the out-of-core detector system (OCD) and the incore detector system (ICD) instruments.

(2) To verify the adequacy and accuracy of backup imbalance calculations as done in OP-TM-300-203, "Quadrant Power Tilt and Axial Power Imbalance Using the Minimum Incore System."

(3) To determine the core $F_{\Delta H}^{N}$ and F_{q}^{N} at various power imbalances.

b. Test Method

This test was conducted at about 74 %FP. The data showed 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 $F_{\Delta H}^{N}$ and F_{q}^{N} limits would not be exceeded by operating within the flux/delta flux/flow envelope set in the Reactor Protection System.

In accordance with the AREVA Power Escalation Test Specification, this test was conducted at TMI-1 without using CRG-8, the Axial Power Shaping Rods as in Cycle 17. CRG-7 was moved to establish the various imbalances by adjusting RCS boron concentration. The integrated control system (ICS) automatically compensated for the boron-induced reactivity changes by repositioning CRG-7 to maintain a constant power level.

The core offset as measured by the full incore detector (ICD) system was plotted against the offset as indicated by the out-of-core (OCD) NIs. The slope of the correlation for each NI must be between 0.98 and 1.10 to meet the acceptance criterion.

c. Test Results

The relationship between the ICD and OCD offset was determined at about 74 %FP by changing axial imbalance through adjustment of the boron concentration, and resulting Group 7 control rod position. The results met the criteria for conducting the test using three measurements of the ICD to OCD offset slope. The slope measured on the four OCDs ranged from 1.007 (NI-8) to 1.022 (NI-5). Results indicated that none of the four detectors required an adjustment to their respective scaled difference amplifier gains.

A comparison of the ICD offset versus the OCD detector offset and the resulting slopes obtained for each NI channel is shown in Table 5.3-1.

Core power distribution measurements were taken at the most positive and negative imbalances at about 74 %FP. All values were within their respective acceptance criteria.

Backup offset calculations using OP-TM-300-203 compare well with the computer-calculated offset. Table 5.3-2 lists the core offsets calculated using the full incore system as well as the offsets obtained using the ICD backup recorders.

d. Conclusions

Backup imbalance calculations performed in accordance with OP-TM-300-203 provide an acceptable alternate method to computer-calculated values of imbalance.

Power distribution parameters were within Technical Specifications limitations.

The "as found" slopes of the ICD to OCD correlations indicated that none of the four power range detectors required adjustment to be within the acceptance criteria.

TABLE 5.3-1

INCORE OFFSET VS OUT-OF-CORE OFFSET

Incore Offset, %	Out-of-Core Offset, %			
	NI-5	NI-6	NI-7	NI-8
12.64	13.65	13.66	13.69	13.39
0.05	0	0.08	0.08	0.01
-12.72	-12.26	-12.17	-12.19	-12.16
Resulting Incore vs. NI Slope	1.022	1.018	1.020	1.007

TABLE 5.3-2

FULL INCORE OFFSET VS BACKUP RECORDER OFFSET

Full Incore Offset, %	Backup Recorder Offset, %
12.64	12.3
0.05	-0.3
-12.72	-11.0

5.4 CORE POWER DISTRIBUTION VERIFICATION

a. Purpose

To measure the core power distributions during the power escalation and at 100 %FP to verify that the core axial imbalance, quadrant power tilt, $F_{\Delta H}^{N}$, and F_{q}^{N} do not exceed their specified limits. Also, to compare the measured and predicted power distributions.

b. Test Method

Core power distribution measurements were performed at approximately 41%FP during the power escalation and at 100 %FP under steady state conditions. To provide the best comparison between measured and predicted results, three-dimensional equilibrium xenon conditions were established for the full power test. Data collected for the measurements consisted of power distribution information from the fixed incore detector system. The worst case core thermal conditions were calculated using this data. The measured data was compared with calculated predictions.

c. Test Results

The acceptance criteria for power distribution require that all new fuel be within limits for radial and total peaking. Also, the root mean square (RMS) of the differences between measured and predicted HFP radial peaks for all fuel (eighth core) should be less than 0.05.

A summary of the Core Power Distribution test results is found in Table 5.4-1. The table lists the core power level, control rod positions, cycle burnup, boron concentration, axial imbalance, maximum quadrant tilt, $F_{\Delta H}^{N}$ and F_{q}^{N}, and power peaking data for each measurement.

Note that the radial and total peak data are not necessarily for the maximum peaks in the core, but for the locations with the largest difference between the predicted and measured data for new fuel. The radial peak and total peak limits are shown. The largest deviation between the maximum measured and maximum predicted peak value was -2.93% for total peaking at 100 %RP for location L-14. This met its acceptance criterion of >-6.5%.

The minimum F_{q}^{N} margin (LOCA LHR) was 24.70% at 100 %RP. The minimum $F_{\Delta H}^{N}$ margin (DNBR) was 9.65% at 100 %RP. Both values were within their expected ranges.

The quadrant power tilt and axial imbalance values measured were all within the allowable limits. Table 5.4-1 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 41%FP and 100 %FP. Comparison of measured and predicted results show good agreement. All fuel locations met their acceptance criteria.

The measured values of F_q^N and $F_{\Delta H}^N$ were all within the allowable limits. All quadrant power tilts and axial core imbalances measured during the power distribution test were within the applicable Technical Specifications, COLR, or normal operating limits.

5.5 REPLACEMENT OTSG PERFORMANCE

a. Purpose

The purpose of the OTSG performance tests was to verify level stability, verify sufficient Reactor Coolant System flow, and to verify adequate steam superheat.

b. Test Method

Testing was conducted by passive data collection and analysis.

c. Results: Replacement OTSG Level Testing

1. Both the A and B startup level indication showed no oscillations (allowed: 60 cycles) during the one hour test.
2. At low powers on the operating range level indication a total of six tests were performed. The highest peak to peak was 2.15% on the A OTSG and 2.4% on the B OTSG with a limit of 4%.
3. At about 72% power the operating range level indication showed a peak to peak of 0.83% on A OTSG and 0.71% on B OTSG.
4. Three tests were performed above 75% power on the operating range level indication with the highest reading on A OTSG at 1.18% and on the B OTSG at 1.11%.The peak to peak readings continued to get smaller as power was raised.
5. At 100%RP the operating range level indication for the A OTSG was 63.4% and for the B OTSG 63.5%. The minimum level limit is 50% and the maximum level limit is 80%.

d. Results: Reactor Coolant System (RCS) Flow

1. The RCS flow noise was within the allowed 2.5%.
2. All four RCS total flow channels were within $\pm$ 1.5% of their average.
3. The RCS flow at 100%RP was 147.26 Mlb/hr with an allowed minimum of 137.7 MLB/hr.

e. Results: Superheat

Clean, unplugged replacement OTSGs were expected to deliver >56°F superheat at the nominal full power, pressure, temperature, and feedwater flow conditions.

1. A OTSG measured superheat was 56.9°F
2. B OTSGmeasured superheat was 58.5°F

f. <u>Conclusions</u>

Replacement OTSG thermal-hydraulic performance was within expectations and met acceptance criteria for level stability, RCS flow, and steam superheat.

TABLE 5.4-1

CORE POWER DISTRIBUTION RESULTS

Power Plateau	Escalation 41%	Steady State 100%
Date	24 January 2010	29 January 2010
Reactor Power (%FP)	41.30	99.96
CRG 1-4 (%WD)	100	100
CRG 5 (%WD)	100	100
CRG 6 (%WD)	100	100
CRG 7 (%WD)	40.7	91.7
CRG 8 (%WD)	99.0	98.9
Cycle Burnup (EFPD)	0.152	4.545
Boron Concentration (ppmB) Corrected for B-10 depletion	1802	1476
Imbalance (%)	-6.97	2.19
Maximum Tilt (%)	2.17	0.97
Tilt Limit (%)	6.83	4.34
Minimum $F^N_{\Delta H}$ margin, % (DNBR)	27.67	9.65
Minimum F^N_q margin, % (LOCA LHR)	63.05	24.70
Maximum Radial Peak Difference, New Fuel		
Location	L-14	L-14
Measured Peak	1.229	1.181
Predicted Peak	1.206	1.152
Deviation (%)	-1.91	-2.49
Acceptance Criterion (%)	>-5.0	>-5.0
Maximum Total Peak Difference, New Fuel		
Location	L-14	L-14
Measured Peak	1.651	1.421
Predicted Peak	1.618	1.381
Difference (%)	-2.03	-2.93
Acceptance Criterion (%)	>-6.5	>-6.5
Eighth-Core RMS of Absolute Differences for Radial Peaks, All Fuel		
Measured	0.0191	0.0154
Acceptance Criterion	≤0.05	≤0.05