HEDL-SA-2163

INITIAL PHYSICS MEASUREMENTS ON FFTF

R.A. Bennett, J.W. Daughtry,
R.A. Harris, D.H. Jones,
T.L. King, J.C. Midgett, J.L. Rathbun,
R.B. Rothrock, B.D. Zimmerman

American Nuclear Society

June 8-13, 1980

Las Vegas, Nevada

HANFORD ENGINEERING DEVELOPMENT LABORATORY
Operated by Westinghouse Hanford Company, a subsidiary of Westinghouse Electric Corporation, under the Department of Energy Contract No. DE-AC14-76FF02170
P.O. Box 1970, Richland, Washington 99352

DISCLAIMER

HEDL-SA-2163

INITIAL PHYSICS MEASUREMENTS ON FFTF

R.A. Bennett, J.W. Daughtry,
R.A. Harris, D.H. Jones,
T.L. King, J.C. Midgett, J.L. Rathbun,
R.B. Rothrock, B.D. Zimmerman

American Nuclear Society

June 8-13, 1980

Las Vegas, Nevada

I. INTRODUCTION

Initial criticality of the Fast Flux Test Facility (FFTF) was achieved on February 9, 1980 at 3:45 p.m. The FFTF is operated by Westinghouse Hanford Company (WHC) for the US Department of Energy.

During the period November 27, 1979 to March 8, 1980, fuel was loaded into the FFTF core, initial criticality was achieved, and several subcritical physics measurements were performed. This initial phase of FFTF nuclear operation was the culmination of nearly fifteen years of effort by WARD,* WHC, and many other organizations within the United States. There is, therefore, considerable interest from the fast breeder reactor community in having available the data obtained to date. In response to this interest, the data obtained from initial FFTF nuclear operation are presented in this report. These data are of further interest with respect to the evaluation of nuclear methods used to design FFTF. It is expected that the data will have application in evaluation of methods being or to be employed in design of follow-on FBR systems.

Specifically, the absolute and relative neutron count rates were predicted for the bulk of the seventy-three fuel loadings of FFTF. Agreement between predicted and observed values is illustrated. Severe variations of fission chamber detection efficiency in the reactor shield is contrasted with that near the core center.

Control rod worths, measured by the rod drop inverse kinetics method, are compared with predictions based upon Engineering Mockup Critical (EMC) evaluations. Control rod reactivity worth curves measured by rod run-in inverse kinetics are given. Rod interactions are apparent in rod worth tables. Isothermal temperature coefficient near the refueling temperature of 400°F was determined to be negative and of the expected magnitude (-0.7¢/°F). Flow-induced reactivity effects were measured to be small. Accuracy of Modified Source Multiplication (MSM) method of reactivity

*Westinghouse Advanced Reactors Division.

assessments at fully shut down conditions was determined and found to be acceptable. In summary, FFTF nuclear characteristics are essentially as designed; all safety requirements are satisfied. From a nuclear point of view, FFTF is qualified to proceed into a power operation mode.

It should be recognized that the data presented herein have been evaluated only to the extent necessary to ensure that adequate data were obtained. Interpretation of the data and detailed comparisons with prediction techniques will be the subjects of future reports.

II. FUEL LOADING AND INITIAL CRITICALITY

At 3:45 p.m. on February 9, 1980, the first self-sustaining nuclear chain reaction occurred in the reactor core of the Fast Flux Test Facility (FFTF). This event was achieved with fifty-nine fuel assemblies installed in the core with the three primary safety rods fully withdrawn to a height of 36.5 inches and the six secondary control rods banked at a nominal height of 31.3 inches.

The initial approach to criticality actually commenced on November 27, 1979, at six minutes before 1:00 a.m., when the first fuel assembly was inserted into a core position. This date marked the beginning of a fuel loading and nuclear status monitoring process that ended on February 19, 1980, at 7:36 a.m., when the seventy-third and final fuel assembly was inserted into the core.

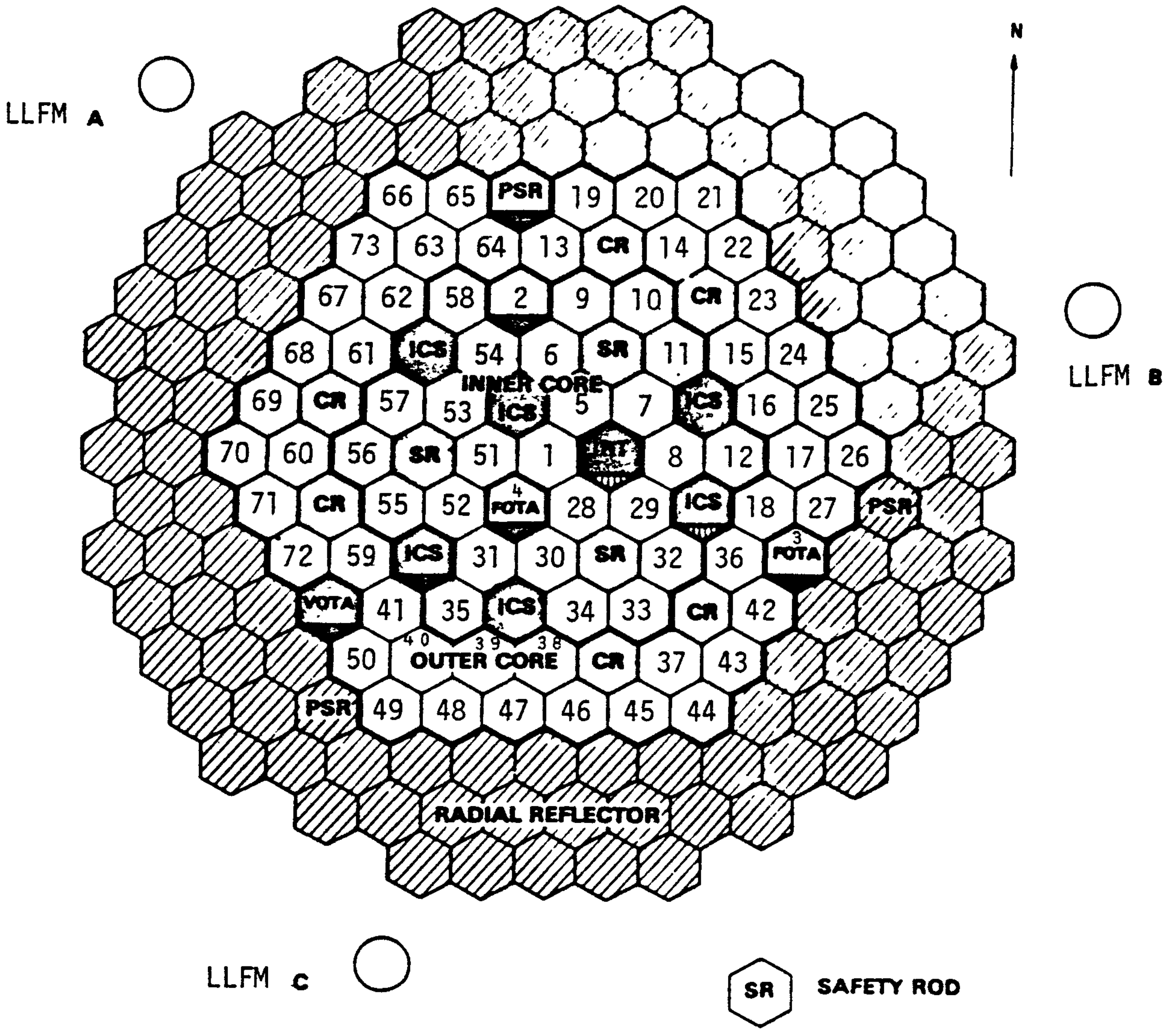
To prepare for the loading of the first fuel assembly, all nonfuel assemblies, including reflector assemblies, control rods, safety rods, shim rods, and specially instrumented assemblies, were installed into the hexagonal array core by November 13, 1979. All core positions planned for fuel assemblies had been loaded with Simulated Core Assemblies (SCA), which were to be replaced with fuel assemblies, one at a time, in a preplanned manner,[1] the sequence of which will be identified in later paragraphs. In addition, final check-out of the instrumentation system that would monitor the nuclear status of the reactor throughout the fuel loading process was completed. Figure 1 presents a top view of the core arrangement just prior to the start of fuel loading.

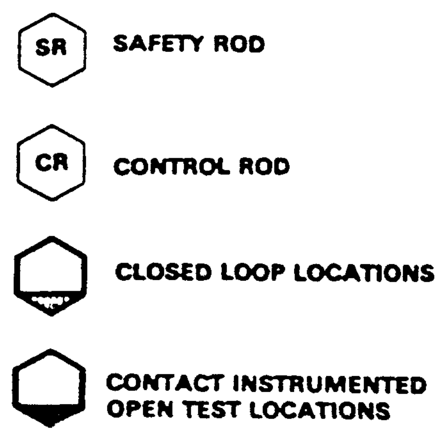
The neutron monitors used for initial fuel loading consisted of six ^{235}U fission chambers, three of which were the standard plant equipment Low Level Flux Monitors (LLFM). The LLFMs are symmetrically located at the core midplane at three positions in the surrounding radial shielding 113 cm from the core centerline. The LLFMs are capable of being retracted to a position approximately five feet above the core midplane, thereby extending the power level monitoring range capability of the LLFMs. The other three chambers were special, temporary startup chambers used only during the initial fuel loading process. One of the chambers was located at the core midplane and the other two were located vertically above and below it, near the top and bottom of the active fuel region in the In-Reactor Thimble (IRT), a reentrant tube installed in a test position adjacent to the central core position. Shown in Figure 1 are the locations of the three LLFMs, labeled A, B and C, and the IRT. Also shown in Figure 1 are the locations of fuel assemblies ordered in sequence from the first through the last assembly loaded. The LLFM monitors are manufactured by Westinghouse Electric Corporation, Model WL-23831, each containing 2.3 grams of Uranium-235. The IRT monitors were Reuter-Stokes, Inc. units, Model RS-C3-2510-114, each containing 1.3 grams of Uranium-235.

Counting data from the LLFMs and IRT chambers were collected, analyzed and recorded by a small computer and printer system. The input to the computer consisted of real time, integrated counts from six individual scalers, and manual entries of selected plant parameters. Each scaler received ^{235}U fission chamber pulse signals from one of the six monitors. The scalers were gated simultaneously by a single timer.

Figure 1.

FFTF CORE MAP
NO FUEL LOADED





Legend

PSR - Peripheral Shim Rod
ICS - In-Core Shim Assembly
FOTA - Fuel Open Test Assembly
VOTA - Vibration Open Test Assembly
A,B,C - Locations of Low Level Flux Monitors (fission chambers)
IRT - In-Reactor Thimble (contains three fission chambers)
1,2,3 etc. - Denotes fuel assembly positions ordered from first through last (seventy-third) loaded

HEDL 7912 014 1

The first fuel assembly loaded was inserted into the central position of the hexagonal array core. Just prior to insertion of the fuel assembly, a background assessment was performed, yielding count rates ranging from 0.4 to 1.5 counts per second on the monitors. During and after the insertion of the first fuel assembly, counts from each of the six monitors were collected, analyzed and recorded, as was the case with all subsequent fuel assembly insertions. Figure 2 shows the core arrangement after the loading of the first fuel assembly.

The second fuel assembly was inserted into the core on November 29, 1979, into a test position over which an instrumented stalk was then installed. The next two fuel assemblies installed were Fueled Open Test Assemblies (FOTA) instrumented to provide later confirmation of natural circulation heat removal capability. Figure 3 shows the core arrangement after the loading of the fourth fuel assembly. By this time, the fuel that remained to be loaded could be identified as belonging to one of three trisectors, formed and bounded by the three offset-Y-pattern legs along which lie the test positions (see Figure 3).

After the fourth fuel assembly (the second FOTA) was installed, count rates were obtained that served as the normalization point for beginning the plotting of inverse count rates, and predictions therefrom, of the minimum critical loading of fuel. Initial count rates corrected for background, are given in Table I. Uncertainties are 1σ from counting statistics.

Figure 2.

FFTF CORE MAP
1 FUEL LOADED

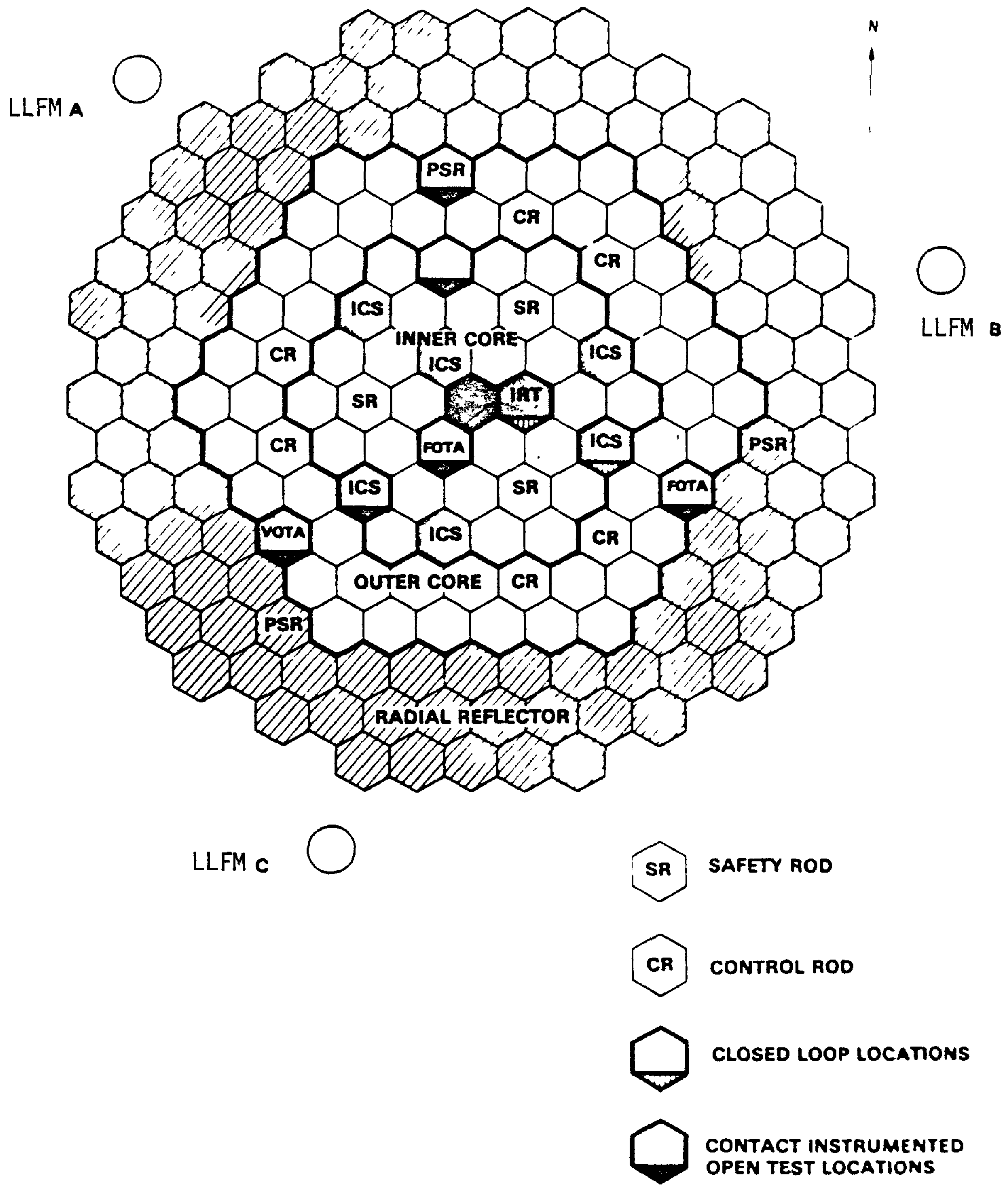
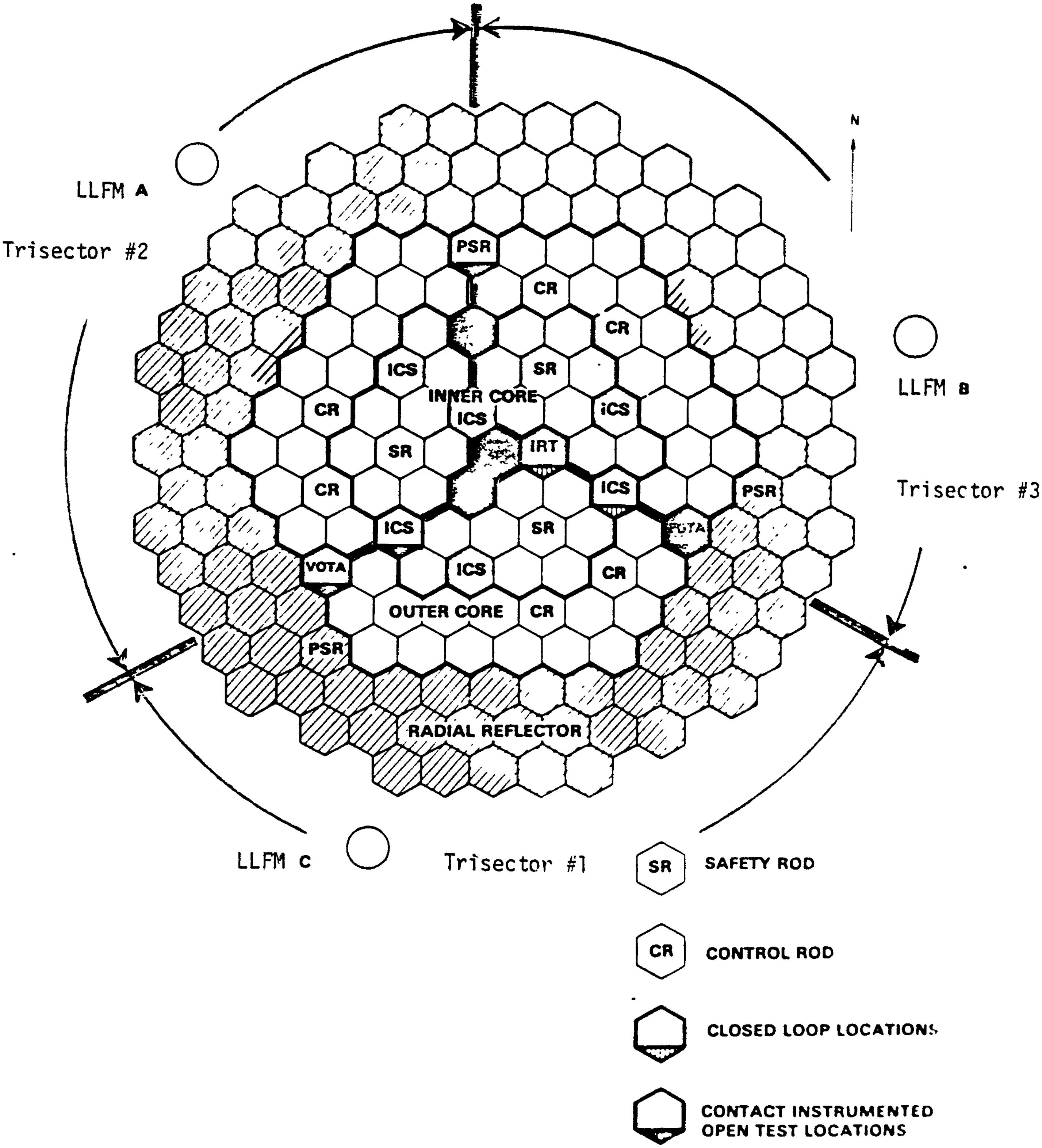


Figure 3.

FFTF CORE MAP FUEL "PRE-LOAD"



HEDL 7912-014.1

TABLE I

INITIAL COUNT RATE NORMALIZATION DATA

Count Rate (cps) With Four Fuel Assemblies Loaded

LLFM*			IRT**		
A	B	C	Top	Middle	Bottom
0.81±0.05	1.25±0.11	0.58±0.06	15.6±0.2	18.7±0.2	13.6±0.2

* neutron sensitivity 1.3 cps/equivalent thermal neutron flux at 1 cps alpha cutoff.

** neutron sensitivity 0.7 cps/equivalent thermal neutron flux at 1 cps alpha cutoff.

The fuel handling system is also divided into three parts, each part of which has an In-Vessel Fuel Handling Machine (IVHM) designed to handle fuel in one of the three trisectors bounded by the test positions. Also, the in-core flow and temperature sensors are bundled into one of three "Instrument Tree" packages, each of which serves one of the three trisectors. Each instrument tree also contains parts of the control rod driveline systems for one primary safety rod and two secondary control rods. Prior to any core component changeout in a given trisector, it is necessary to disengage the instrument tree part of the control rod driveline from the drive motor above and absorber assembly below. This frees the instrument tree to be swung out, allowing the IVHM access to the core. Because of this arrangement, it was most efficient and advantageous to load fuel into one tri-sector at a time. Then, as each trisector loading was completed, the instrument tree for that trisector could be rotated and lowered over the core, the control rod drivelines could be connected, and the control rods in that trisector could then be manipulated through electrical mechanisms located above the reactor vessel closure head.

Inspection of Figure 3 illustrates that completing the loading by trisector shifts the effective nuclear center of the core as the loading progresses. Because of this shifting, the neutron detectors were expected to experience large changes in detection efficiency, making conventional inverse count rate plots difficult to interpret. Therefore, extensive preanalyses were performed, covering the stages of the loading process, to assist in the interpretation of the count rate data, and to establish criteria within which the fuel loading could proceed.

The analyses provided predictions of relative count rates for the IRT monitors and the LLFMs for selected numbers of fuel assemblies loaded and various control rod configurations. The analyses also provided predictions of control rod worths and the fuel loading for all-rods-out criticality, including uncertainty.[2]

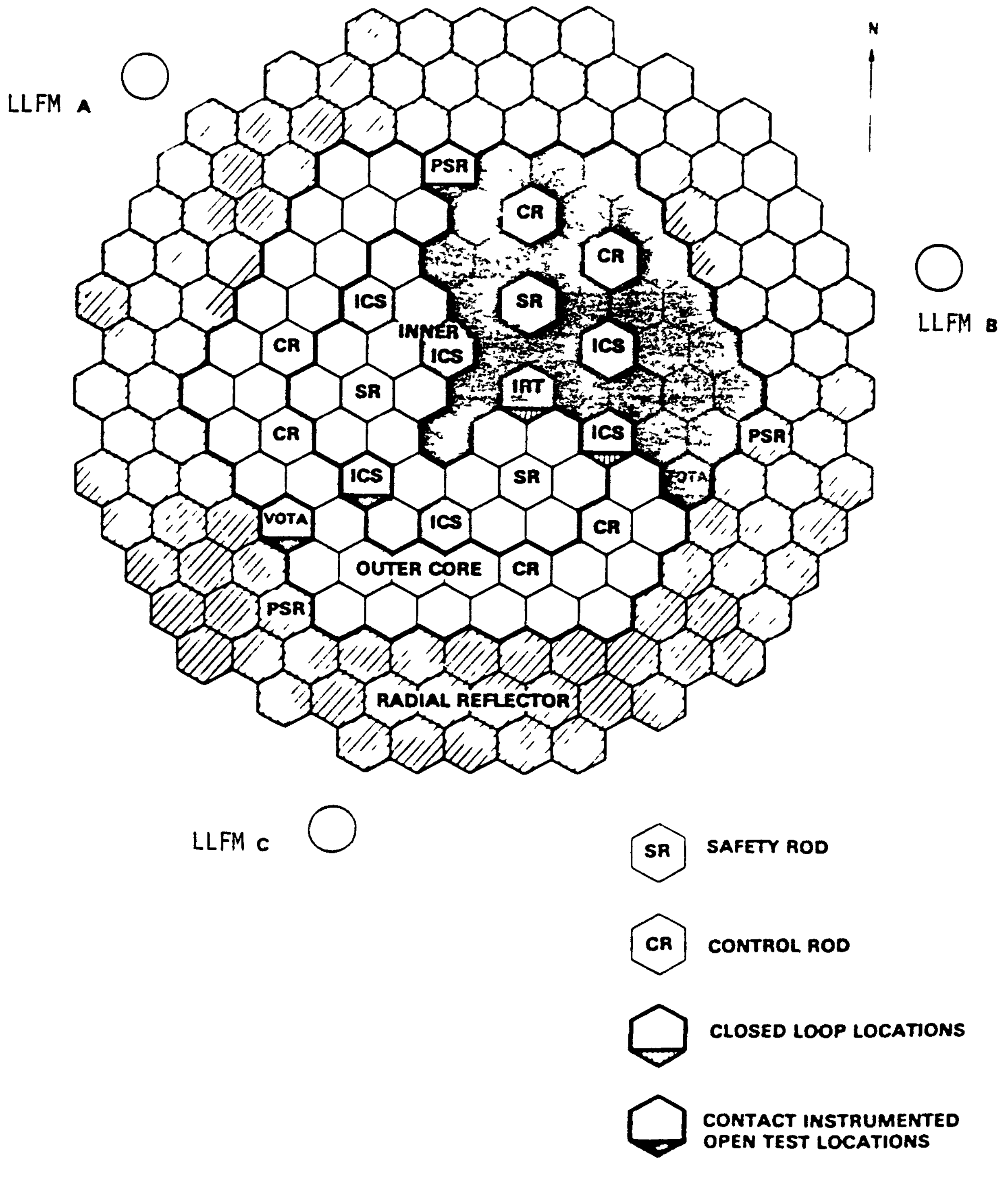
A two-dimensional, multigroup diffusion theory computer code[3] employing a triangular mesh core midplane model was used for the bulk of the analyses. Accuracy of the two-dimensional model was checked using a three-dimensional code[4] for selected configurations. Results were also compared with and adjustments were made[2] through experiment/theory correlations using results of a series of dedicated critical experiments.[5,6]

Four-group neutron cross sections were prepared[2] from the FTR Set 300S, 42-group cross section library[7] using the 1DX computer program.[8]

The first trisector loaded with fuel was Trisector #3, adjacent to LLFM B (see Figure 3). Loading commenced on December 11, 1979, with the loading of the fifth fuel assembly into the core, and the trisector was completed on December 23, 1979, with the loading of the twenty-seventh fuel assembly into the core. Figure 4 shows the core arrangement after completion of fuel loading into the first trisector.

Figure 4.

FFTF CORE MAP
COMPLETION OF 1st TRISECTOR



HEDL 7912 014 4

Figures 5 through 10 show normalized inverse count rates that were obtained during loading of the first trisector. Shown are the results of data obtained, along with predictions, for each of the six fission chamber monitors. Relative count rate predictions were provided for every other fuel assembly loaded, for the fourth assembly through the twenty-fourth assembly. A prediction was also provided for the twenty-seventh (last) fuel assembly loaded into the trisector. The prediction lines shown in Figures 5 through 10 are merely straight-line connections between fuel assembly values at which the predictions were made. The figures also show the count rates, from Table I, that were the inverse count rate normalization values used.

Between December 24, 1979 and January 10, 1980, final preparations were completed that enabled the FFTF to operate as a nuclear reactor facility, including containment isolation, plant protection system actuation, and insertion and withdrawal capability of the three absorber rods in Trisector #3.

Checks on the worths of the three absorber rods in Trisector #3 (rods 3, 8 and 9) were made by collecting and analyzing count data with each rod fully withdrawn, and comparing with predictions, the difference in count rate with all rods fully inserted. Acceptance criteria, which basically stated that the measured worths should equal or exceed 50% of the predicted worths, were satisfied.

The second trisector loaded with fuel was Trisector #1, adjacent to LLFM C (see Figure 3). Loading commenced on January 10, 1980, with the loading of the twenty-eighth fuel assembly into the core, and the trisector was completed on January 26, 1980, when the fiftieth fuel assembly was loaded into the core. Figure 11 shows the core arrangement after completion of fuel loading into the second trisector.

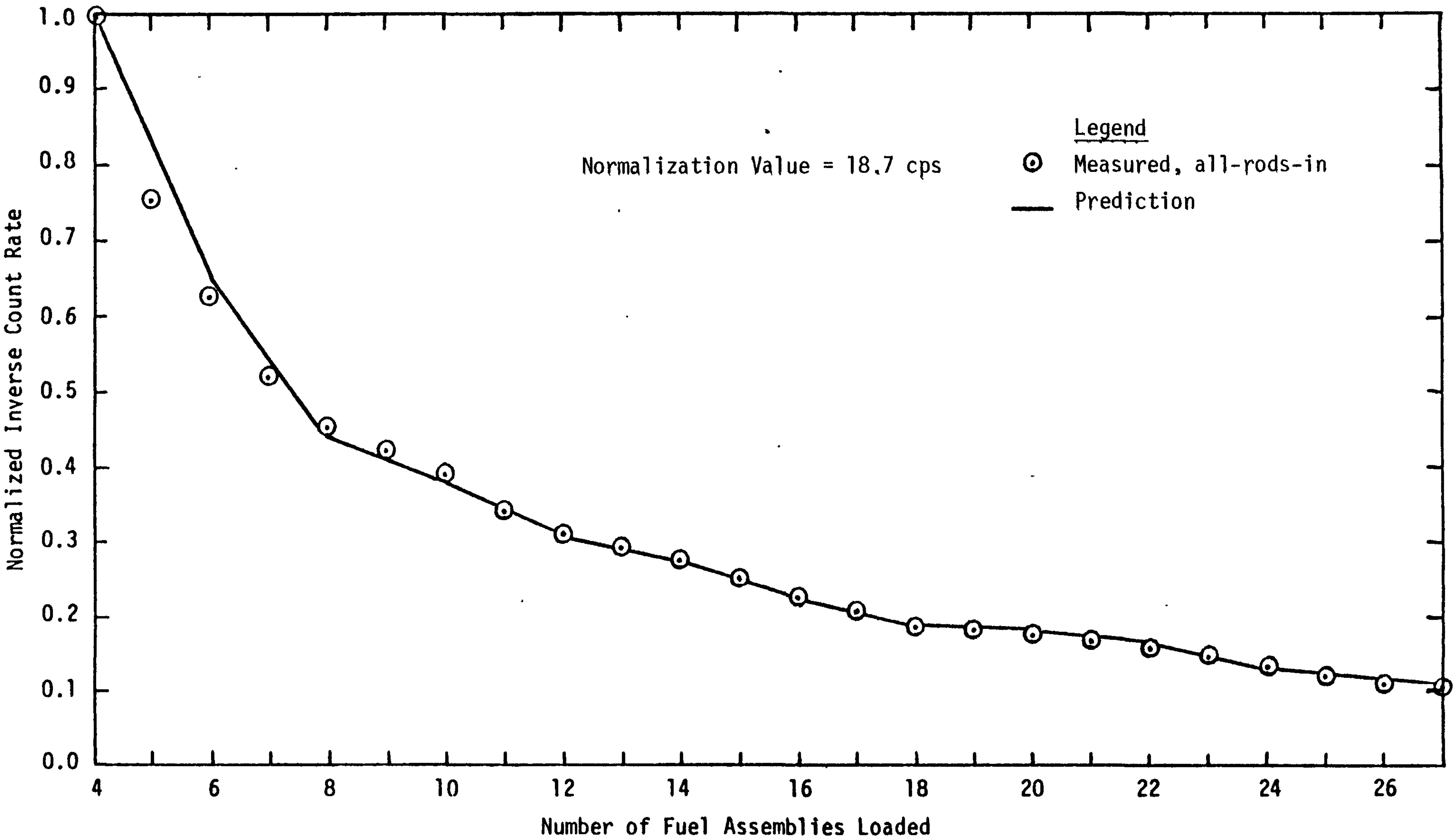


Figure 5. Inverse Count Rate vs. Fuel Assemblies Loaded in First Trisector - IRT Middle.

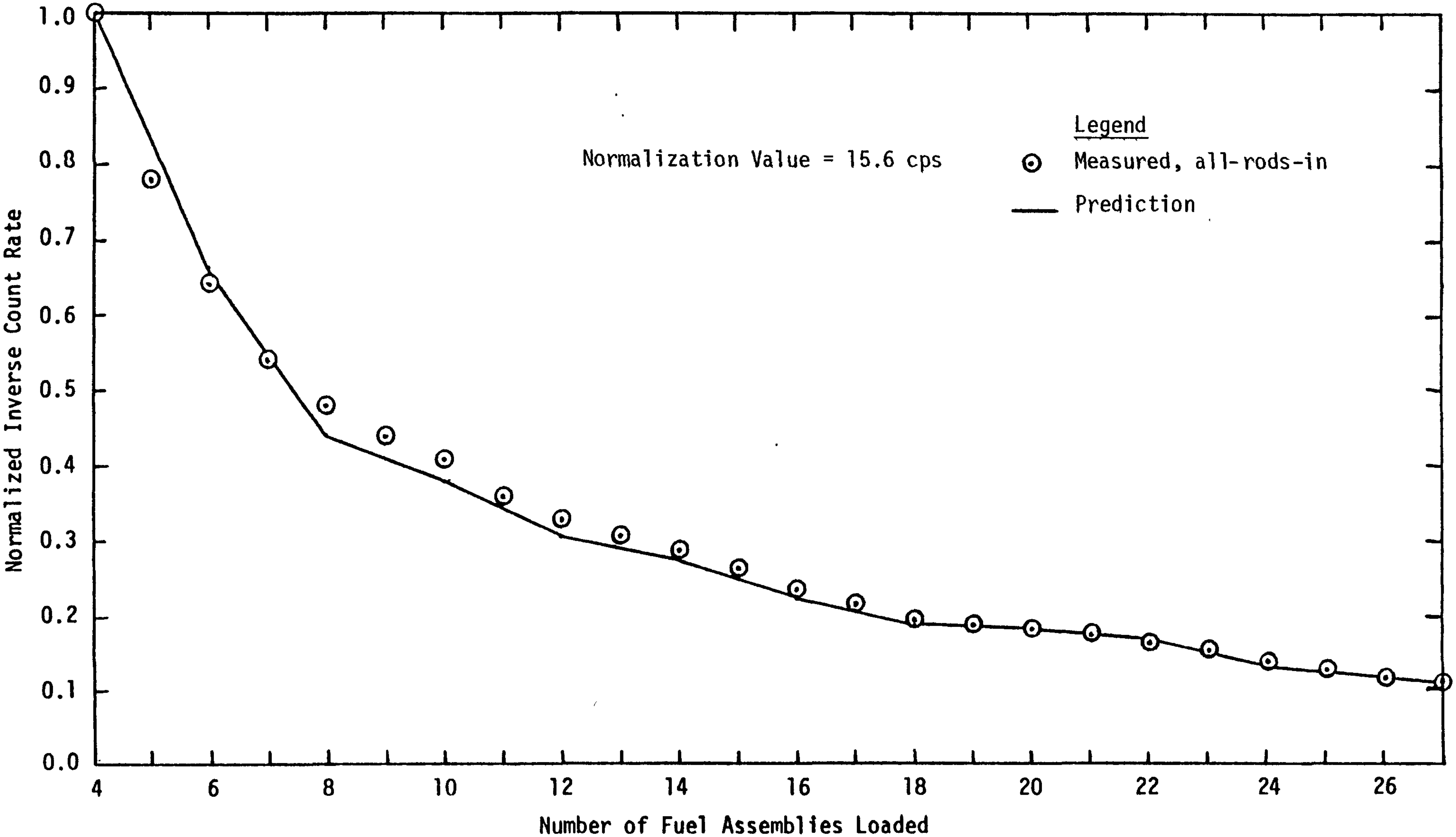


Figure 6. Inverse Count Rate vs. Fuel Assemblies Loaded in First Trisector - IRT Top.

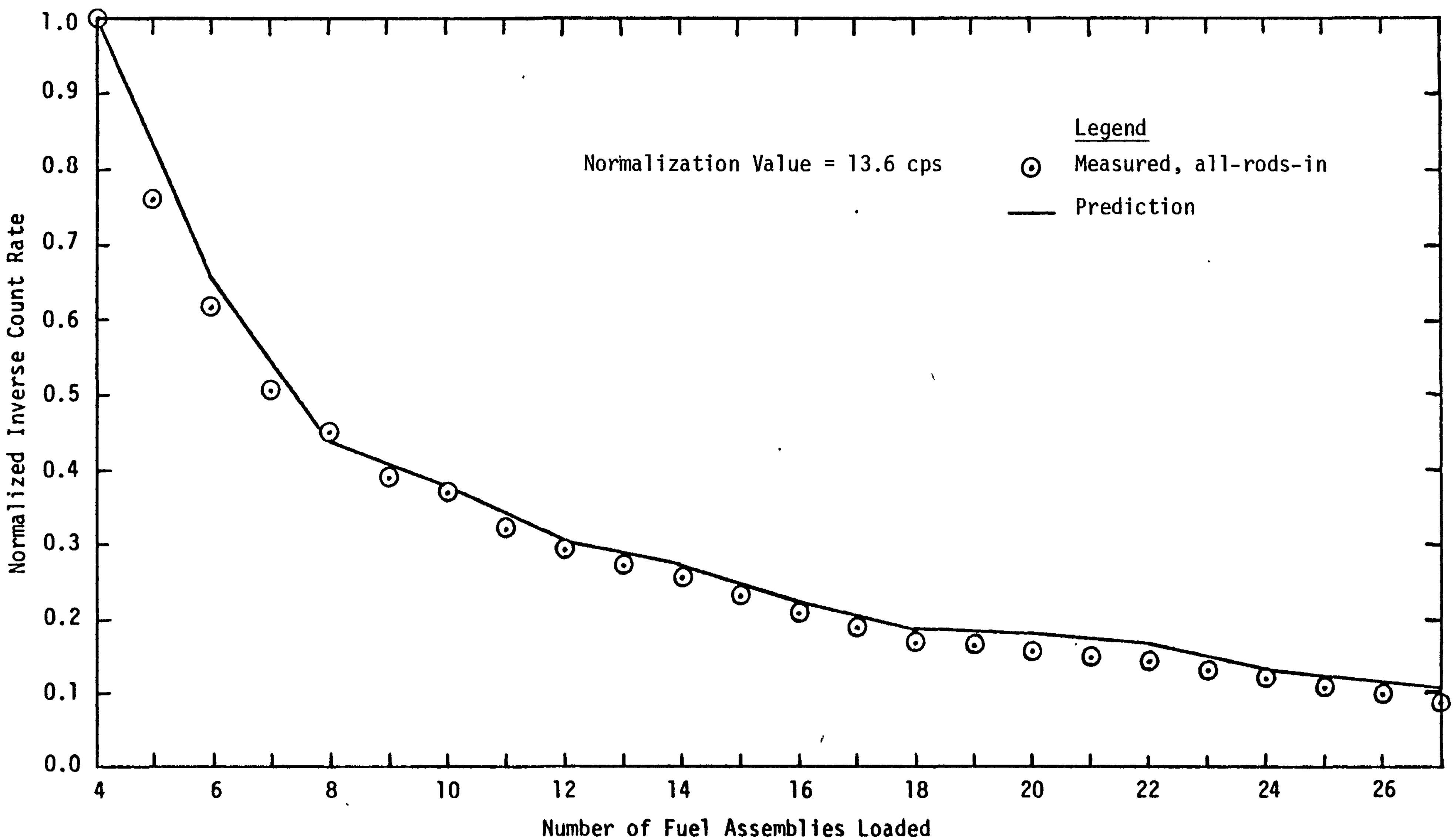


Figure 7. Inverse Count Rate vs. Fuel Assemblies Loaded in First Trisector - IRT Bottom.

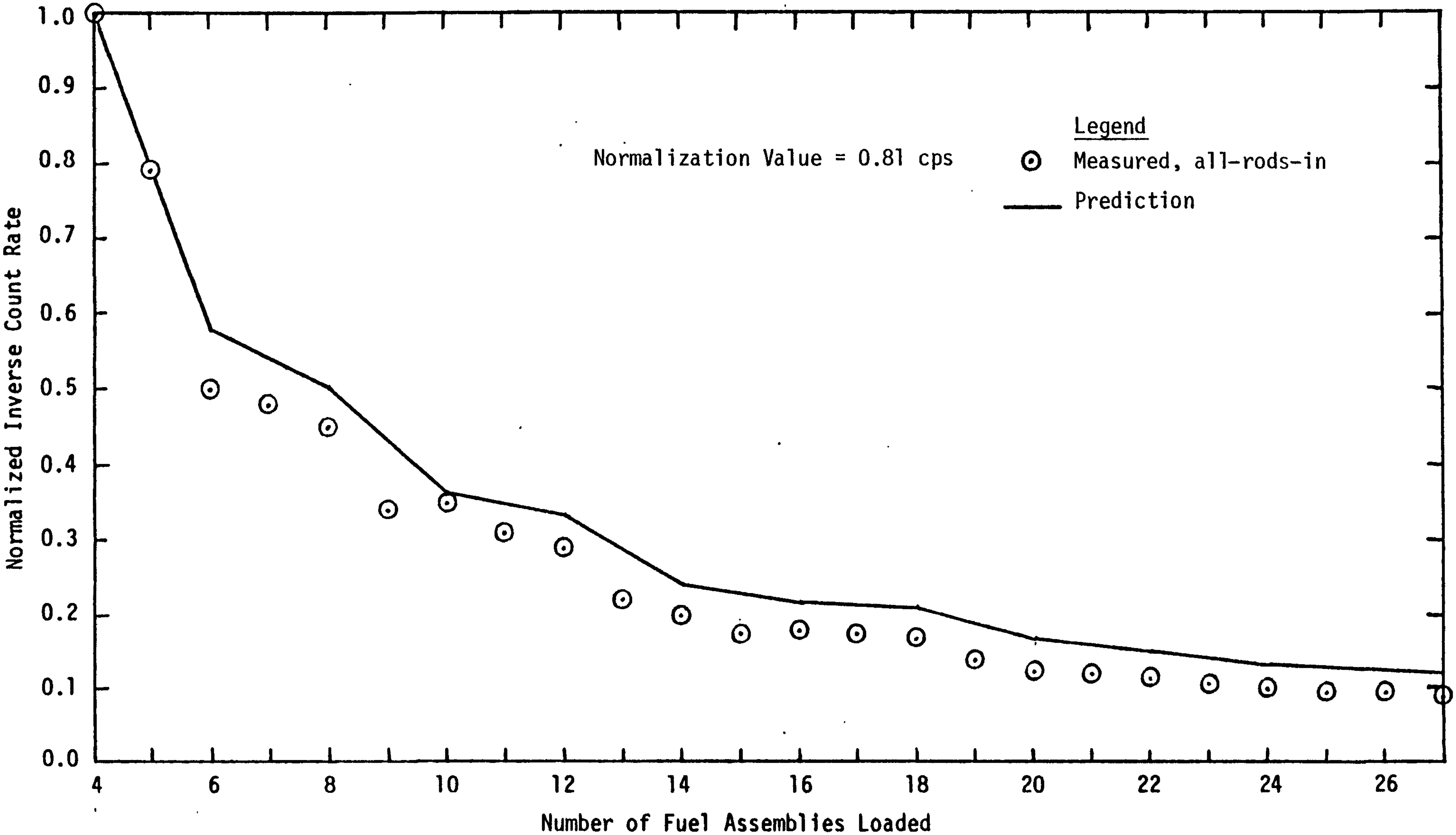


Figure 8. Inverse Count Rate vs. Fuel Assemblies Loaded in First Trisector - LLFM A.

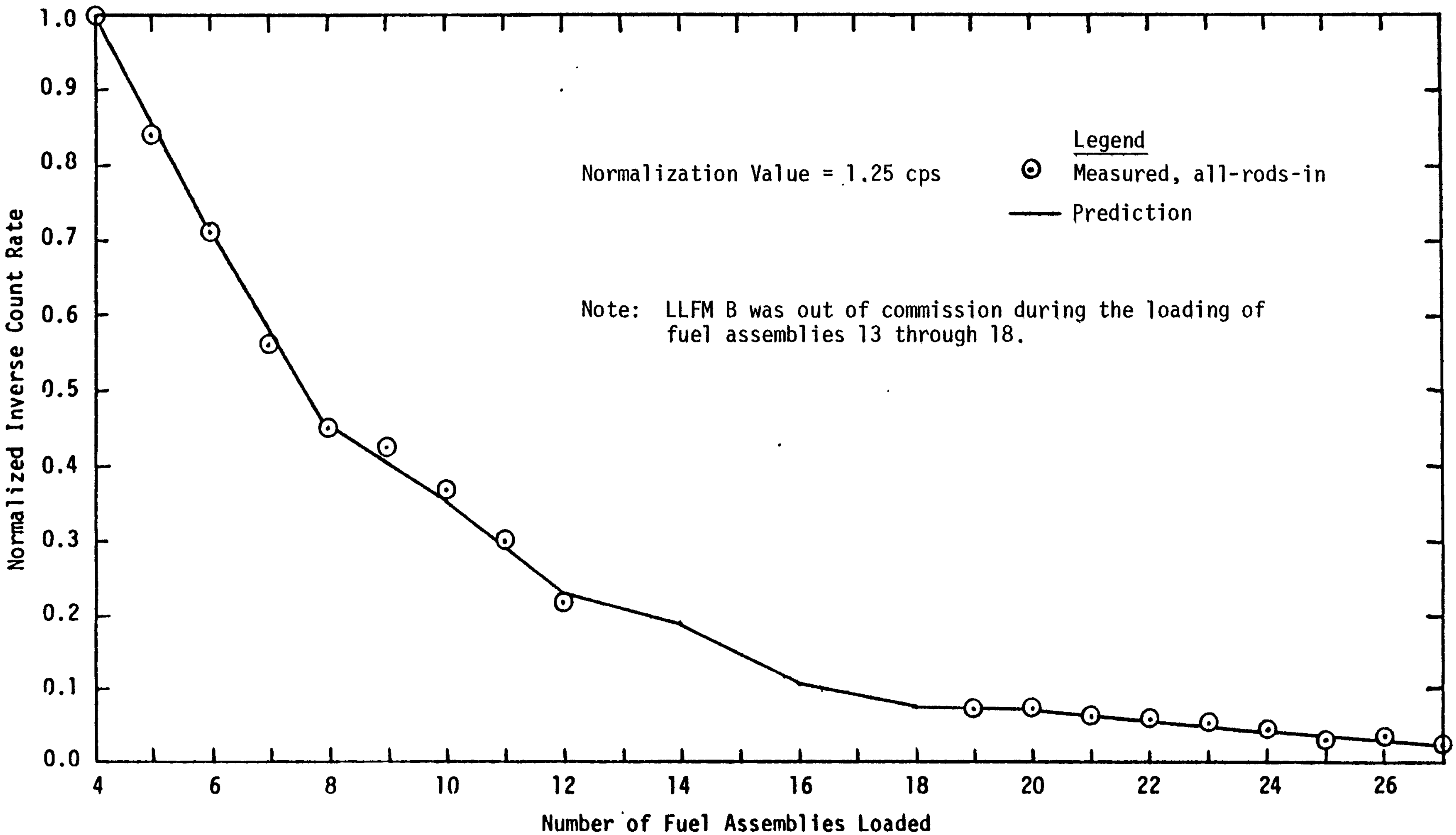


Figure 9. Inverse Count Rate vs. Fuel Assemblies Loaded in First Trisector - LLFM B.

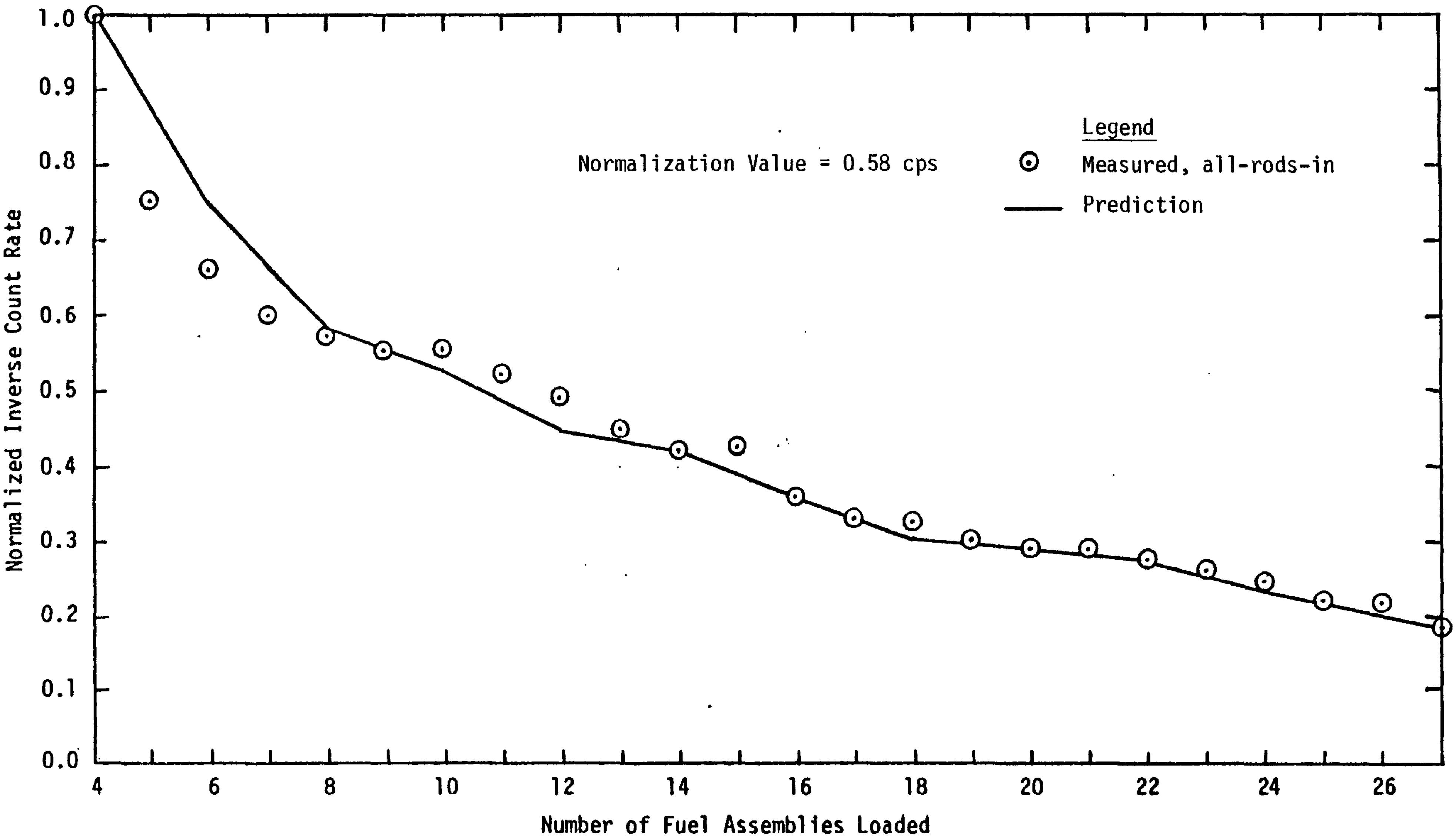
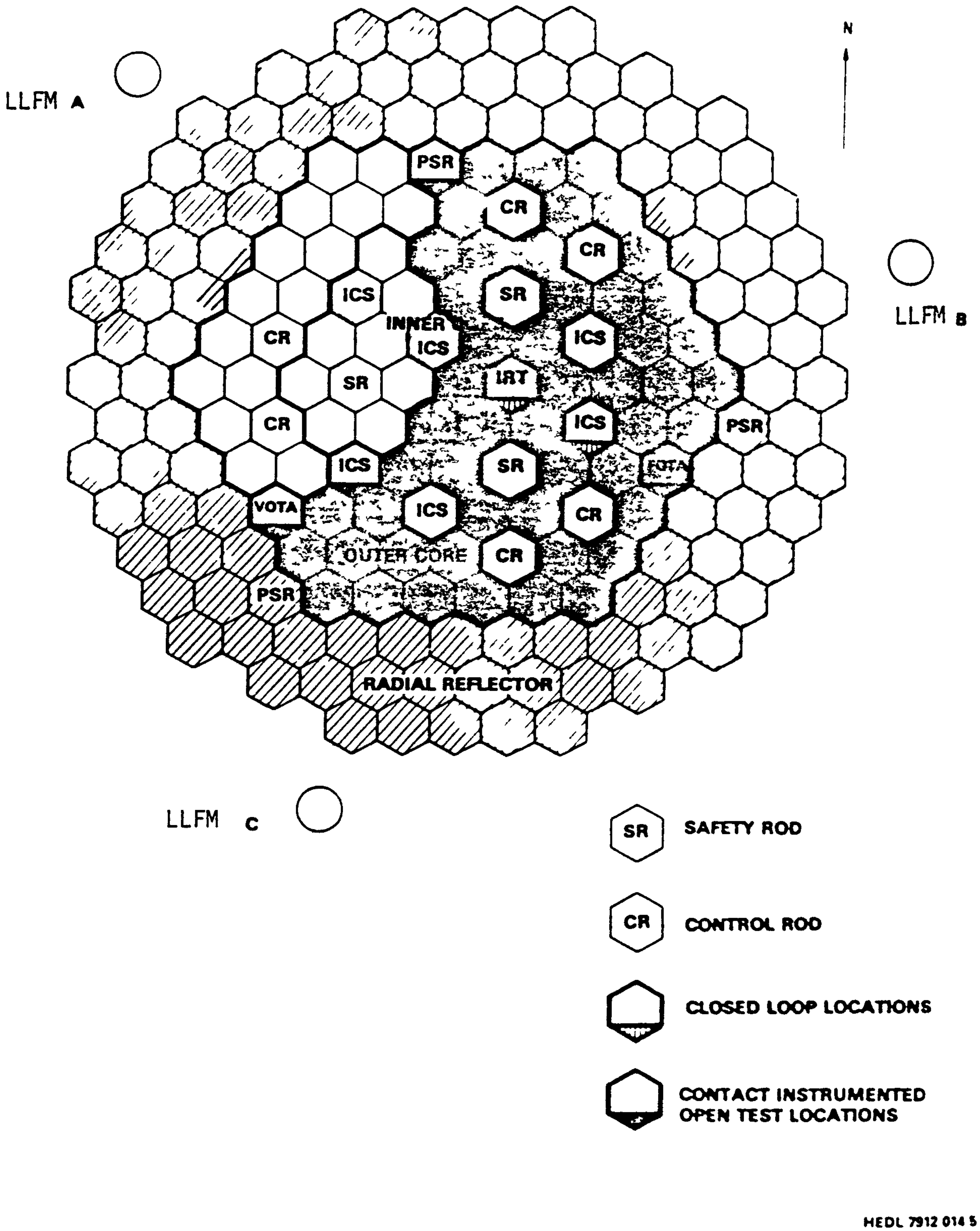


Figure 10. Inverse Count Rate vs. Fuel Assemblies Loaded in First Trisector - LLFM C.

Figure 11.

FFTF CORE MAP
COMPLETION OF 2nd TRISECTOR



Figures 12 through 17 show inverse count rates, normalized to count rates obtained after loading the twenty-ninth fuel assembly, that were obtained during the loading of the second trisector. Shown are the results of data obtained, along with predictions, for each of the six fission chamber monitors. In addition to data taken with all rods in, as was the case during the loading of the first trisector, count data were taken with primary safety rod #3 withdrawn, and with primary safety rod #3 and secondary control rods #8 and #9 withdrawn. The inverse count rates shown in Figures 12 through 17 for rods-withdrawn configurations are normalized to the all-rods-in configuration with twenty-nine fuel assemblies loaded. The measured count rates used for the normalization values are also shown in the figures. Also shown in the figures for continuity, are continuations of the data shown in Figures 5 through 10 normalized to the beginning of loading of the first trisector.

Relative count rate predictions for fuel loading into the second trisector were provided for the twenty-ninth fuel assembly (the normalization point), and for every fourth fuel assembly loaded, beginning with the thirty-second assembly. A prediction was also provided for the fiftieth (last) fuel assembly loaded into the trisector. The prediction lines shown in Figures 12 through 17 are merely straight-line connections between fuel assembly values at which the predictions were made.

After loading of the twenty-ninth fuel assembly, a fuel loading criterion was applied to data obtained following the loading of each subsequent assembly. The criterion basically stated that if two or more of the IRT monitors' experimental all-rods-in points for a given fuel assembly fell below the inverse count rate criterion line (see Figures 12, 13 and 14) fuel loading would be suspended pending the outcome of an Engineering/ Safety review and approval of a course of action. The criterion line was conceived to assure that the core reactivity would not be increased so that criticality could be achieved with all rods out after the addition of one more fuel assembly. As shown in Figures 12, 13 and 14, the criterion was never violated.

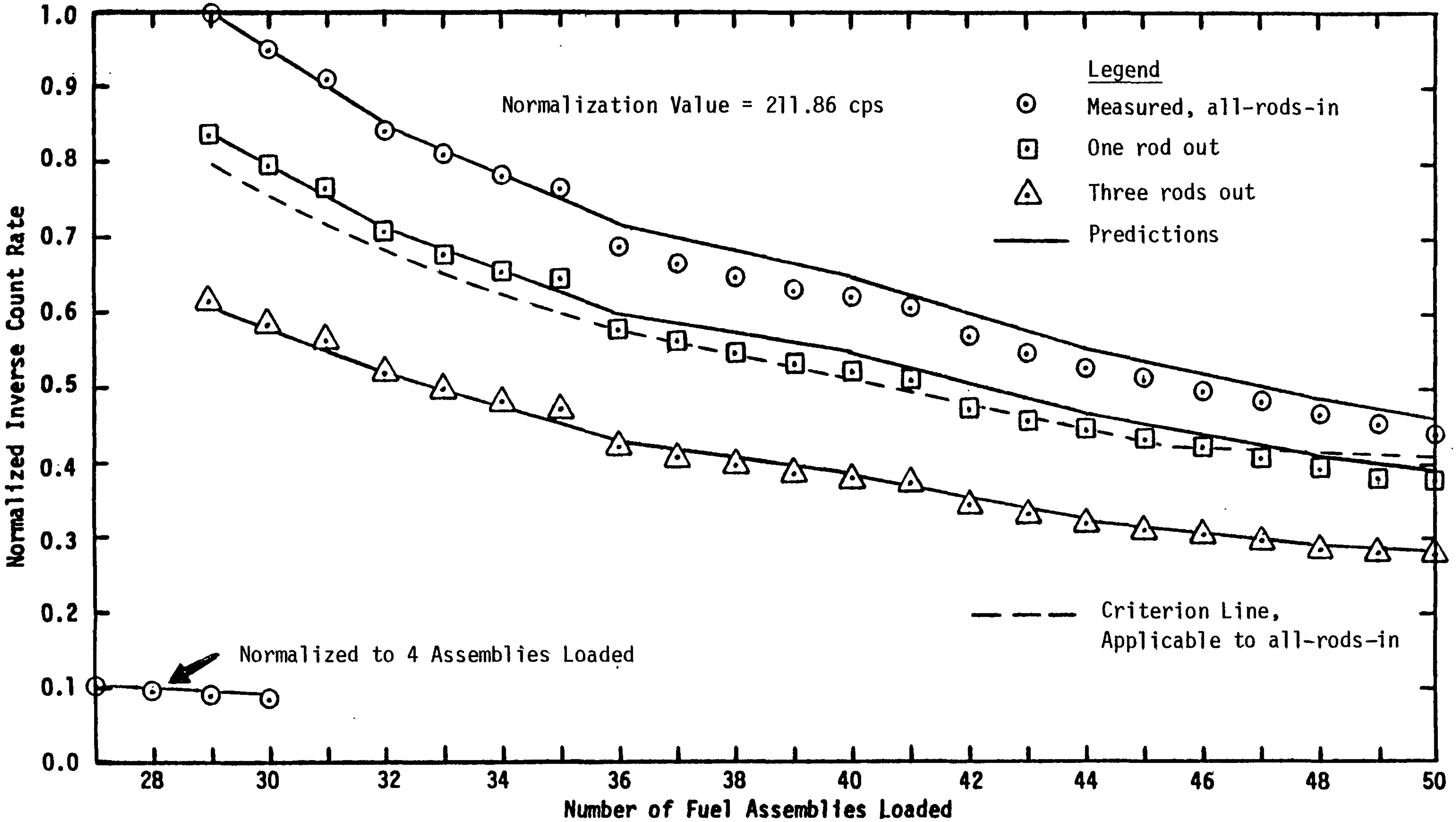


Figure 12. Inverse Count Rate vs. Fuel Assemblies Loaded in Second Trisector - IRT Middle.

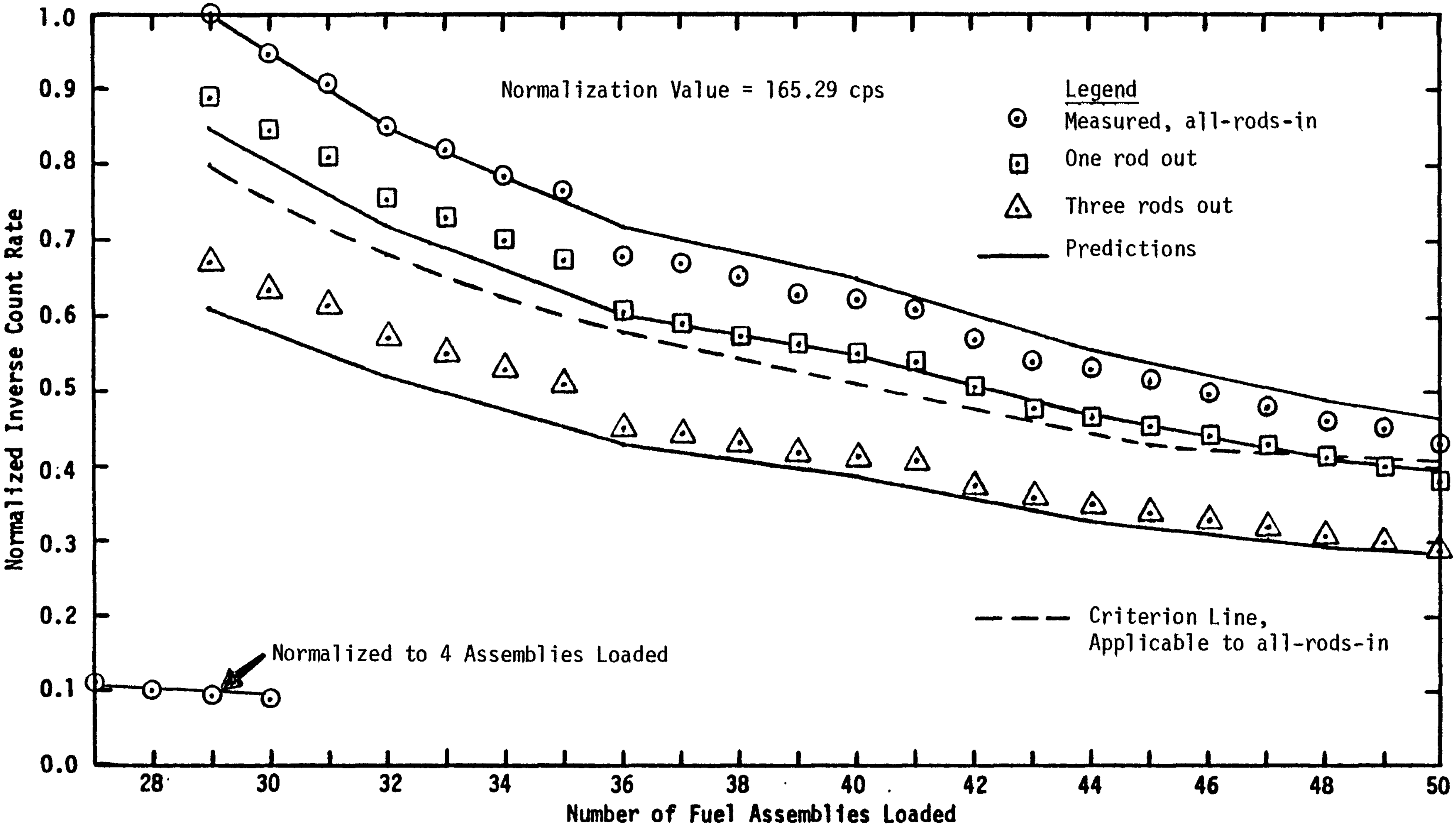


Figure 13. Inverse Count Rate vs. Fuel Assemblies Loaded in Second Trisector - IRT Top.

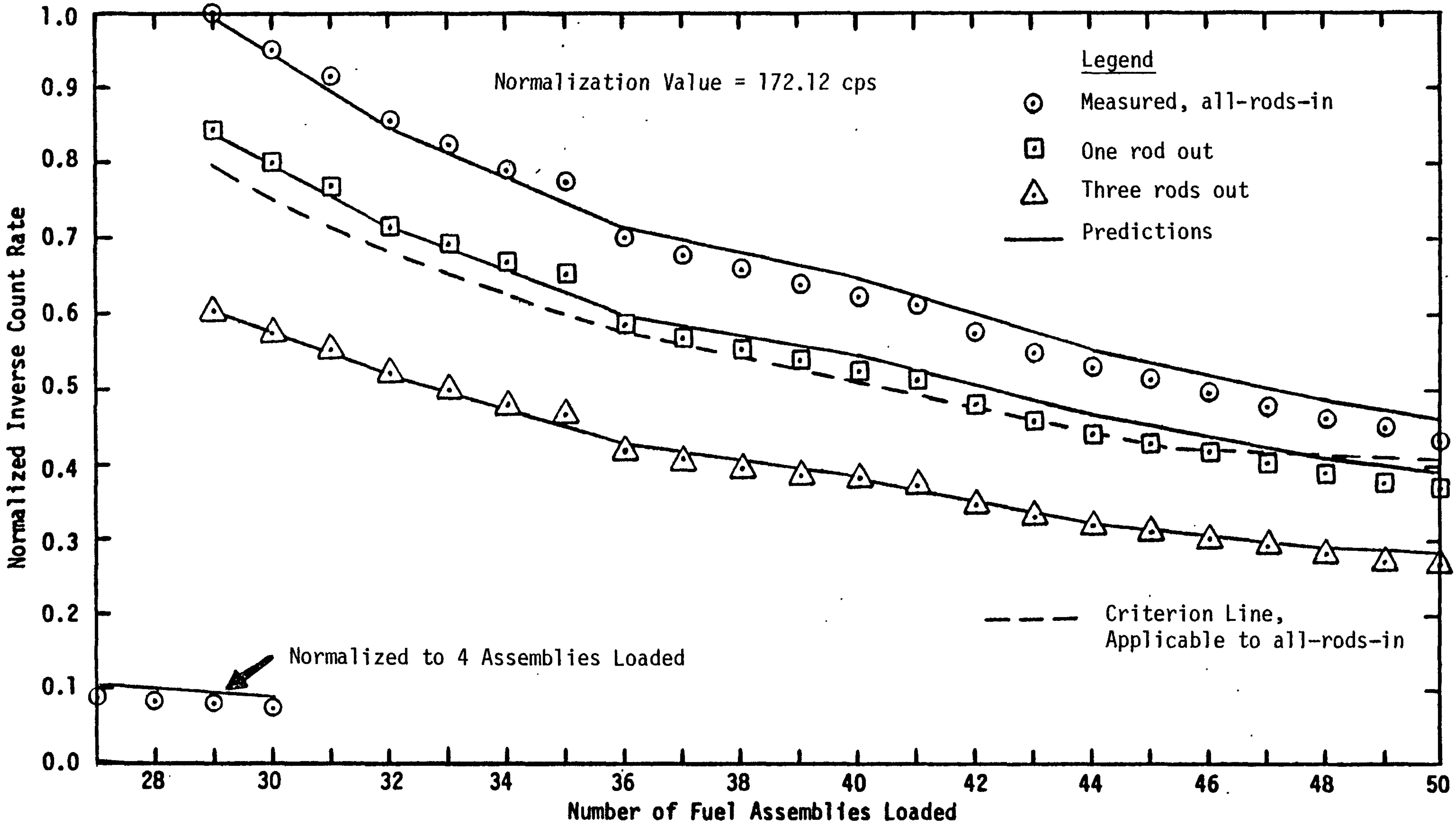


Figure 14. Inverse Count Rate vs. Fuel Assemblies Loaded in Second Trisector - IRT Bottom.

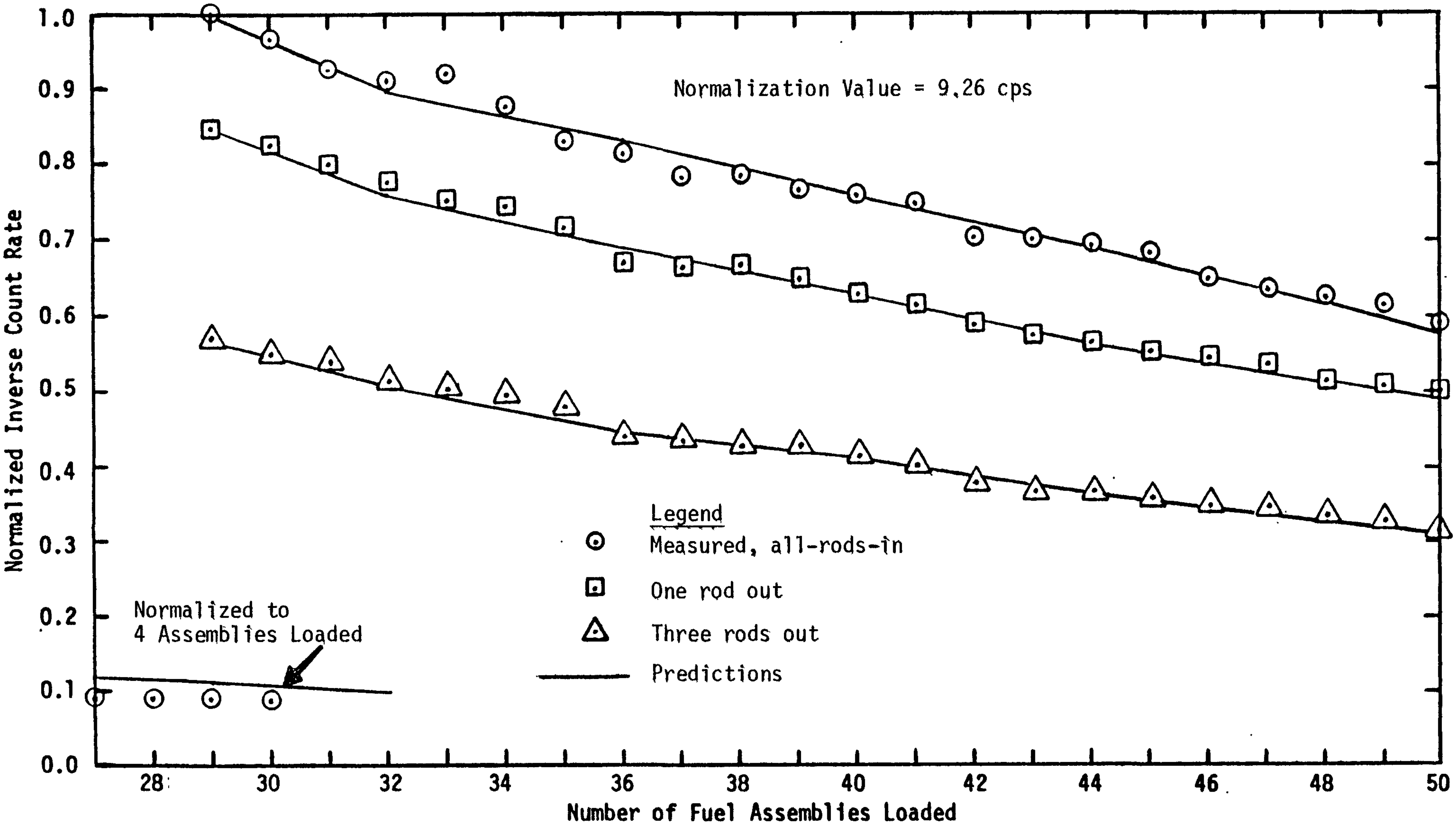


Figure 15. Inverse Count Rate vs. Fuel Assemblies Loaded in Second Trisector - LLFM A.

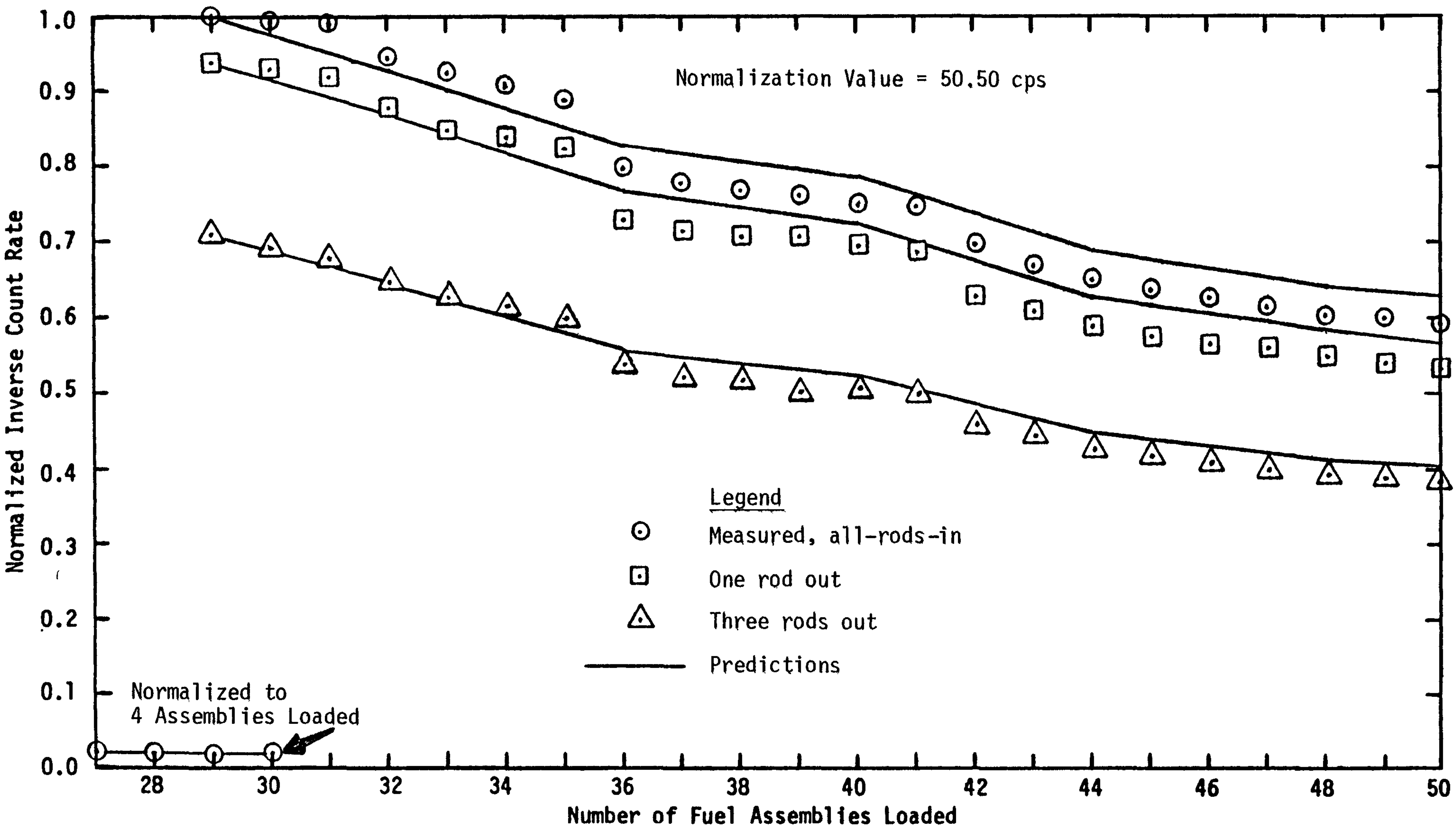


Figure 16. Inverse Count Rate vs. Fuel Assemblies Loaded in Second Trisector - LLFM B.

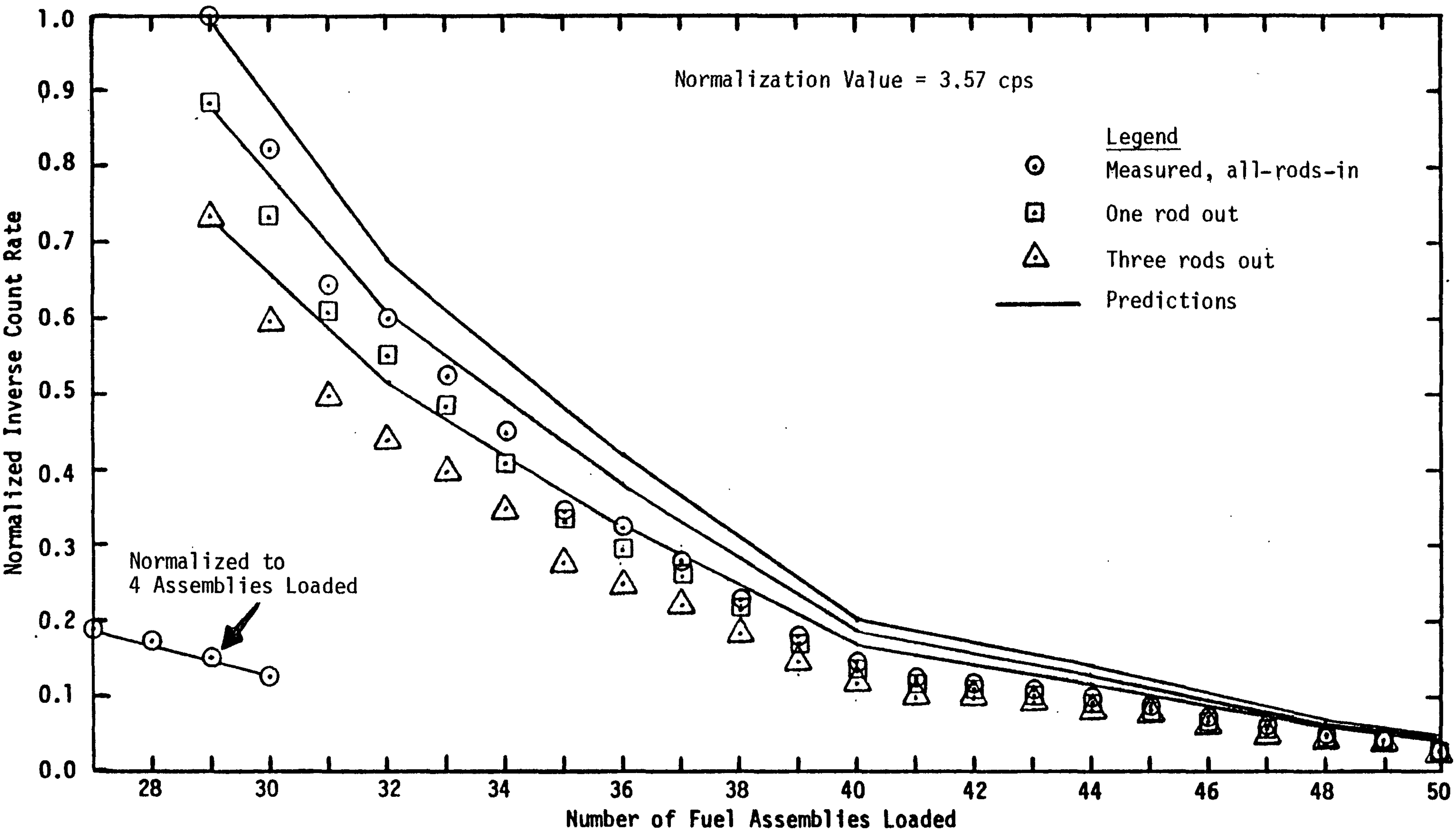


Figure 17. Inverse Count Rate vs. Fuel Assemblies Loaded in Second Trisector - LLFM C.

Following the loading of the second trisector (Trisector #1), checks on the worths of the three absorber rods in Trisector #1 (rods 1, 4 and 5) were made by collecting and analyzing count data with each rod fully withdrawn, and comparing with predictions, the difference in count rate with the rods fully inserted. Acceptance criteria, the same as for Trisector #3 rods, were satisfied.

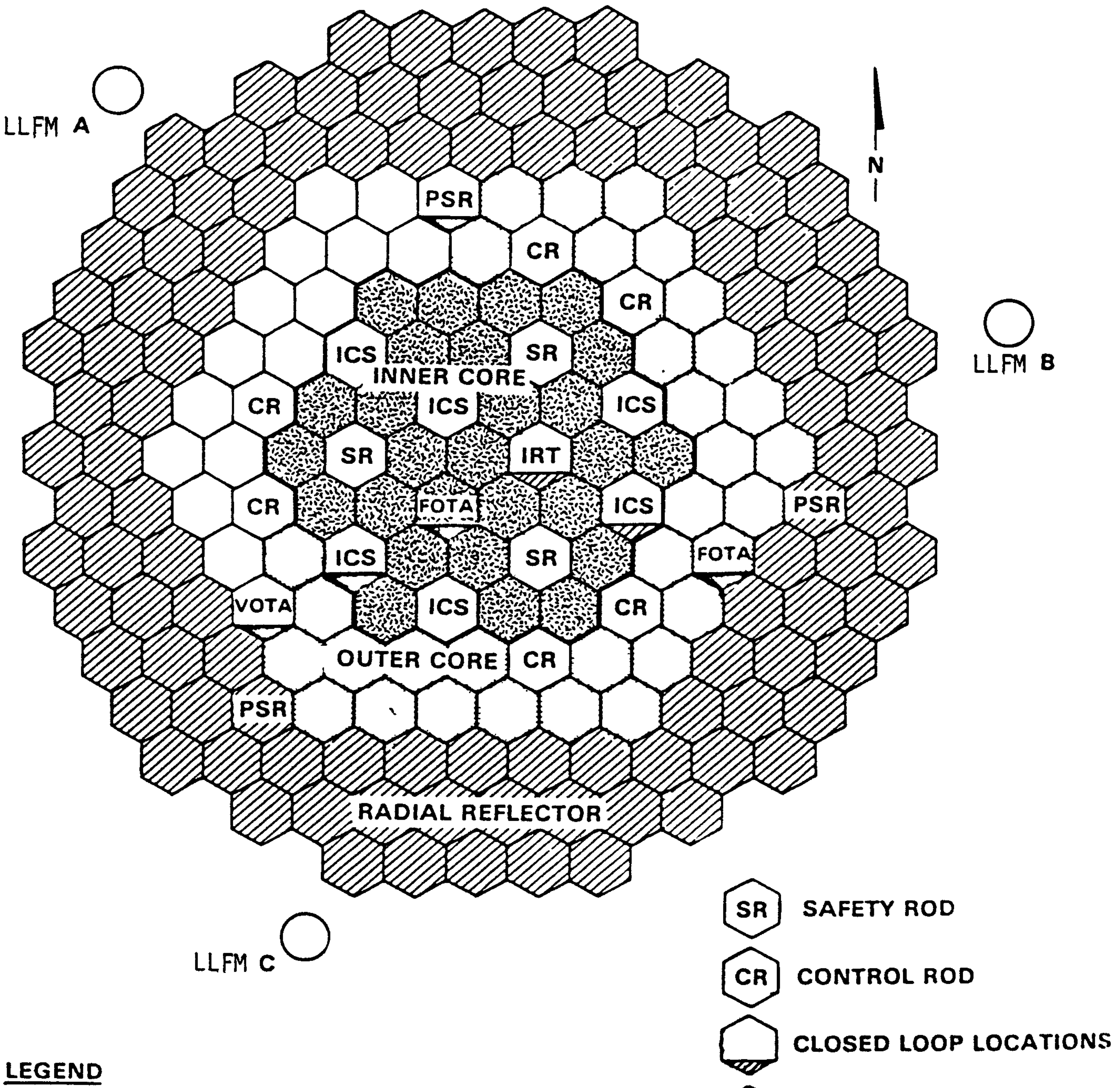
The third and final trisector loaded with fuel was Trisector #2, adjacent to LLFM A (see Figure 3). Loading of the final trisector commenced on January 30, 1980, with the loading of the fifty-first fuel assembly into the core. Fuel loading was halted temporarily on February 3, 1980, after the fifty-ninth fuel assembly was loaded into the core, to prepare for the initial criticality attempt. Figure 18 shows the core arrangement for the initial criticality attempt.

Figures 19 through 24 show inverse count rates, normalized to count rates obtained after loading the fiftieth fuel assembly, that were obtained during the loading of the third trisector. Shown are data for all rods in, one rod out (rod 3), three rods out (rods 1, 4 and 5) and six rods out (rods 1, 3, 4, 5, 8 and 9). All data are normalized to the all-rods-in data for fifty fuel assemblies loaded. The count rates used for normalization are shown in the figures.

Relative count rate predictions for fuel loading into the final trisector were provided for the fiftieth fuel assembly (the normalization point), and for selected fuel assembly additions and rod configurations. The prediction lines shown in Figures 19 through 24 are straight-line connections between fuel assembly values at which predictions were made. Three rods out and six rods out predictions were not made beyond fifty-eight fuel assemblies loaded, the calculated nominal loading for all rods out criticality.

Figure 18.

FFTF CORE MAP
INITIAL CRITICAL CONFIGURATION



LEGEND

PSR - PERIPHERAL SHIM ROD
ICS - IN-CORE SHIM ASSEMBLY
FOTA - FUEL OPEN TEST ASSEMBLY
VOTA - VIBRATION OPEN TEST ASSEMBLY
A, B, C - LOCATIONS OF LOW LEVEL FLUX MONITORS (FISSION CHAMBERS)
IRT - IN-REACTOR THIMBLE (CONTAINS THREE FISSION CHAMBERS)

HEDL 8002 [illegible]

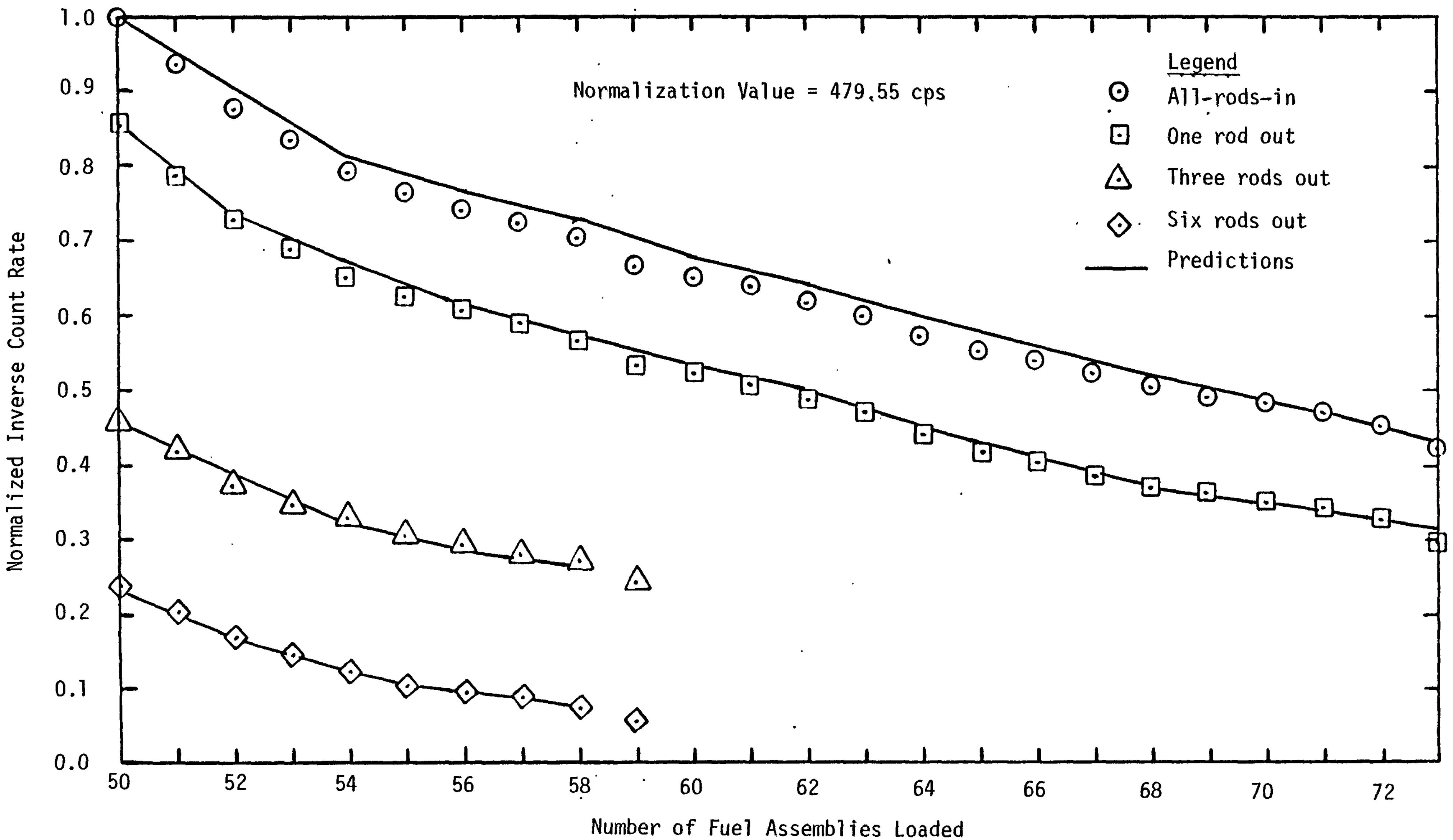


Figure 19. Inverse Count Rate vs. Fuel Assemblies Loaded in Third Trisector - IRT Middle.

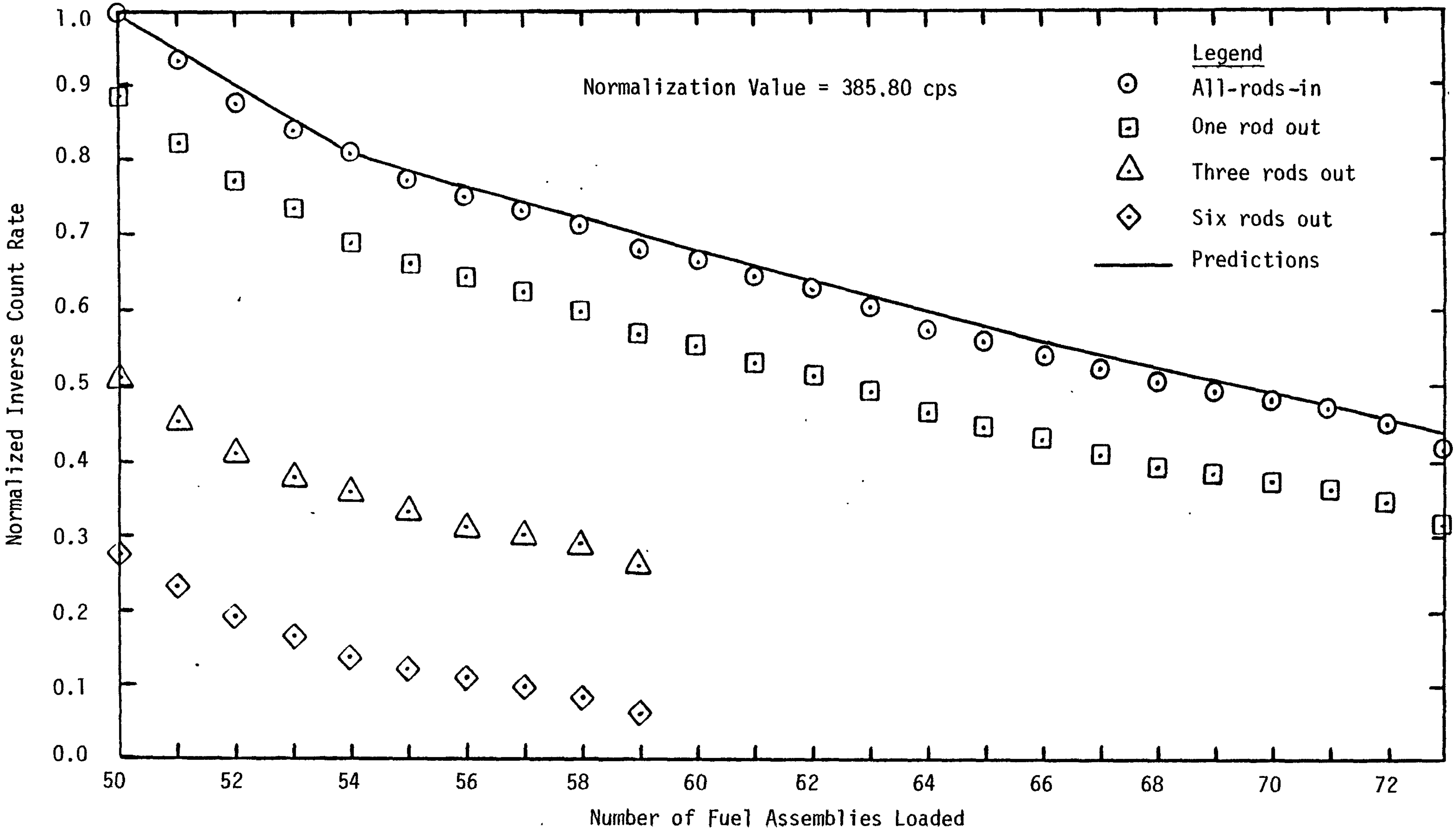


Figure 20. Inverse Count Rate vs. Fuel Assemblies Loaded in Third Trisector - IRT Top.

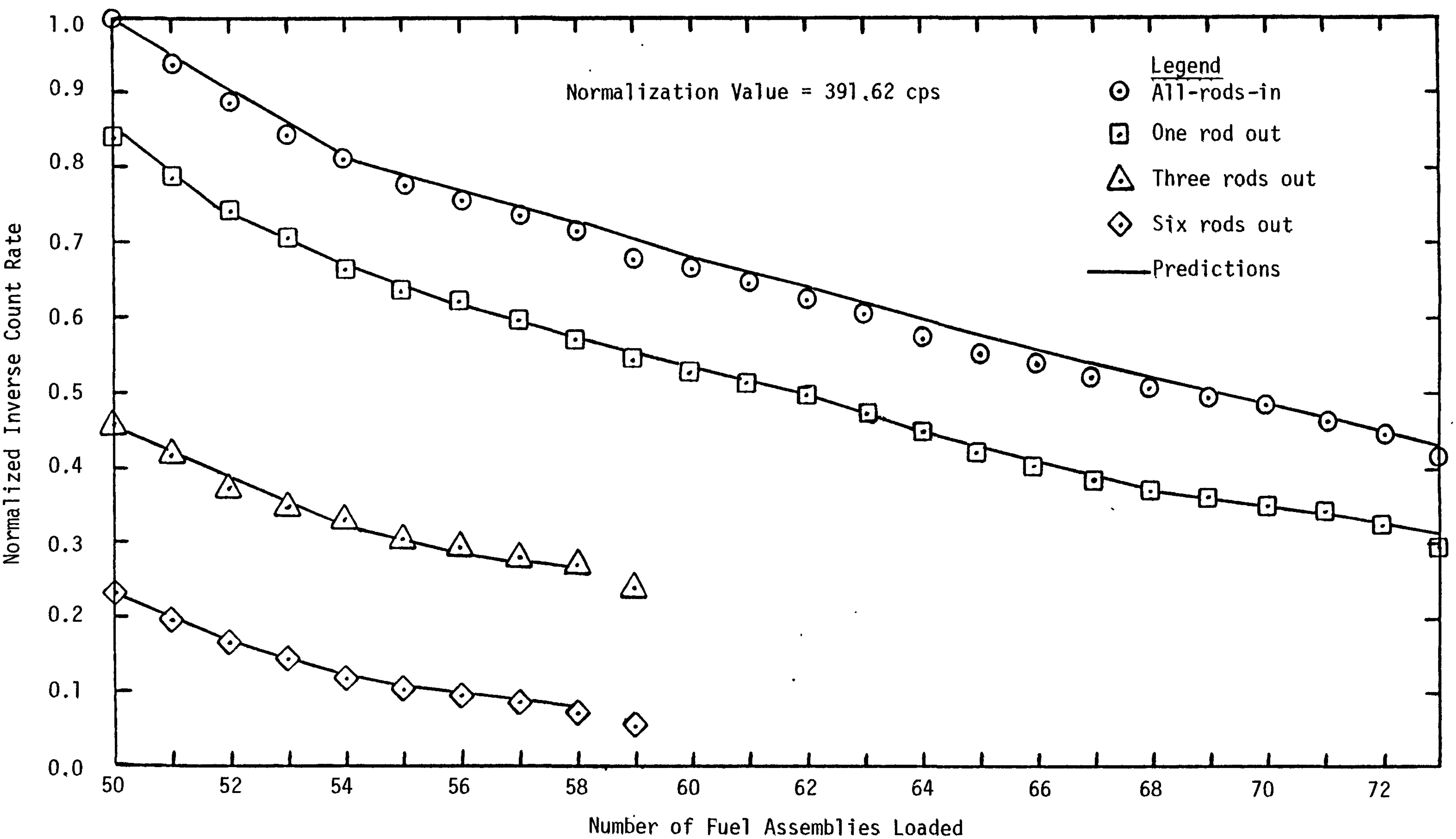


Figure 21. Inverse Count Rate vs. Fuel Assemblies Loaded in Third Trisector - IRT Bottom.

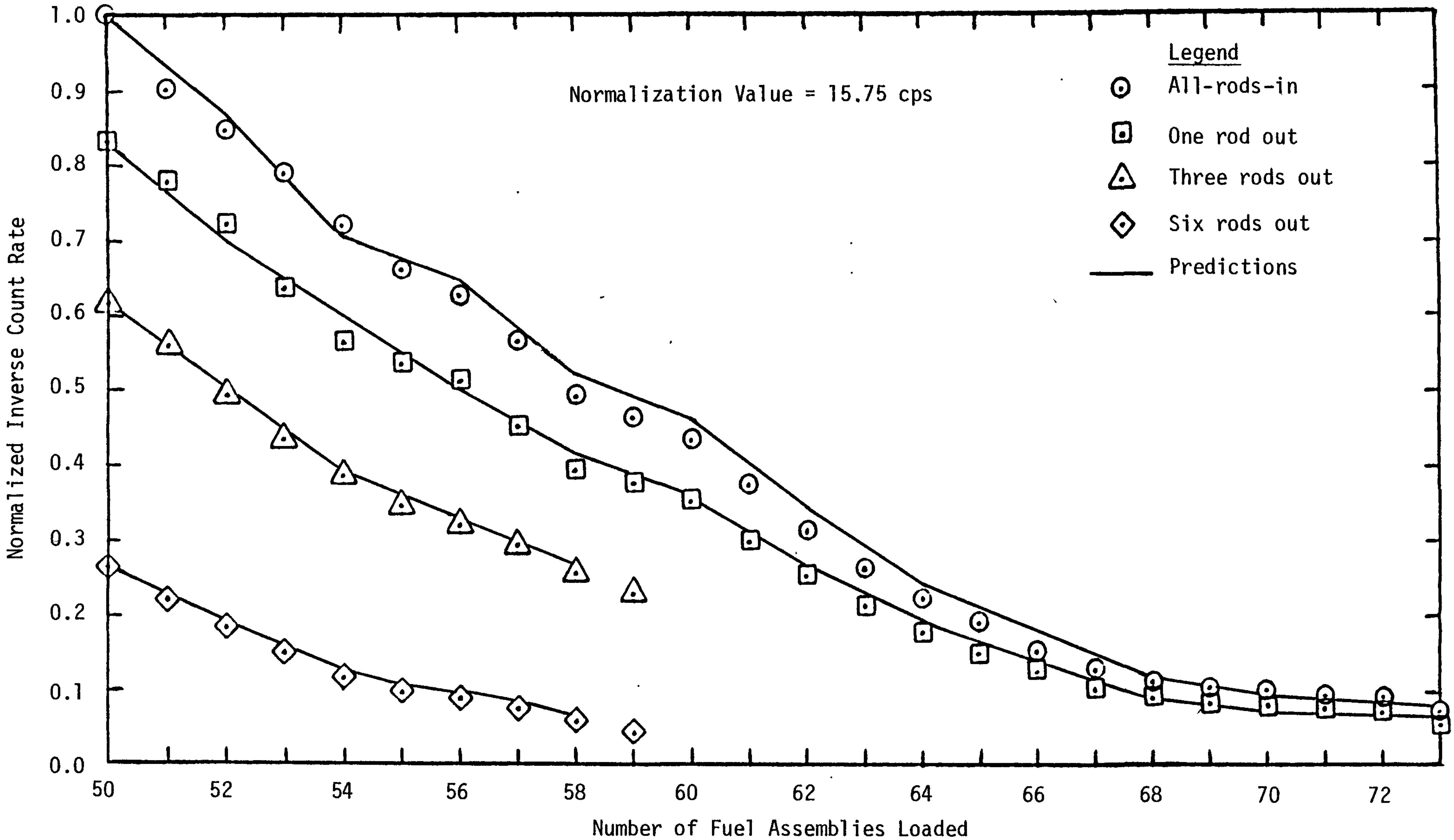


Figure 22. Inverse Count Rate vs. Fuel Assemblies Loaded in Third Trisector - LLFM A.

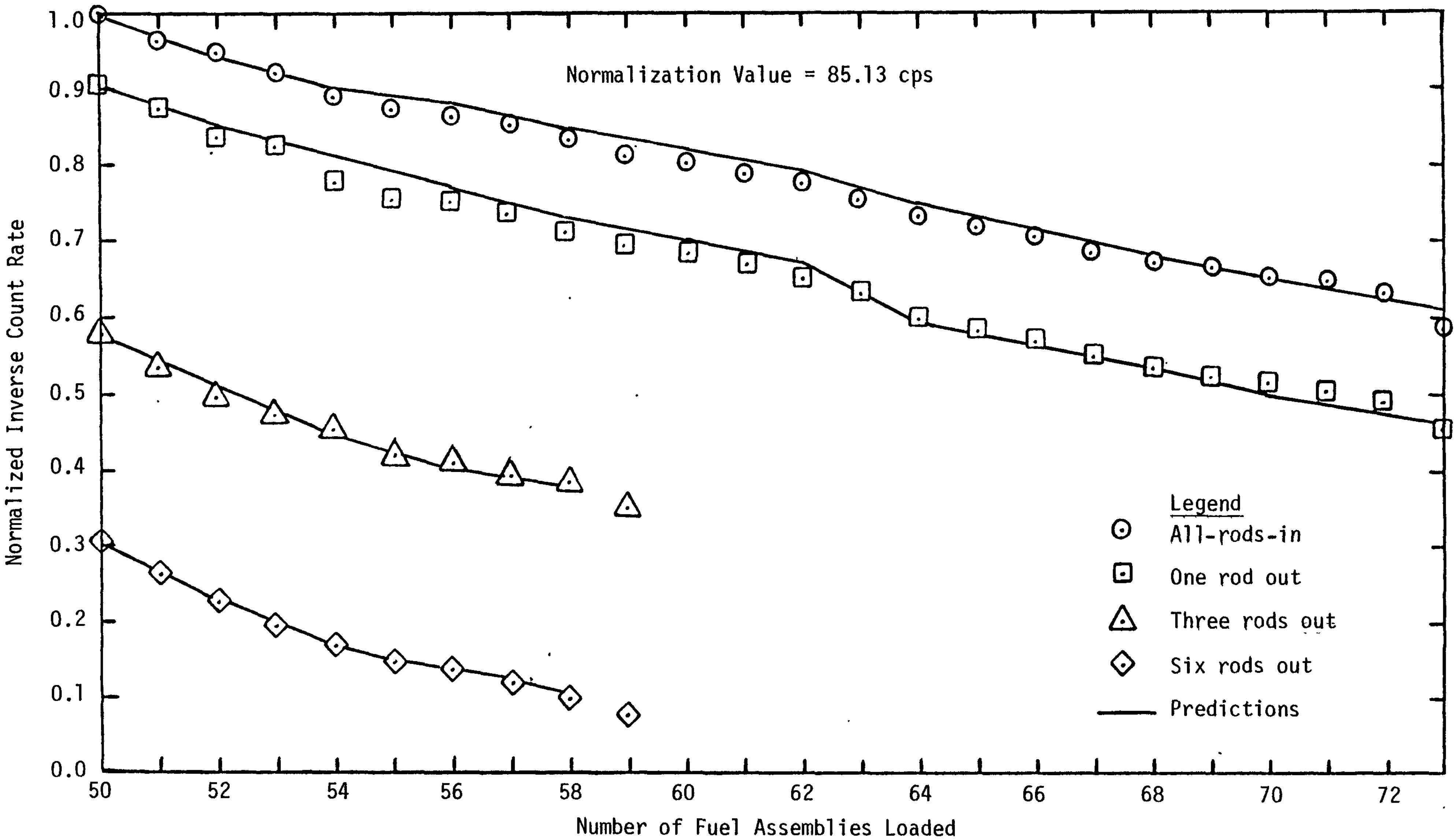


Figure 23. Inverse Count Rate vs. Fuel Assemblies Loaded in Third Trisector - LLFM B.

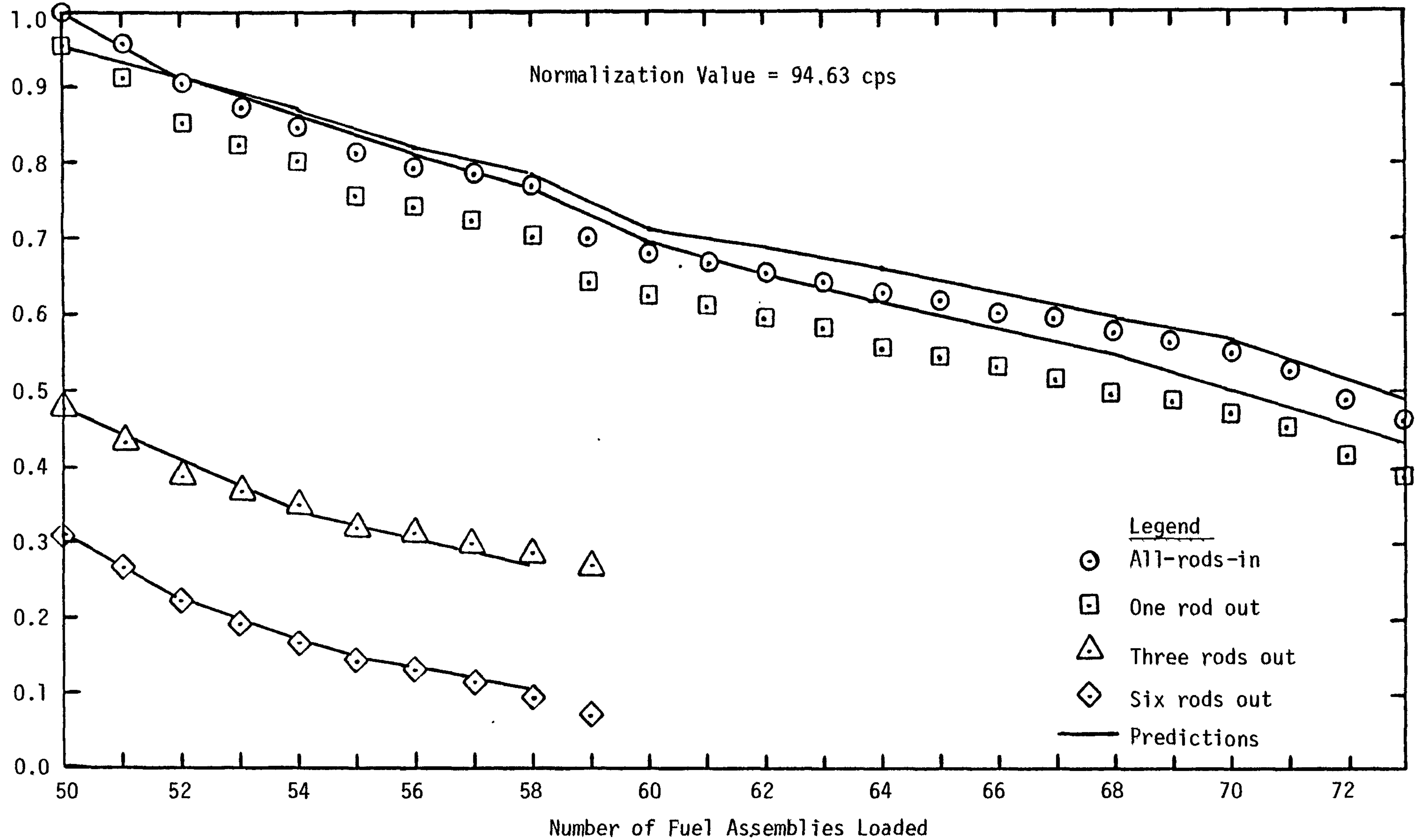


Figure 24. Inverse Count Rate vs. Fuel Assemblies Loaded in Third Trisector - LLFM C.

During the loading of the third trisector, a second criterion was applied. This criterion basically stated that if at least two of the three IRT monitors' measured data fell below the "Criticality Estimation Curve," it was expected that criticality could be achieved by withdrawing all nine absorber rods. The criticality estimation curve was based on the calculated worth of the three absorber rods in the final trisector as a function of the number of fuel assemblies loaded. The criterion was applied to six-rods-out normalized inverse count rate as measured by the IRT monitors.

Figure 25 shows the six-rods-out inverse count rate relationship with the criticality estimation curve. During the loading of the third trisector the criticality estimation curve was revised as shown in Figure 25, based on the experience gained from measured rod worths in the other two trisectors and the difference between the measured and predicted six-rods-out inverse count rate. The intersection of the measurement data curve with the criticality estimation curve represents the fuel loading at which there would be a 50-50 chance of achieving criticality with all nine rods out.

After the fifty-sixth fuel assembly was loaded, extrapolation of the measured data showed that the criticality estimation curve would be intersected following loading of one of the next two fuel assemblies. The decision was made to load the next (#57) fuel assembly, and, if the resulting normalized inverse count rate was equal to or greater than 0.34, the fifty-eighth fuel assembly was also to be loaded into the core. As shown in Figure 25, the measured value was 0.35. After the fifty-eighth fuel assembly was loaded, it was decided to proceed with the loading of the fifty-ninth assembly, after which criticality was to be attempted.

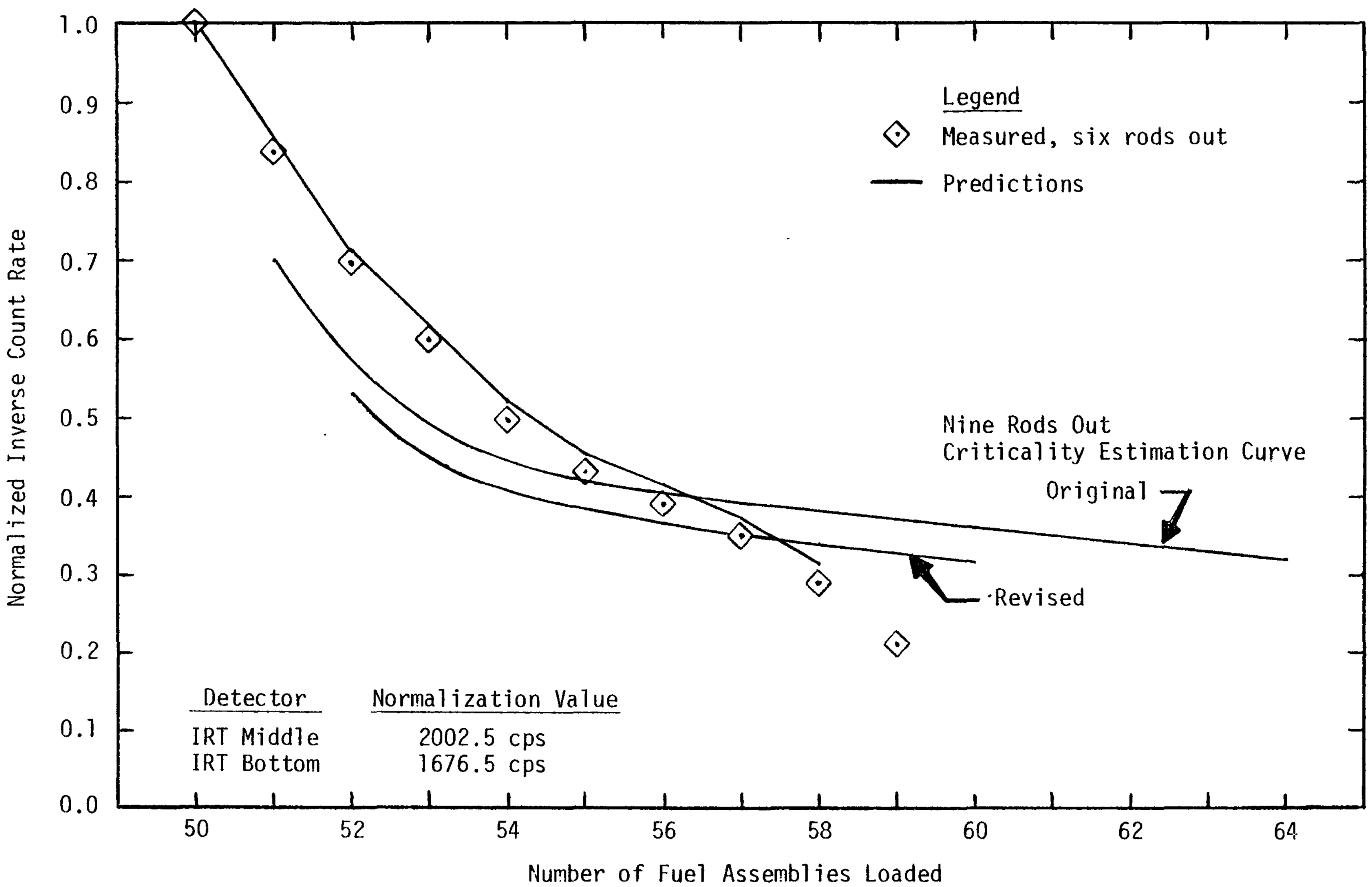


Figure 25. Inverse Count Rate vs. Fuel Assemblies Loaded - Criticality Estimation.

On February 3, 1980, after fifty-nine fuel assemblies had been installed in the core, further fuel loading was suspended and the third instrument tree, for Trisector #2, was rotated and lowered over the core. The drive-lines for the last three absorber assemblies (rods 2, 6 and 7) were connected and the fuel transfer ports through the reactor vessel head were closed and sealed. All mechanical actions needed for the initial approach to criticality were completed by February 6, 1980.

On February 9, 1980, the rod withdrawals to achieve criticality began. The primary safety rods (rods 1, 2 and 3) were fully withdrawn and count data were recorded that provided the normalization point for the inverse count rate as a function of secondary rod bank height. At 9:30 a.m. the plot of inverse count rate as a function of secondary rod bank height had begun. The count rates that were recorded with the primary safety rods fully withdrawn and the secondary control rods fully inserted are given in Table II. Uncertainties are 1σ from counting statistics.

TABLE II

INITIAL CRITICALITY COUNT RATE NORMALIZATION DATA

Count Rate (cps) With 59 Fuel Assemblies Loaded and Primary Rods Out

LLFM			IRT		
A	B	C	Top	Middle	Bottom
65.9±0.4	173.4±0.7	224.9±0.9	1062±2	1469±2	1193±2

The secondary rods (rods 4, 5, 6, 7, 8 and 9) were then withdrawn, one at a time, a preplanned distance or until one or more of the observed count rates doubled. Rod pulls were made in three-inch increments until a bank height of twenty-seven inches from full insertion was achieved. At each three-inch increment in secondary rod bank height, count data were recorded and plotted. The secondary rods were then pulled to bank heights of 29", 30", 30.6" and 31", with count data taken at each bank height. By 2:30 p.m., control rod bank withdrawal had proceeded to the height of thirty-one inches, the height from which the next rod bank movement was expected to achieve criticality. Figure 26 shows the inverse count rate as a function of secondary rod bank height. The extrapolated critical rod bank height, shown in Figure 26, was 31.3".

For efficiency purposes it was decided not to pull the secondary rods further as a bank to achieve criticality, but to pull one rod from the bank. Rod 4 was selected and at 3:45 p.m. was pulled from the bank sufficient to achieve initial criticality at a startup rate of approximately 0.9 decade per minute. Startup was terminated by reinsertion of rod 4 when approximately 1 kW of fission power was reached. Figure 27 is a reproduction of the actual strip chart recording of LLFM count rate during the achievement of initial criticality.

Subsequent to initial criticality, Trisector #2 was reopened and fuel loading continued to complete the loadout of the core. The sixtieth fuel assembly was inserted into the core on February 12, 1980, and core loading was completed by inserting the seventy-third fuel assembly into the core on February 19, 1980.

Figure 28 shows the final, fully loaded core arrangement. During the loading of the last fourteen fuel assemblies, count rate data were recorded for two rod configurations; all-rods-in and rod 3 out. The inverse count rates obtained, along with predictions, are shown in Figures 19 through 24.

During final loadout of the core the secondary system shutdown margin was also estimated, and a criterion for continuation of fuel loading based on margin assessment was invoked. The margin value was to remain negative at all times. Consequently, the criterion stated that if the margin value became positive based on data from two out of three LLFMs, the Engineering/ Safety review committee, with DOE approval, would determine the course of action. Figures 29, 30 and 31 show the margin values that were recorded. As shown in the figures, the criterion was not violated.

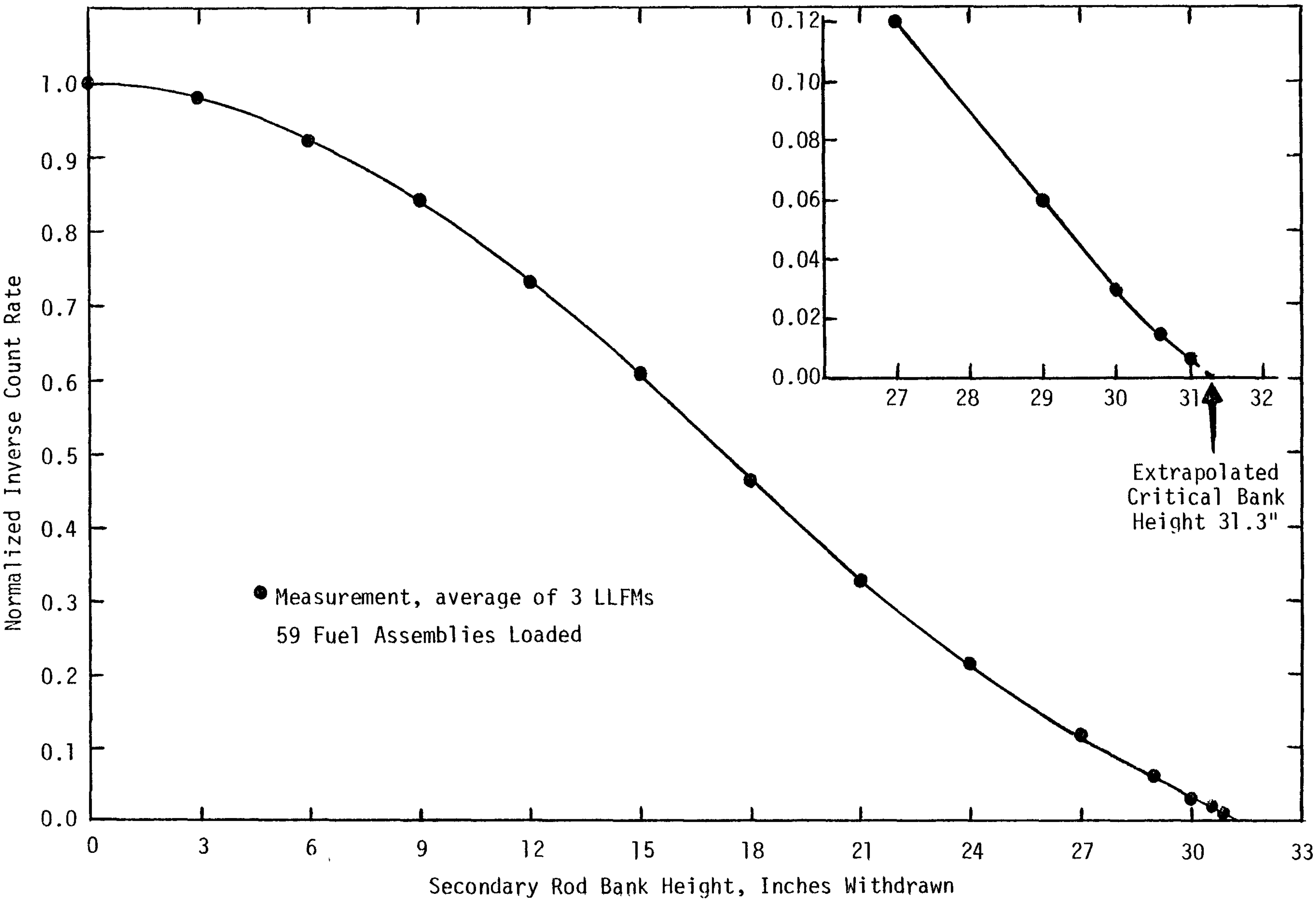


Figure 26. FFTF Initial Approach to Critical.

Figure 27.

LOW LEVEL FLUX MONITOR NEUTRON COUNTRATE

FFTF INITIAL CRITICALITY FEBRUARY 9, 1980

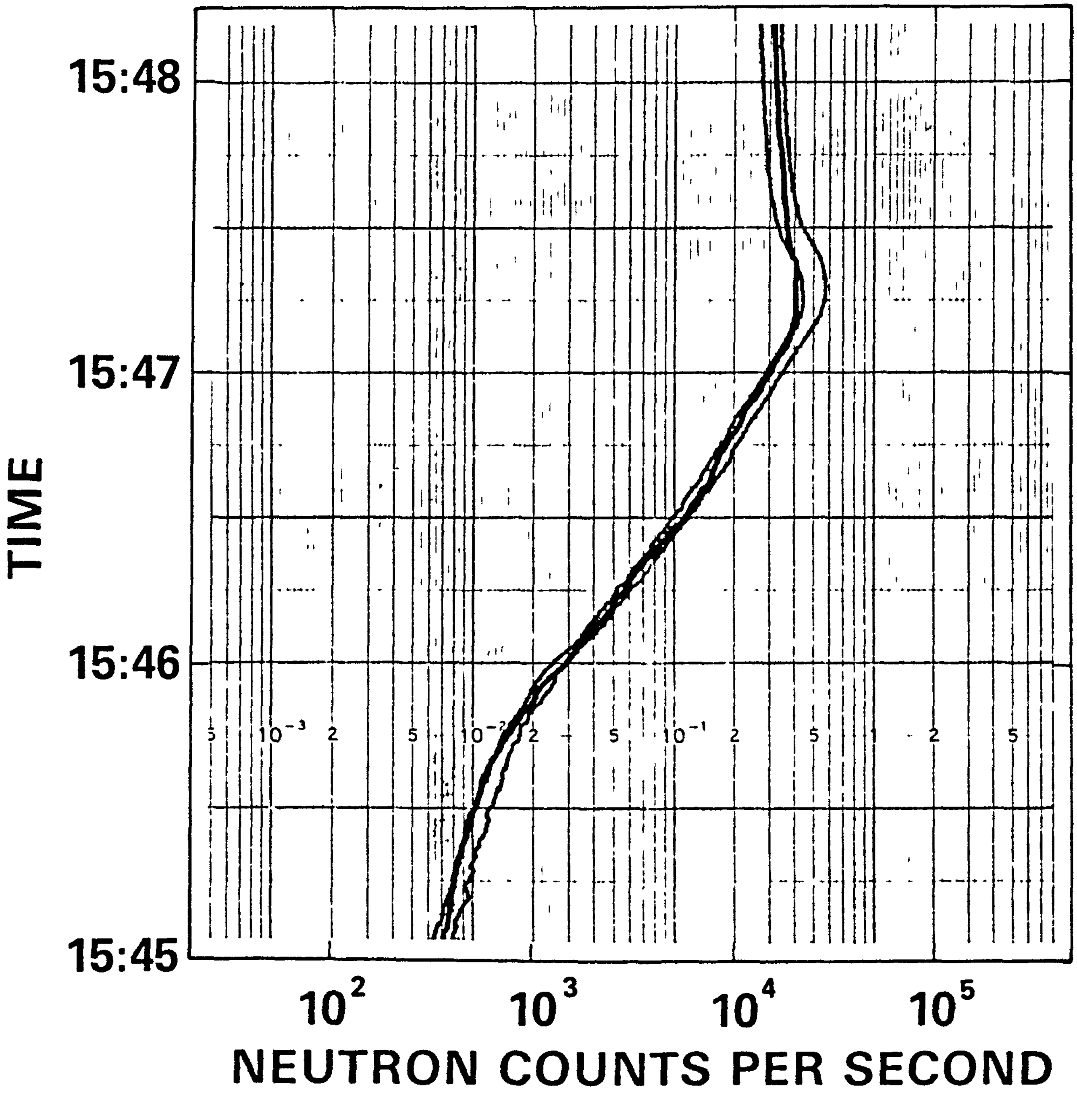
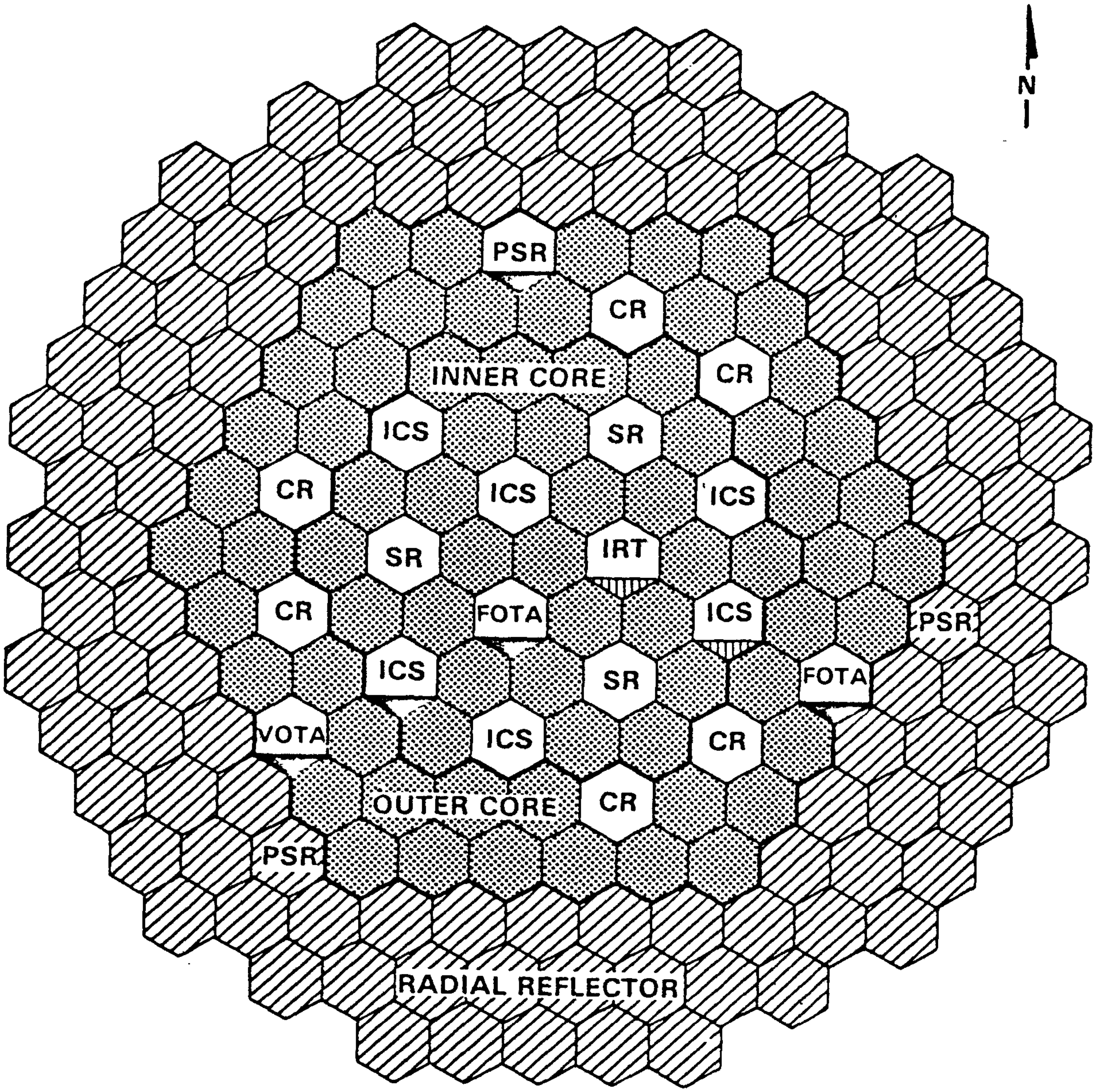


Figure 28. **FULLY LOADED FFTF CORE FEBRUARY 19, 1980**



- SR — SAFETY ROD
- CR — CONTROL ROD
- FUEL DRIVER ASSEMBLIES
- CLOSED LOOP LOCATIONS
- CONTACT INSTRUMENTED OPEN TEST LOCATIONS

- PSR - PERIPHERAL SHIM ROD
- ICS - IN-CORE SHIM
- VOTA - VIBRATION OPEN TEST ASSEMBLY
- FOTA - FUELS OPEN TEST ASSEMBLY
- IRT - IN-REACTOR THIMBLE

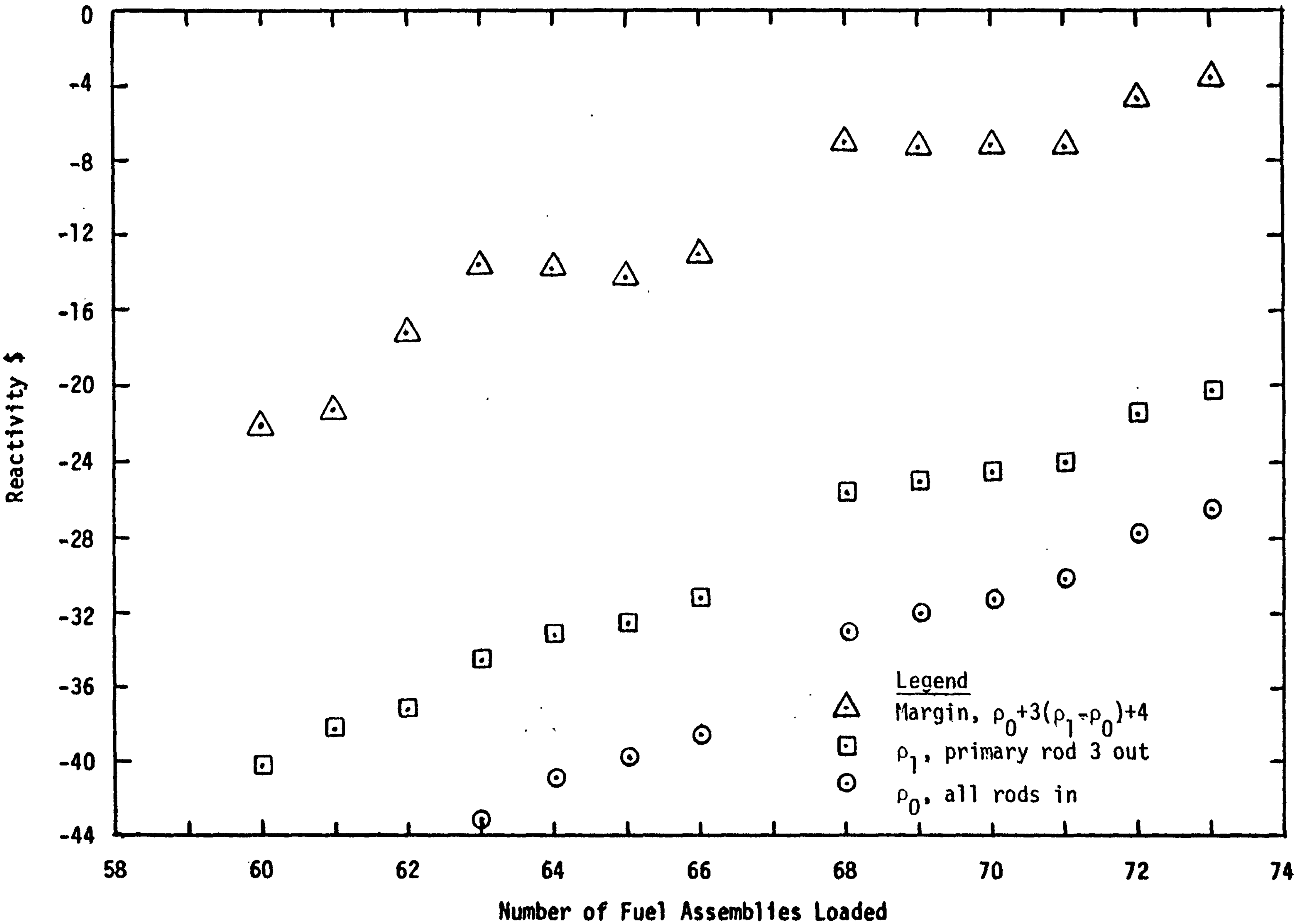


Figure 29. Reactivity During Final FFTF Core Loadout - LLFM A.

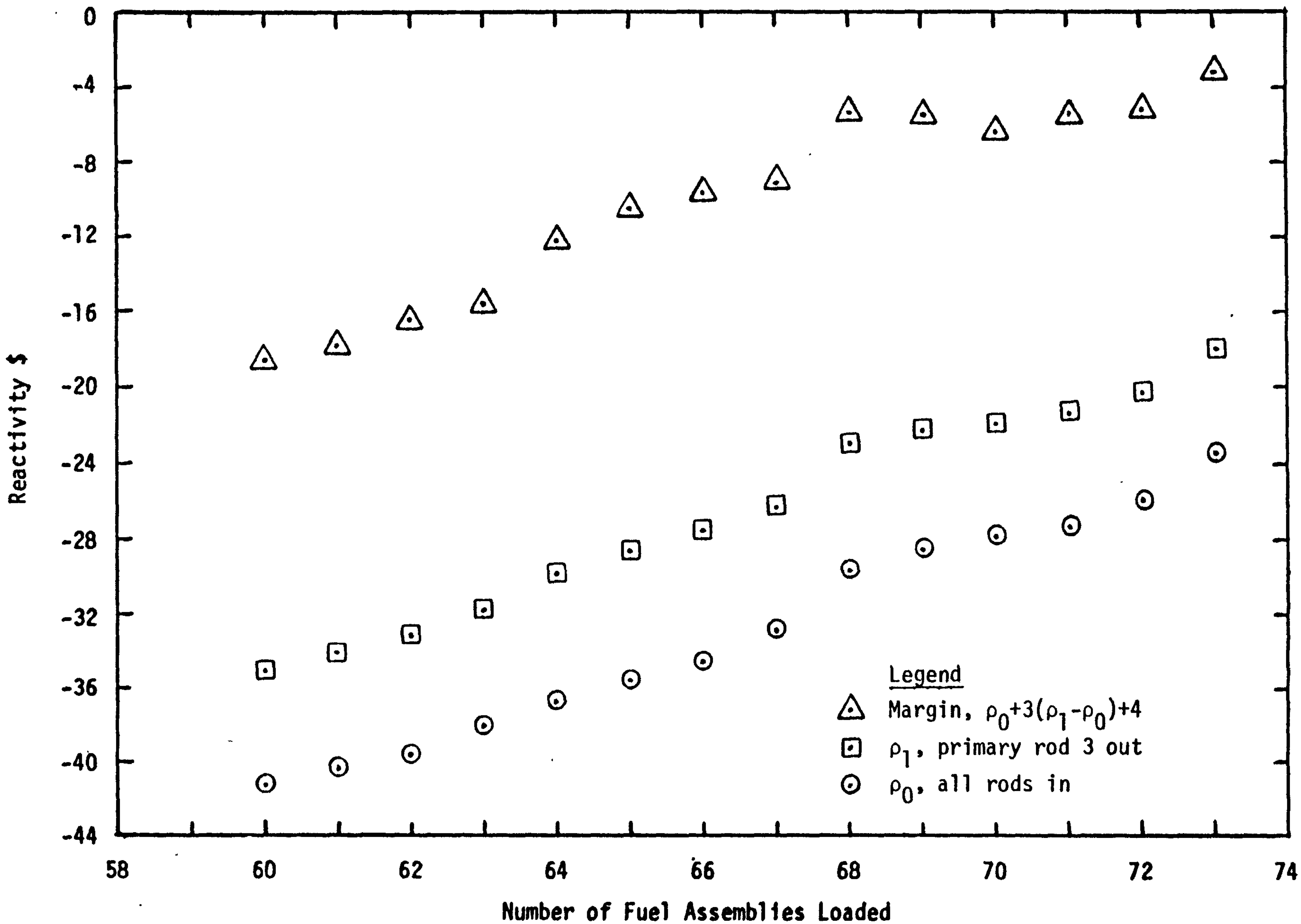


Figure 30. Reactivity During Final FFTF Core Loadout - LLFM B.

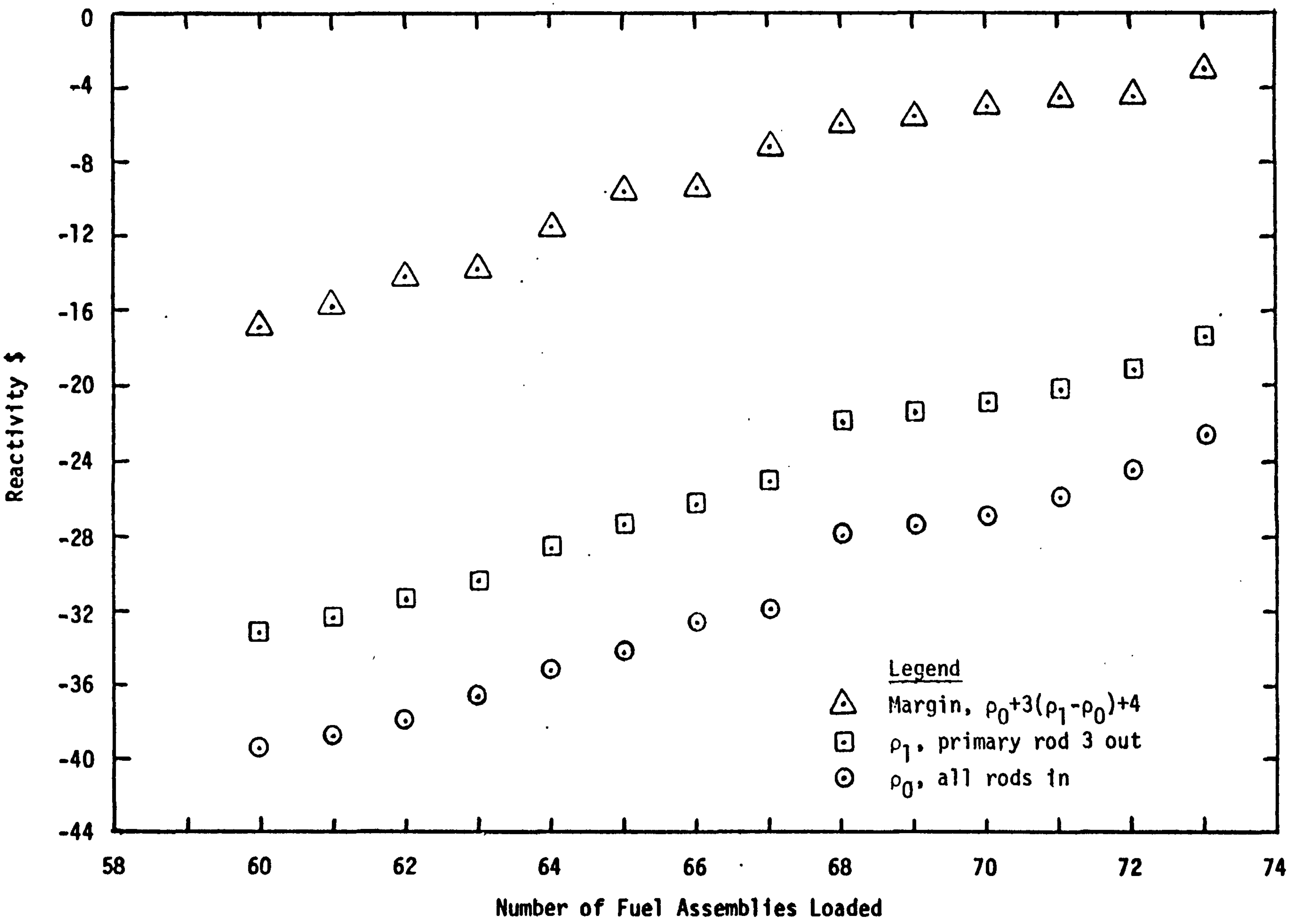


Figure 31. Reactivity During Final FFTF Core Loadout - LLFM C.

On February 22, 1980, the reactor was again taken to a near-critical state; the first time since the completion of fuel loading. By this time the special startup chambers in the IRT had served their purpose and had been removed from the reactor. Consequently, count rate data were recorded for the LLFMs only. At 12:30 p.m. on February 22 the three primary rods had been withdrawn and the plot of inverse count rate as a function of secondary rod bank height had begun. The count rates that were recorded with the primary safety rods fully withdrawn and the secondary control rods fully inserted are given in Table III. Uncertainties are 1σ from counting statistics. The secondary rods were withdrawn, one at a time, a preplanned distance or until the count rate doubled. Rod pulls were made in two-inch increments until a bank height of twelve inches from full insertion was achieved. At each two-inch increment in secondary rod bank height, count data were recorded and plotted. The secondary rods were then pulled to bank heights of 13", 13.5", 13.8" and 13.9", with count data taken at each bank height. By 5:40 p.m. the rod pulls had proceeded to the secondary rod bank height of 13.9", from which it was expected that one more rod pull would achieve criticality. Figure 32 shows the inverse count rate as a function of secondary rod bank height. The extrapolated critical rod bank height, shown in Figure 32, was 14.1".

TABLE III

CORE FULLY LOADED CRITICALITY APPROACH COUNT RATE NORMALIZATION DATA

Count Rate (cps) With 73 Fuel Assemblies Loaded and Primary Rods Out

LLFM		
A	B	C
632.6±1.5	424.3±1.4	603.8±1.5

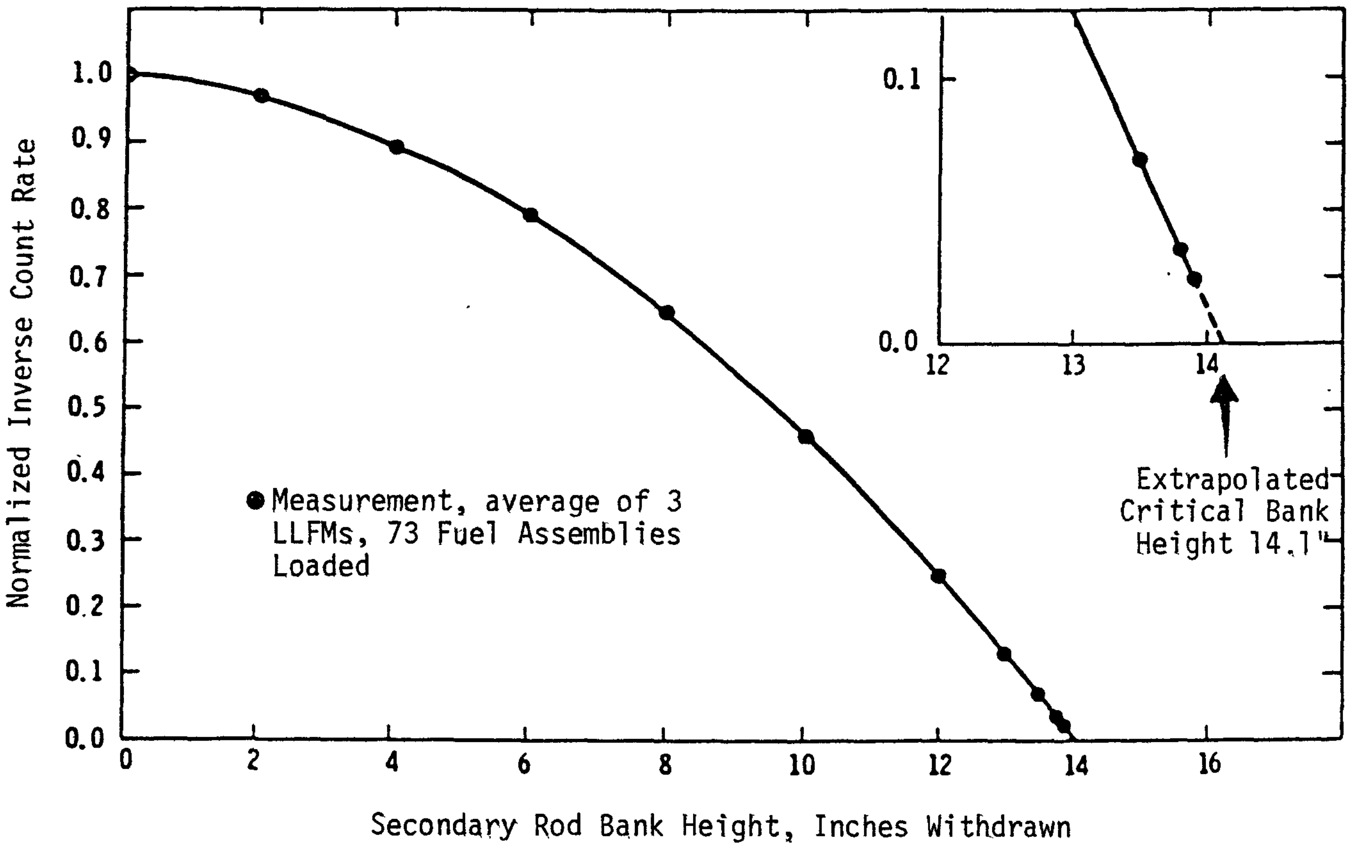


Figure 32. FFTF Approach to Critical After Core Fully Loaded.

However, the objective of the February 22 approach to criticality was not to achieve criticality but to assess the subcritical reactivity state by the inverse kinetics method analysis of a rod drop experiment. This analysis was performed by the IKRD[10] computer program. The reactivity state, as determined by dropping rod 3 and analyzing the results, was -16¢. This experiment set the stage for performing subsequent subcritical reactivity measurements, rod worth measurements, and shutdown system margin measurements, as discussed in Section III.

On March 8, 1980, at 8:13 a.m., following completion of the subcritical measurements mentioned above, the reactor achieved criticality for the first time following the completion of core loading. Rod movements were performed in a manner similar to the February 22 criticality approach, until a secondary rod bank height from full insertion of thirteen inches was achieved. Then each secondary rod was pulled one inch to a fourteen inch height, with count data taken after each individual rod pull. With all secondary rods banked at fourteen inches, rod 4 was pulled to a 15.4" height, achieving criticality and a startup rate of 0.3 decade per minute. Startup was terminated prior to reaching 1 kW of fission power by reinsertion of rod 4.

III. SUBCRITICAL REACTIVITY, ROD WORTH AND SHUTDOWN MARGIN MEASUREMENTS

(a) Reactivity Comparisons, Inverse Kinetics Rod Drop vs. Modified Source Multiplication (MSM)

MSM[9] is the method that will be used to monitor the reactivity of the FFTF during all subcritical operations such as, startup, refueling, and reactor component changeout. The method is based on measured neutron count rates and core configuration factors calculated for the neutron detectors. These factors account for neutron detection efficiency changes resulting from core configuration changes, such as rod movements.

Before MSM was fully implemented as a reactivity monitoring system for FFTF, its accuracy was determined from reactivity comparisons with IKRD.[10] This latter method is a technique of measuring control rod worths by rod-drop experiments performed with the reactor near critical.

On March 2, a near-critical, 0.28$ subcritical, core configuration was achieved with three primary rods and control rod 5 fully withdrawn, and control rod 7 withdrawn twenty-eight inches. This established a reference configuration from which individual as well as multiple rods were dropped. The neutron level transient resulting from each rod drop was analyzed by the IKRD method to obtain the predrop and postdrop reactivities and thereby the worth of the dropped rod. These analyses also yielded the effective neutron source strength or calibration constants required by the MSM technique. The constants obtained from dropping control rod 7 were used for all subsequent MSM assessments. Equilibrium count rate measurements made before and after the drop completed the data necessary for the MSM assessment of the predrop and postdrop reactivities to obtain the worth of the dropped rod. These experiments and reactivity evaluations provided a direct comparison of the two methods for reactivities to about 16$ subcritical. The initial core configuration and combinations of rods to be dropped were chosen so that significant (up to 19%) detection efficiency changes would be introduced. This ensured that the MSM technique would be taxed to its anticipated limits.

The results of these experiments and assessments are shown in Table IV. Also shown for comparison are predicted values of the reactivity worths of the experiments obtained from two-dimensional HEX geometry calculations using twelve neutron energy groups. The uncertainties indicated include contributions from both the uncertainties in the measurements as well as the estimated uncertainty in the detector efficiency corrections required. These results show that rod worths evaluated by MSM are accurate to within 5% even as far subcritical as $\sim$16$.

TABLE IV

MSM EVALUATION EXPERIMENTS NEAR CRITICAL

Rods Dropped	Rod Worths ($)			Ratios	
	Calculated	MSM[9]	IKRD[10]	Calculated/ IKRD	IKRD/MSM
5	3.85	3.75 ± 0.07	3.84 ± 0.05	1.00 ± 0.01	1.02 ± 0.02
1	6.11	6.00 ± 0.11	6.04 ± 0.09	1.01 ± 0.01	1.01 ± 0.02
2	6.00	5.78 ± 0.09	5.89 ± 0.09	1.02 ± 0.02	1.02 ± 0.02
3	4.78	4.59 ± 0.07	4.65 ± 0.07	1.03 ± 0.02	1.01 ± 0.02
5,1	9.12	8.89 ± 0.19	9.00 ± 0.16	1.01 ± 0.02	1.01 ± 0.03
5,3	9.08	8.74 ± 0.16	8.80 ± 0.14	1.03 ± 0.02	1.01 ± 0.02
1,2	12.66	11.99 ± 0.19	12.45 ± 0.29	1.02 ± 0.02	1.04 ± 0.03
1,3	11.33	10.75 ± 0.21	11.01 ± 0.25	1.03 ± 0.02	1.02 ± 0.03
5,1,2	16.07	15.13 ± 0.29	15.84 ± 0.35	1.01 ± 0.02	1.05 ± 0.03
5,2,3	15.96	14.81 ± 0.25	15.22 ± 0.39	1.05 ± 0.03	1.03 ± 0.03

(b) Reactivity Comparisons, Estimated Worths - Full Shutdown vs. Modified Source Multiplication

Following the rod drop near critical on March 2, 1980, the reactor was fully shut down and individual and multiple rods were withdrawn to verify the accuracy of MSM[9] assessments during refueling operations. Since the IKRD[10] technique does not yield accurate results far from critical, experimental reactivity worths for comparison with the MSM results were estimated from the IKRD measurements made near critical. These estimates consisted of multiplying the IKRD rod drop reactivity worths by the calculated change in the worths from near critical to full shutdown.

The results of the MSM measurements and rod worth estimates are shown in Table V where the MSM technique is seen to be accurate to within 1$ for core reactivity changes as small as ∿4$. Further analyses are required to verify the rod worth estimates.

TABLE V

MSM EVALUATION EXPERIMENTS SHUTDOWN

Rod(s) Withdrawn	Rod Worths ($) IKRD Near Critical	Rod Worths ($) Estimated for Withdrawn	Rod Worths ($) MSM Shutdown	Estimated MSM
5	3.84 ± 0.05	4.37 ± 0.06	3.65 ± 0.24	1.20 ± 0.08
1	6.04 ± 0.09	6.39 ± 0.10	6.59 ± 0.14	0.97 ± 0.03
2	5.89 ± 0.09	6.26 ± 0.10	5.53 ± 0.21	1.13 ± 0.05
3	4.65 ± 0.07	6.04 ± 0.09	5.56 ± 0.16	1.09 ± 0.04
5,3	8.80 ± 0.14	9.50 ± 0.15	8.55 ± 0.21	1.11 ± 0.03
5,2,3	15.22 ± 0.39	14.88 ± 0.38	13.95 ± 0.21	1.07 ± 0.03

(c) Full Shutdown Reactivity Assessments

The all-control-rods-inserted reactivity assessment of the FFTF reactor was provided by the MSM[9] method using calibration constants and equilibrium count rates measured for each of the three LLFMs located in the shield region of the core. The calibration constants were obtained from a rod-drop experiment performed with the core 0.15$ subcritical, with three primary rods fully withdrawn, and the secondary rods in a bank at 13.9 inches. The result of this MSM assessment demonstrates that the full shutdown reactivity of the core is 23.6$ with an uncertainty of 0.2$. The predicted value of this quantity using the FFTF design methods[2] was 25.7$.

(d) Total Worth of Primary Rods

The total reactivity worth of the three primary rods was determined by measuring the subcriticality of the reactor with the three primary rods fully withdrawn, and again with the three primary rods fully inserted (the six secondary control rods remained fully inserted), and then subtracting the two measured subcriticality values. The worth was found to be 16.3$ at refueling temperature with an uncertainty of 0.1$. The FFTF design methods[2] yielded a total primary rod worth of 19.43$.

(e) Total Worth of Secondary Rods

The total reactivity worth of the six secondary control rods was determined with the primary control rods fully withdrawn and fully inserted. In the first configuration the subcriticality of the reactor with the secondary control rods fully inserted and the primary control rods fully withdrawn was determined using the MSM technique, to be 7.32±0.06$. From the observed reactivity worth profile of the secondary control rods during the approach to critical with 59 fuel subassemblies loaded, it was estimated that 33.9±1.7% of the total worth would be realized by withdrawing the rods to the estimated critical position of ∿14.1 inches. This implies a total secondary control rod worth of 22±1$. The predicted value for this parameter was 23$.

For the second configuration the subcriticalities of the reactor with all control rods fully inserted and with only the six secondary control rods withdrawn were determined using the MSM technique. The difference between these reactivities and thus the worth of the secondary rods with the primary control rods fully inserted was 19.9±0.1$. This worth was predicted to be 24$.

(f) Individual Control Rod Reactivity Worths

The reactivity worth of each of the three primary control rods (1, 2, 3) and each of the six secondary control rods was measured using the IKRD[10] rod drop technique. Technical specifications on the operation of FFTF require that no secondary control rod worth exceed 5$ and no primary control rod worth exceed 8$. From the results shown in Table VI it can be seen that these specifications were met easily. The uncertainties quoted for the IKRD results include a contribution due to the detector efficiency changes but are predominantly the result of the random noise on the neutron data. The estimated magnitude of this uncertainty was verified by repeating the measurement for control rod 5, five additional times. The variance in the data for the repeated measurements was consistent with the uncertainty shown in Table VI.

The calculated rod reactivity worths were obtained using the FFTF design methods and appear to contain a bias of from 3 to 5%.

TABLE VI

REACTIVITY WORTHS OF INDIVIDUAL CONTROL RODS

Rod	Rod Worths ($)		Ratio
	Calculated	IKRD[10]	IKRD/Calculated
1	5.47	5.82 ± 0.08	1.064 ± 0.015
2	5.30	5.52 ± 0.07	1.042 ± 0.013
3	5.16	5.40 ± 0.06	1.047 ± 0.012
4	3.90	4.07 ± 0.04	1.044 ± 0.010
5	3.94	4.11 ± 0.02	1.043 ± 0.005
6	3.52	3.57 ± 0.04	1.014 ± 0.011
7	3.82	3.86 ± 0.04	1.010 ± 0.010
8	3.18	3.17 ± 0.03	.997 ± 0.009
9	3.70	3.83 ± 0.04	1.035 ± 0.011

(g) Maximum Reactivity Addition Rates

In the transients considered for the Final Safety Analysis Report (FSAR) for FFTF, it was assumed that the maximum possible reactivity addition rate resulting from the unrestricted withdrawal of a secondary control rod at its maximum speed of 9.8 inches/minute was 3.4¢/second. The rod was assumed to be worth 5$ in reactivity and a calculated reactivity versus position profile was used. This profile was verified for control rod 5 by running it in at nine inches/minute from a slightly subcritical state. An inverse kinetics analysis of the resulting neutron count rate history using the point kinetics parameters obtained from a rod drop IKRD analysis of the same rod yielded the reactivity of the core as a function of time. The maximum rate of change of reactivity was found to be 2.7¢/second. Had control rod 5 been worth 5$ this implies a rate of 3.3¢/second would have been obtained, thus verifying the FSAR assumption.

(h) Primary System Shutdown Margin

Since the primary control rods may be needed to shut down the reactor in an accident or potential accident condition, their effectiveness in performing this function for the final reactor as-built configuration must be verified. It is presumed that the reactor is initially critical and operating at power, and then, for an unspecified reason, the most reactive secondary rod is fully withdrawn. It is required by technical specification that insertion of the two least reactive primary rods shut down the reactor in this configuration, and maintain it in a shutdown condition if cooling to refueling temperature is also experienced. This is equivalent to requiring that the worth of the two least reactive primary rods, minus the power defect from power operation to refueling temperature, be greater than the worth of the most reactive secondary rod. The worth of the two least reactive primary rods was determined by the IKRD technique to be 11.22±0.18$. This worth minus a calculated power defect of 3.16±0.63$ yields 8.06±0.64$ which is substantially greater than the maximum secondary control rod worth of 4.11±0.02$.

(i) Secondary System Shutdown Margin

Technical specifications for FFTF operation require that sufficient control exist in the secondary control rod system to shut the reactor down to at least hot standby conditions even if the most reactive secondary control rod is fully withdrawn.

This requirement is most difficult to meet when the core is most reactive; i.e., at the beginning of an operating cycle with the primary control rods fully withdrawn. The subcritical reactivity of the core was found to be 7.32±0.06$ with the primary control rods withdrawn and all six secondary control rods fully inserted. Correcting this reactivity to hot standby temperatures yields 8.52±0.26$ which indicates that the reactor would remain subcritical even if the most reactive secondary control rod (4.11$) was withdrawn.

(j) Excess Reactivity

The reactivity control remaining in the secondary control rod system after criticality or full power is reached determines how long the reactor can operate with the current fuel load. From previous estimates of the total reactivity worth of the secondary control rod system (22±1$) and the subcritical reactivity with the primary control rods fully withdrawn and the secondary control rods fully inserted (7.32±0.06$) it is estimated that the excess reactivity loss (power defect) of 3.2±0.6$ in attaining full power conditions yields an estimated excess reactivity at full power of 11±1$. Normal operating cycles are anticipated to require 10$ to complete.

(k) Temperature Coefficient of Reactivity

The isothermal reactivity temperature coefficient was measured over the temperature range of 383°F to 417°F, using secondary pump work and Dump Heat Exchanger (DHX) adjustments to obtain the temperature increase. The change in the reactivity of the core was measured by performing IKRD experiments before and after the temperature increase from reproducible initial control rod configurations. The difference between the predrop reactivities from the IKRD analyses yielded the desired reactivity change.

The temperature coefficient obtained (-0.70¢/°F) was in good agreement with the predicted coefficient (-0.64¢/°F).

(l) Differential Rod Worths

The reactivity worth of moving individual secondary control rods a small increment from the mean secondary control rod bank position is required for a variety of tests scheduled for FFTF. Assessments of these worths were made from a single bank position (∿13.7 inches withdrawn) using the MSM technique. Each secondary control rod was first withdrawn one inch and then inserted two inches. The average differential reactivity worths obtained are shown in Table VII. The uncertainties shown include a small contribution arising from the MSM measurements but are predominantly due to the uncertainty in positioning the control rods. This latter uncertainty is estimated to be ±0.025 inches at the 1σ level.

The calculated differential worths were obtained by determining the total reactivity effect of moving all the secondary rods from previous approach to critical data and then distributing this effect among the secondary control rods according to their relative measured reactivity worths. As shown in Table VII, this calculational technique will yield differential worths accurate to better than 5%.

TABLE VII

DIFFERENTIAL CONTROL ROD WORTHS

	Diff. Worths (¢/in)		
Rod	MSM[9]	Calculated	Calculated/MSM
4	15.4±0.4	15.7	1.019±0.027
5	15.7±0.4	15.9	1.013±0.027
6	13.6±0.4	13.8	1.015±0.028
7	14.5±0.4	15.2	1.048±0.028
8	12.1±0.4	12.2	1.008±0.029
9	14.3±0.4	14.7	1.028±0.028

(m) Primary Loop Flow Reactivity Effect

A measurement of reactivity change due to primary sodium coolant flow rate change was made by performing IKRD assessments of the core reactivity with the primary coolant flow rate at >90% and again at 8% of full flow. The difference in these reactivities was 4.00±0.64¢. After correcting for a slight change in the coolant temperature at the core inlet, the total flow reactivity difference was found to be 5.1±0.7¢.

IV. REFERENCES

1. R.A. Bennett et al, "Nuclear Startup, Testing, and Core Management of the FTR," IAEA-SM-244/68, International Symposium on Fast Reactor Physics, September 24-28, 1979, Aix-en-Provence, France.

2. R. B. Rothrock, FFTF Initial Fuel Load Nuclear Analysis, HEDL-TME 80-29. Hanford Engineering Development Laboratory, Richland, WA.

3. W. W. Little, Jr. and R. W. Hardie, 2DB User's Manual - Rev. 1, BNWL-831, Rev. 1, Battelle Northwest, Richland, WA, August 1969.

4. R. W. Hardie and W. W. Little, J , 3DB, A Three-Dimensional Diffusion Theory Burnup Code, BNWL-1264, Battelle Northwest, Richland, WA, 1970.

5. J. W. Daughtry et al, "ZPR-9 Assembly 27: The Fast Test Reactor Engineering Mockup Critical (FTR-EMC)," Applied Physics Division Annual Report, July 1, 1970 to June 30, 1971, pp. 33-39, ANL-7910, Argonne National Laboratory, Argonne, IL.

6. E. M. Bohn, "FTR Critical Assemblies," Reactor Development Program Progress Report-September 1974, pg. 6.3, ANL-RDP-32, Argonne National Laboratory, Argonne, IL, September 1974.

7. R. B. Kidman and R. E. Schenter, FTR Set 300-S, Multigroup Cross Sections for FTR Shielding Calculations, HEDL-TME, 71-184, Hanford Engineering Development Laboratory, Richland, WA, December 1971.

8. R. W. Hardie and W. W. Little, Jr., 1DX, A One-Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections, BNWL-954, Battelle Northwest, Richland, WA, September 1968.

9. R. M. Fleischman, J. W. Upton, Jr. and R. A. Bennett, Subcritical Reactivity Surveillance Procedures for the Fast Flux Test Facility, HEDL-TME-73-43, Hanford Engineering Development Laboratory, Richland, WA, May 1973.

10. a) R. A. Sevenich, "Inverse Kinetics," Reactor and Safety Engineering Technical Progress Report - April-June 1975, pp. 1-14, HEDL-TME 75-70, Hanford Engineering Development Laboratory, Richland, WA, August 1975.

    b) C. Y. Yang and R. W. Albrecht, "Inverse Kinetics Determination of Subcriticality," Nuclear Technology, 22, pg. 323, 1974.