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CRITICAL EXPERIMENTS WITH THE UO_2-2 wt.% PuO_2 BATCH CORE IN THE PRTR

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ABSTRACT

The initial phase of the Batch Core Experiment comprised the loading of a new core of UO_2-2 wt% PuO_2 fuel elements and the performance of a series of critical tests to determine the physics parameters of this new core. Measurements included the critical loadings for boron concentrations ranging from zero to about 21 parts per million by weight (wppm) of ^{10}B in the D_2O moderator, moderator level coefficients of reactivity over a 25-inch range of moderator heights at several boron concentrations, moderator void worths, thermal neutron flux distributions within the core, the effect of loss of coolant from fuel channels, fuel element worths, and kinetics parameters. The results of these measurements are presented in this report.

At specified burnup levels during the Batch Core Experiment, interim critical tests were conducted. These tests included measurements of core excess reactivity, fuel element worths, poison rod worths, and kinetics parameters. The results of these measurements are also presented in this report.

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I. INTRODUCTION

A major obstacle to the use of plutonium as fuel in present-generation power reactors has been the uncertainties which must be attached to the values of core reactivity lifetime predicted by calculation. To help reduce these uncertainties, a large-scale fuel burnup experiment (The Batch Core Experiment)[1] was conducted in the Plutonium Recycle Test Reactor (PRTR). The basic fuel loading remained essentially unperturbed throughout the reactivity lifetime of the core, thus providing a unique set of data for use in checking methods of calculating reactivity lifetimes of reactor cores.

The initial phase of the Batch Core Experiment comprised the loading of a new core of UO_2-2 wt% PuO_2 fuel elements and the performance of a series of critical tests to determine the physics parameters of this new core. Measurements included the critical loadings for boron concentrations ranging from zero to about 21 parts per million by weight (wppm) of ^{10}B in the D_2O moderator, moderator level coefficients of reactivity over a 25-inch range of moderator heights at several boron concentrations, moderator void worths, thermal neutron flux distributions within the core, the effect of loss of coolant from fuel channels, fuel element worths, and kinetics parameters. The results of these measurements are presented in this report.

At specified burnup levels during the Batch Core Experiment, interim critical tests were conducted. These tests included measurements of core excess reactivity, fuel element worths, poison rod worths, and kinetics parameters. The results of these measurements are also presented in this report.

II. REACTOR DESCRIPTION

A. Reactor Type

The PRTR[2] is a vertical pressure tube reactor, heavy water moderated and cooled. The reactor was designed for diversified programs of fuel irradiation and testing. Heavy water as coolant and moderator has many safety advantages for this reactor and does not impose as many restrictions upon the variety of fuel configurations as do most other alternatives.

B. Core and Reflector

The unpressurized, heavy water moderator, nominal purity 99.75%, is contained in an aluminum tank, the calandria, which has an inside diameter of 84 inches and height of 115 inches. Passing through the calandria are 84 fuel channels, each having an outside diameter of 4.25 inches and a wall thickness of 0.065 inches, and one central tube which has a 6-inch outside diameter and a wall thickness of 0.085 inches. These fuel channels are in a triangular lattice with a pitch of 8 inches. The core plan and location index is shown in Figure 1.

In addition to the fuel channels, there are: thirteen vertical flux monitoring channels (1.5 in. O.D x 0.049 in. wall aluminum) denoted by M-XXX; thirteen openings into the top of the calandria denoted by P-XXX; eighteen openings into the top of the calandria, denoted by C-XXX, which formerly contained mechanical shim assemblies; a capsule containing equipment for measuring creep in structural materials, located in channel 2049, three moderator level probes, two of which were installed

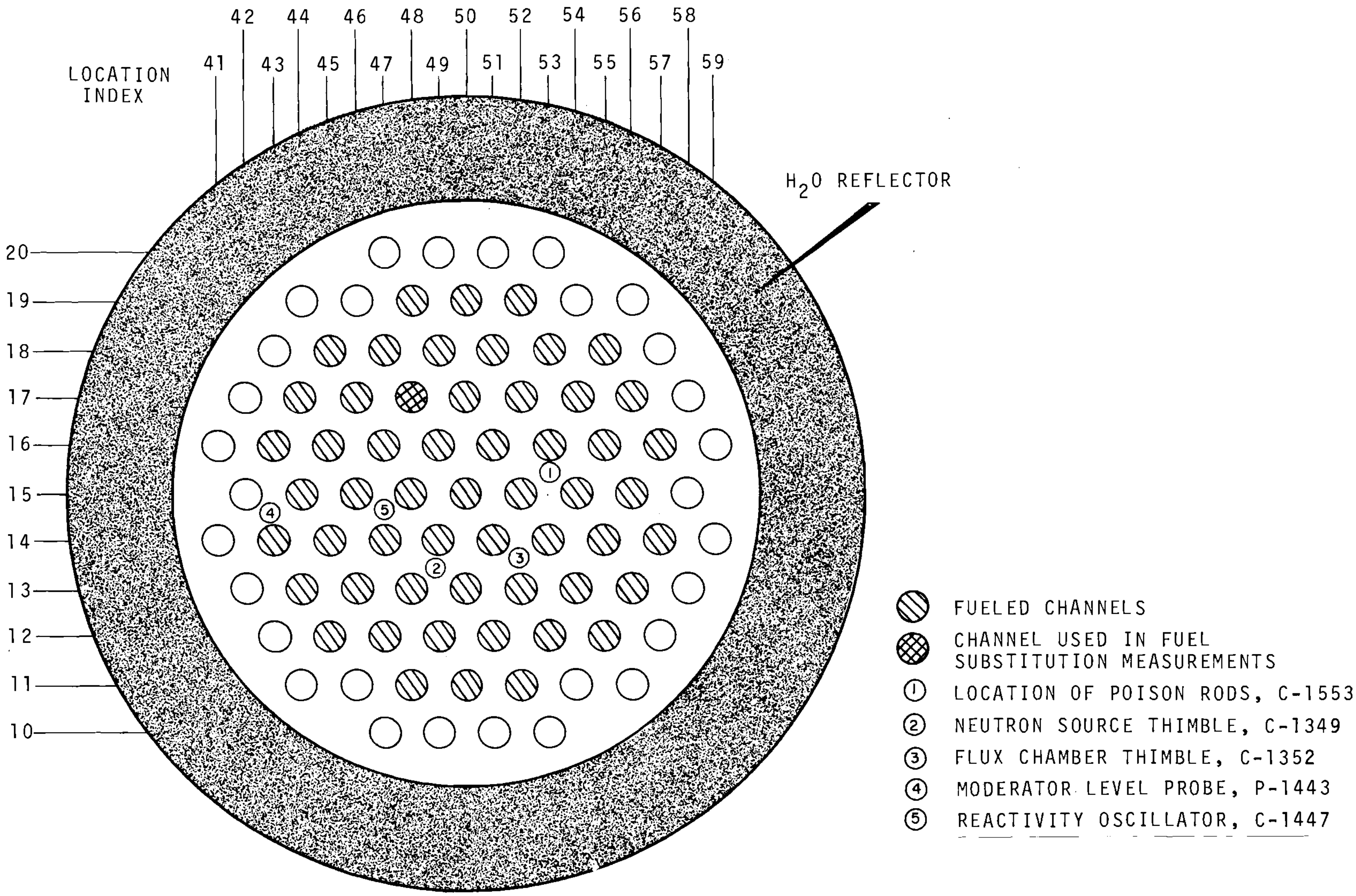


FIGURE 1. PRTR BATCH CORE CONFIGURATION

during the original PRTR startup in 1960 in locations P-1750 and P-1443, and the third which was installed just prior to these critical tests in location C-1047. Detailed descriptions of these devices are contained in Appendix III.

Each fuel channel contains a pressure tube of Zircaloy-2 inside the calandria shroud tube. This tube contains the fuel element and heavy water coolant. The nominal purity of the coolant was 97% D_2O. The tube has an inside diameter of 3.25 inches and wall thickness of 0.154 inch. The gap between the pressure tube and the shroud tube, nominally 0.435 in., contains helium gas as a thermal barrier.

A light water radial reflector is contained in a separate annular tank 24 inches in thickness surrounding the calandria. The reflector does not extend down to the level of the calandria base, but leaves the lower 2 feet of the calandria essentially without radial reflection. This space is a dump chamber which can receive moderator from the calandria very rapidly during an emergency shutdown and makes possible a reduction of moderator level by 2 feet in less than a second. The fuel is located vertically in the calandria to provide 34 inches of bottom reflector and normally from 5 to 9 inches of top reflector, depending on the operating moderator level.

C. Primary Control System

A schematic diagram of the Gas Balance System used to control moderator level is shown in Figure 2, together with a vertical cross section of the reactor. The heavy water moderator has a free surface at an annular weir, extending around the base of the calandria inside the dump chamber.

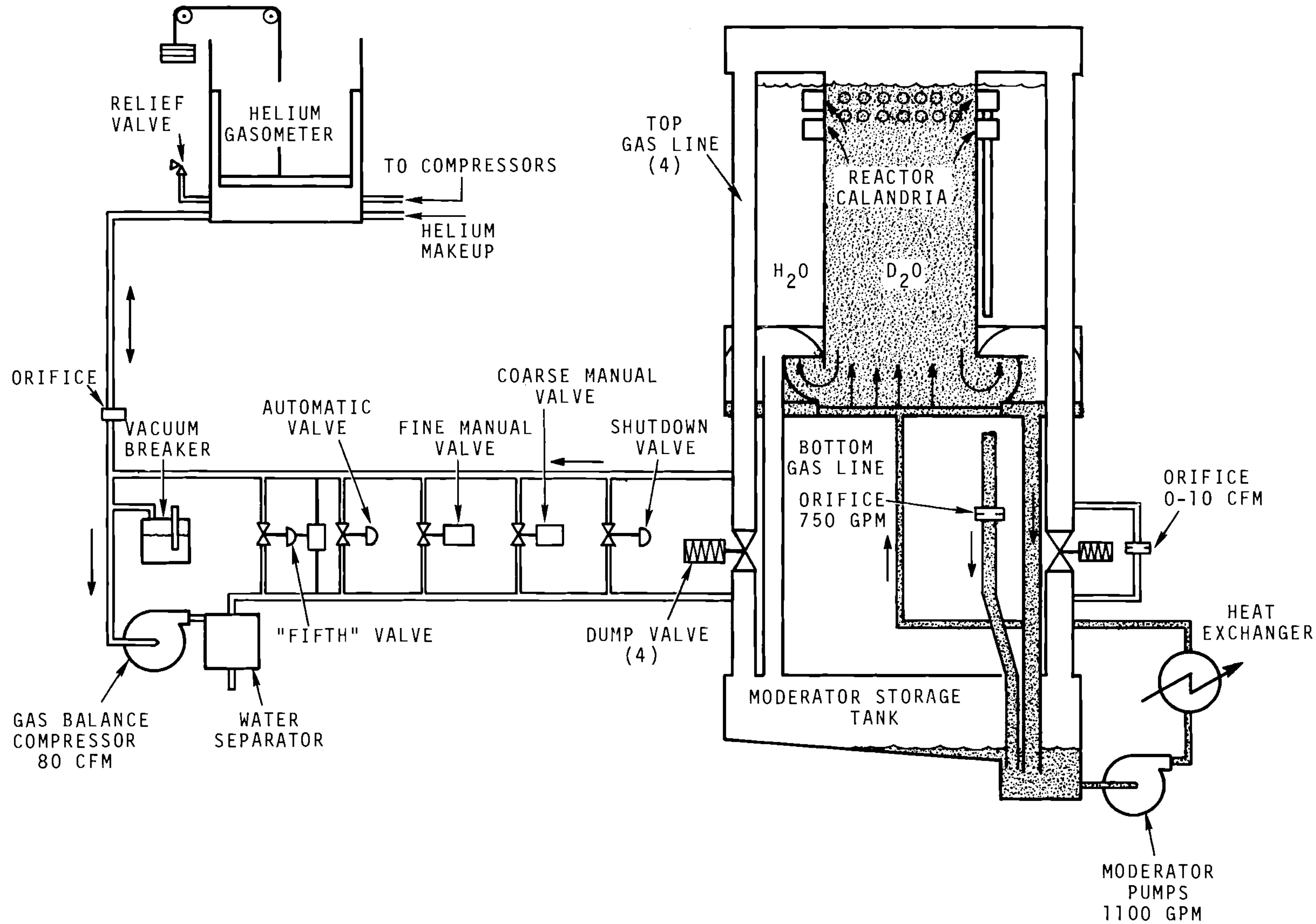


FIGURE 2. PRTR MODERATOR LEVEL CONTROL SYSTEM

The helium atmosphere at this point is maintained at a sufficiently high pressure, relative to the pressure at the top of the calandria, to support the moderator at any desired level in the calandria. Both the helium gas and the moderator itself are continually circulating. The gas pressure difference is maintained by a compressor and a set of bypass valves, which constitute the primary control system of the reactor. Automatic control, fine and coarse manual control, and both normal and emergency shutdowns are provided for by different valves in this system. Primary control is used to maintain or adjust reactor power.

D. Secondary Control System

A chemical shim system is used to control the large amount of excess reactivity contained in the Batch Core Loading (∿260 milli-k).

Boric acid, enriched to 92% in the ^{10}B isotope, is dissolved in D_2O and injected into the moderator system as needed. Removal of boron from the moderator is accomplished by ion exchange. Operation of the chemical shim system is controlled from a console in the PRTR control room.

E. Fuel Elements

A typical fuel element for the Batch Core Experiment is shown in Figure 3. The elements are 19-rod clusters of vibrationally-compacted UO_2-2 wt% PuO_2 rods.[3] The fuel in each rod is 0.505 inches in diameter and 58-3/4 inches in length (including a pellet of depleted UO_2 located at each end) and is contained in tubes of Zircaloy-2 which have wall thicknesses of 0.030 inches. The isotopic compositions of the plutonium

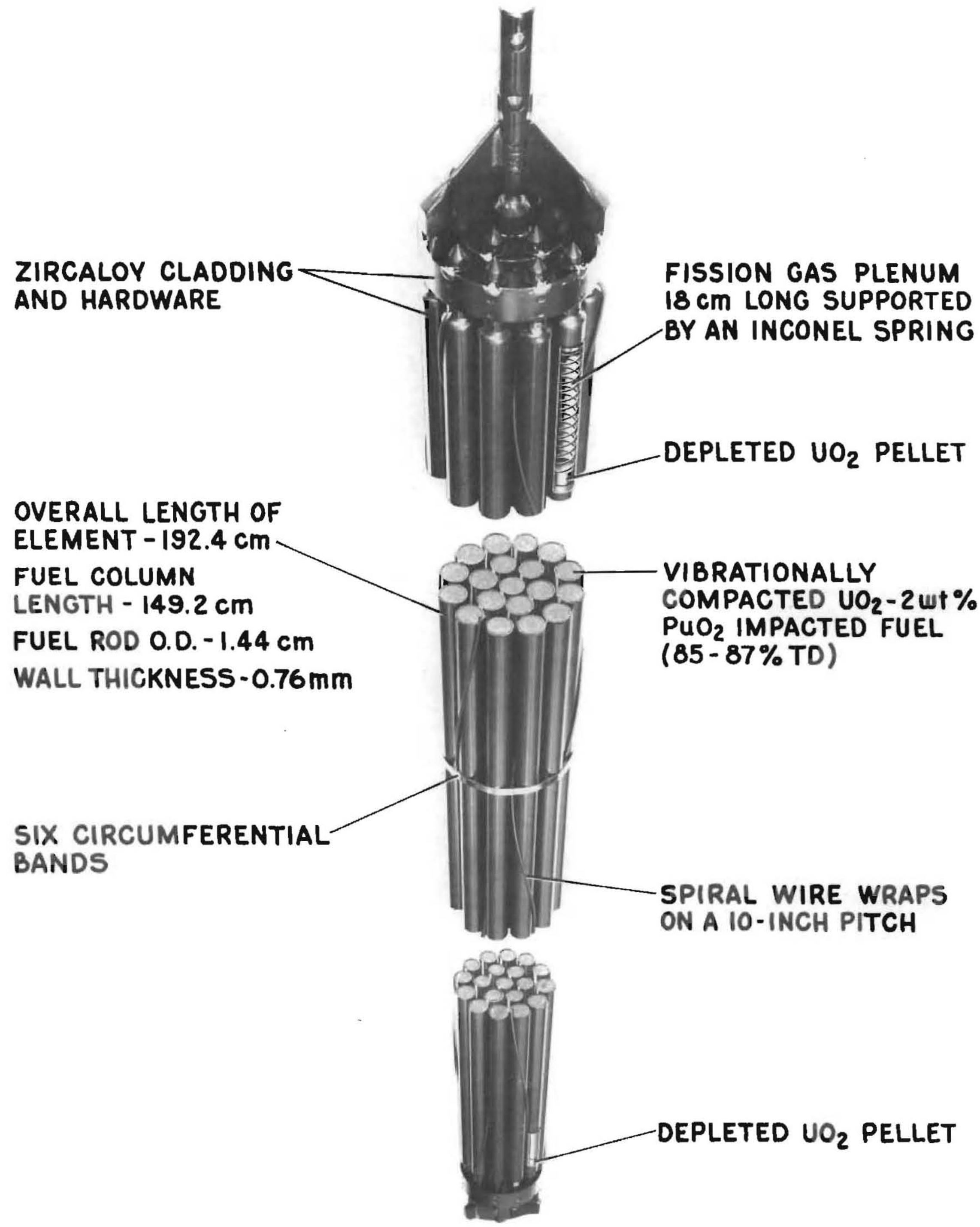


FIGURE 3. PRTR BATCH CORE NINETEEN-ROD CLUSTER FUEL ELEMENT

are: ^{239}Pu, 90.81%; ^{240}Pu, 8.11%; ^{241}Pu, 1.03%; ^{242}Pu, 0.06%, where the percentages are by weight. Each 19-rod element contains an average of 29.794 kg natural uranium and an average of 599.65 grams plutonium.

III. TEST EQUIPMENT

A neutron source (Pu-Be) with a strength of $\sim 8 \times 10^6$ n/sec was installed in an aluminum thimble (2 in. O.D. x 0.063 wall) in location C-1349, to provide a neutron level sufficient for multiplication measurements. The source remained in place during all the initial critical tests.

Supplementary neutron sensitive chambers were installed in an aluminum thimble (2.25 in. O.D. x 0.063 in. wall) in location C-1352, and in a dry fuel channel (PT-1053*) to provide flux monitoring during the initial loading. The normal reactor startup chambers, located in the annular reflector, did not provide adequate multiplication data until the initial critical loading had been achieved. After initial criticality, the chambers in the thimble (C-1352) were removed.

Special piping was attached to Process Tube 1550* to facilitate draining and refilling of that channel independent of the normal primary coolant system. The channel either was filled with stagnant coolant or was dry, depending upon the particular test being performed. Because voiding measurements required reactor operation with the coolant removed, it was necessary to monitor the temperature of the fuel rod cladding to insure against inadvertently overheating the fuel. For that reason, six thermocouples, clad in stainless steel jackets 1/16 inch in diameter, 0.010 inch wall thickness, were installed in contact with the cladding of the fuel rods of the element charged in the central channel in the core (1550). These thermocouples monitored the temperature of the cladding throughout the testing period.

*The terms pressure tube and process tube, and the abbreviation PT are used interchangeably in this report.

Measurements of the axial thermal neutron flux profile were made by activating gold foils (0.006 in. thick x 0.5 in. dia.) mounted on a lucite traverse bar which was installed in an aluminum thimble (1" O.D. x 0.063 in. wall) in location C-1553. The radial flux shape across a lattice cell was measured by activating lutetium oxide foils (0.046 in. thick x 0.447 in. dia.) which were mounted in a special fuel element charged in PT-1550 and on an aluminum traverse device inserted into an access hole, location P-1550. Detailed descriptions of the test equipment and thimbles are contained in Appendix III.

IV. TEST PROCEDURES

The majority of the testing comprised critical approaches via increasing moderator height and measurement of rising reactor periods. Inverse multiplication data were plotted to obtain extrapolated critical moderator heights. The reactivity worths of small increments of moderator height were determined by positive period measurements.

Each core loading configuration was assigned a core loading number. For example, the initial loading was designated A-1, the first multiplication measurement (approach-to-critical) was MA-1. The first critical loading was A-7, the corresponding multiplication measurement was MA-7, the rising period measurements were PA-7-1 through PA-7-3. This nomenclature was followed throughout the performance of the critical tests. Each approach-to-critical was followed with three or more rising period measurements. The moderator level was increased above the critical height by a different amount for each period measurement, thus providing data for the determination of the moderator level coefficient of reactivity at that level.

Interim critical test series 1, 2 and 3 were conducted five days after shutdown from power operation. The short shutdown time made rising period measurements impossible in these series due to the D_2O photoneutron effect from the irradiated fuel. Hence, only approaches to critical were used. Series 4 was conducted after an extended reactor shutdown of 338 days. In this case, rising period measurements were made. For all the interim critical tests the D_2O primary system was cold and depressurized with one pump circulating the coolant at low flow.

Samples of the moderator were taken after each change in the boron concentration. The samples were assigned numbers, C-0 for the initial boron-free case, C-1 for the first boron addition, etc., and placed in an Archive Storage file. The true boron content of each sample was determined later by comparison of its reactivity worth with the worth of a sample of borated D_2O of known concentration, measured in the Thermal Test Reactor (TTR). Details of these measurements are presented in Appendix II.

V. TEST PERFORMANCE

Fuel loading for the Batch Core Experiment began on July 16, 1966, and initial criticality was achieved with nine fuel elements and no boron on July 19. Loading and testing continued on an around-the-clock basis until the basic 55-element core had been charged and all major tests had been completed, on August 25, 1966. Additional measurements were performed at various times in the interval between August 25 and the receipt of authorization to begin Power Tests in December 1966.

The dates and core burnups for each series of interim critical tests are summarized in Table I.

Table I

Interim Critical Test Series

Series	Dates	Batch Core Burnup (MWd/MTM)
1	September 15-18, 1967	1650 $\pm$ 16
2	February 2-5, 1968 March 17, 1968	3560 $\pm$ 34
3	June 6-8, 1968	4930 $\pm$ 47
4	June 17-23, 1969	5330 $\pm$ 50

Series 1, 2, and 3 were conducted with the 55 element Batch Core only. Series 4 included measurements with a 31 element core and a 19 element core in addition to the 55 element core. The average core burnup for the 31 and 19 element configurations were 6139 $\pm$ 58 and 6512 $\pm$ 61 MWd/MTM, respectively.

A detailed procedure was written for each test and approved by line management prior to the start of testing. Training sessions were conducted with the operating staff of PRTR to acquaint them with the planned test program and to answer questions about specific test requirements. The intensive preparations for the critical testing paid good dividends in test performance. All testing was accomplished by the normal operating staff, working from the test procedures, and in less than the scheduled time.

VI. EXPERIMENTAL RESULTS

A complete, sequential tabulation of the data obtained during the test program is contained in Appendix I. The results have been summarized according to the type of experiment and are presented in the following sections.

The main measurement result of importance is the core excess reactivity versus burnup. The measurement of the changing reactivity worth of a fuel element in Ring Three as a function of burnup, combined with destructive isotopic analyses, is also significant since the burnup of an element in this position should be approximately the same as the average burnup for the 55 elements of the Batch Core. This information as well as other data is being used for evaluation of reactor design methods. The results of the evaluations will be the topic of a separate report.

The moderator level coefficient measurements were crucial to the success of the entire Batch Core Experiment because all reactivity measurements were made in terms of differences in the height of the moderator at critical. Hence, these measurements were made quite extensively. The reactivity worth of boron is closely interrelated to the moderator level coefficient, and is thus also vital to the determination of core excess reactivity.

Reactivity effects of voids in either coolant or moderator were of great interest for evaluation of safety considerations made prior to the experiment. Also of interest for the same reason were the moderator temperature coefficient and kinetics measurements.

A. Moderator Level Coefficients of Reactivity

All reactivity measurements were made in terms of differences in the height of the moderator at critical. To interpret these differences in critical height in terms of reactivity, it was necessary to determine the reactivity worths of small increments of moderator height as a function of the moderator height, over the range of critical heights and boron concentrations in the moderator encountered during the test program. Therefore, the reactivity worths of at least three different increments of moderator height, ΔH_i, were determined from rising reactor period measurements each time a critical height measurement was completed. The moderator level coefficient of reactivity (MLCR) was derived from the experimental data by using the following relations:

$$(\mathrm{MLCR})_i = \frac{1}{\bar{k}} \frac{\Delta k_i}{\Delta H_i}$$

where Δk_i = excess reactivity determined from the rising period measurement with the moderator at H_i,

and $\Delta H_i = H_i - H_c$ where H_c is the moderator height at critical.

The value of $(\mathrm{MLCR})_i$ was assigned to the average moderator height,

$$\bar{H}_i = \frac{H_c + H_i}{2} .$$

It has been shown[4] that to first order, the moderator level coefficient of reactivity is proportional to the reciprocal of the cube of the moderator height, H^{-3}, in regions where the migration area, M^2, is constant. Therefore, the experimentally measured MLCR's were fitted with a function of the form

$$\text{MLCR} = A + BH^{-3}$$

using the least squares fitting code LEARN.[5,6] The value of M^2 in the region above the fuel is larger than that in the fuel region, so the slope B is different in the two regions. Data for boron concentrations of zero and 21 wppm ^{10}B in the moderator were fitted both above and below the top of the fuel. Since the MLCR is continuous for a given boron concentration, the function fitted over the fuel region and that fitted over the top reflector region were required to join at the fuel-top reflector interface.

The MLCR's derived from the fitted data for boron concentrations of zero and 21 wppm ^{10}B are shown in Figure 4 as a function of moderator height. The integrals of the MLCR curves are shown in Figure 5. All the reactivity differences measured during the test program have been interpreted through these latter curves. Interpolations were used for intermediate boron concentrations. In Figure 6, moderator level coefficients measured in interim critical test series 4 are compared with the fitted curves obtained in the initial critical tests.

B. Reactivity Worth of ^{10}B in the D_2O Moderator

As additional fuel elements were loaded into the core, the moderator height at critical decreased. When the critical height reached approximately 72-75 inches, boron was added to the moderator in quantities sufficient to raise the critical height again to the 100-inch range.

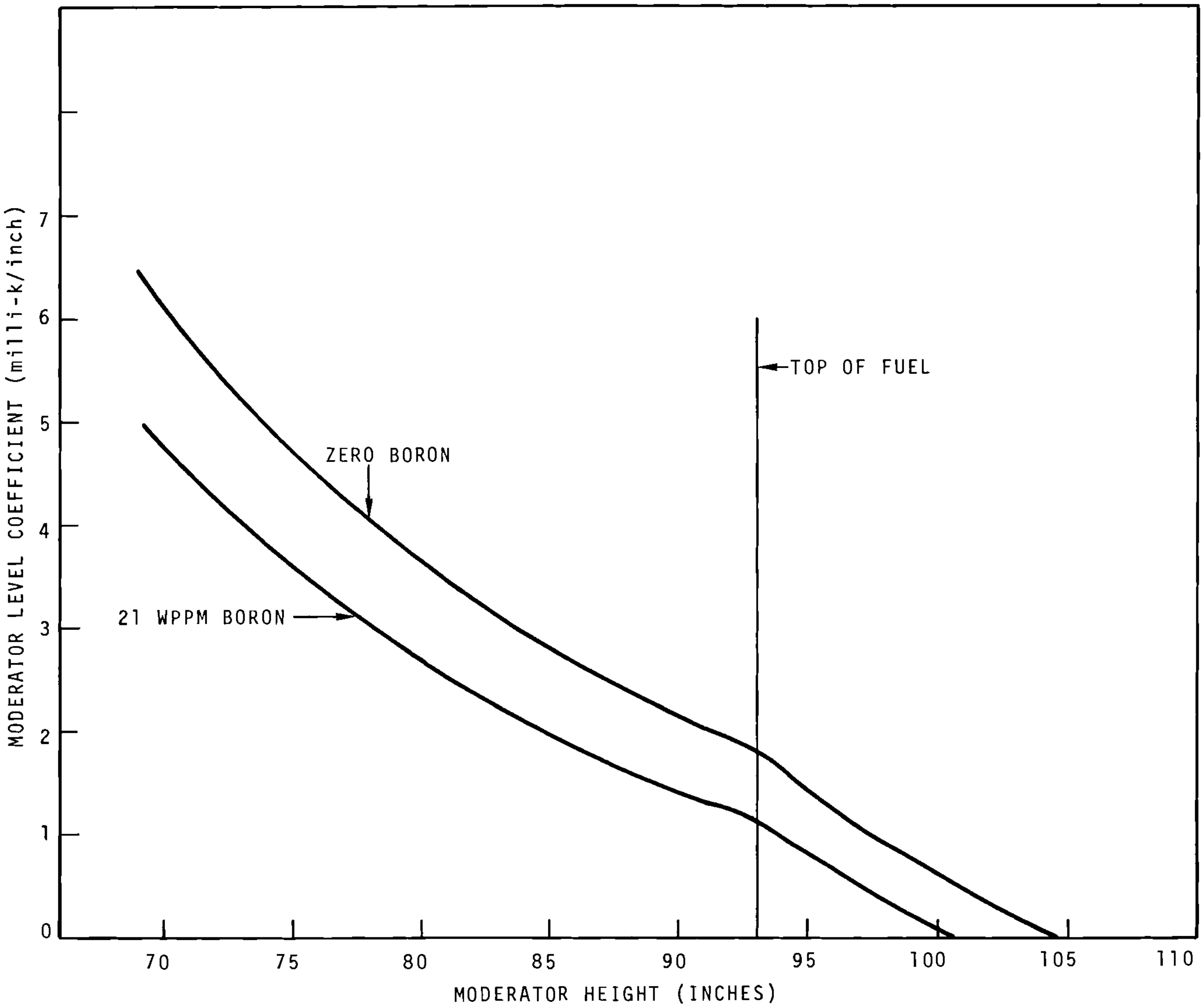


FIGURE 4. MODERATOR LEVEL COEFFICIENT OF REACTIVITY - PRTR BATCH CORE

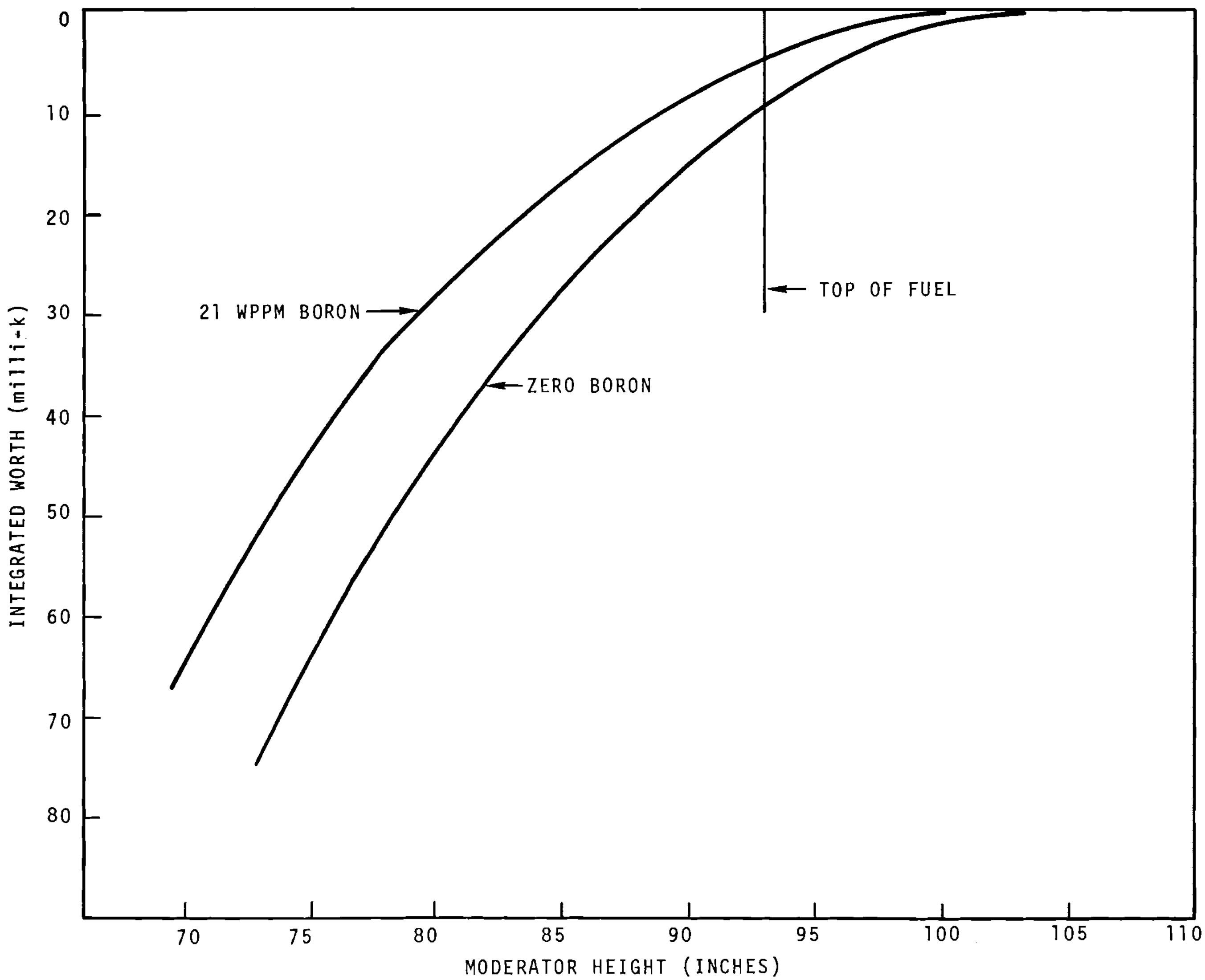


FIGURE 5. MODERATOR REACTIVITY WORTH - PRTR BATCH CORE

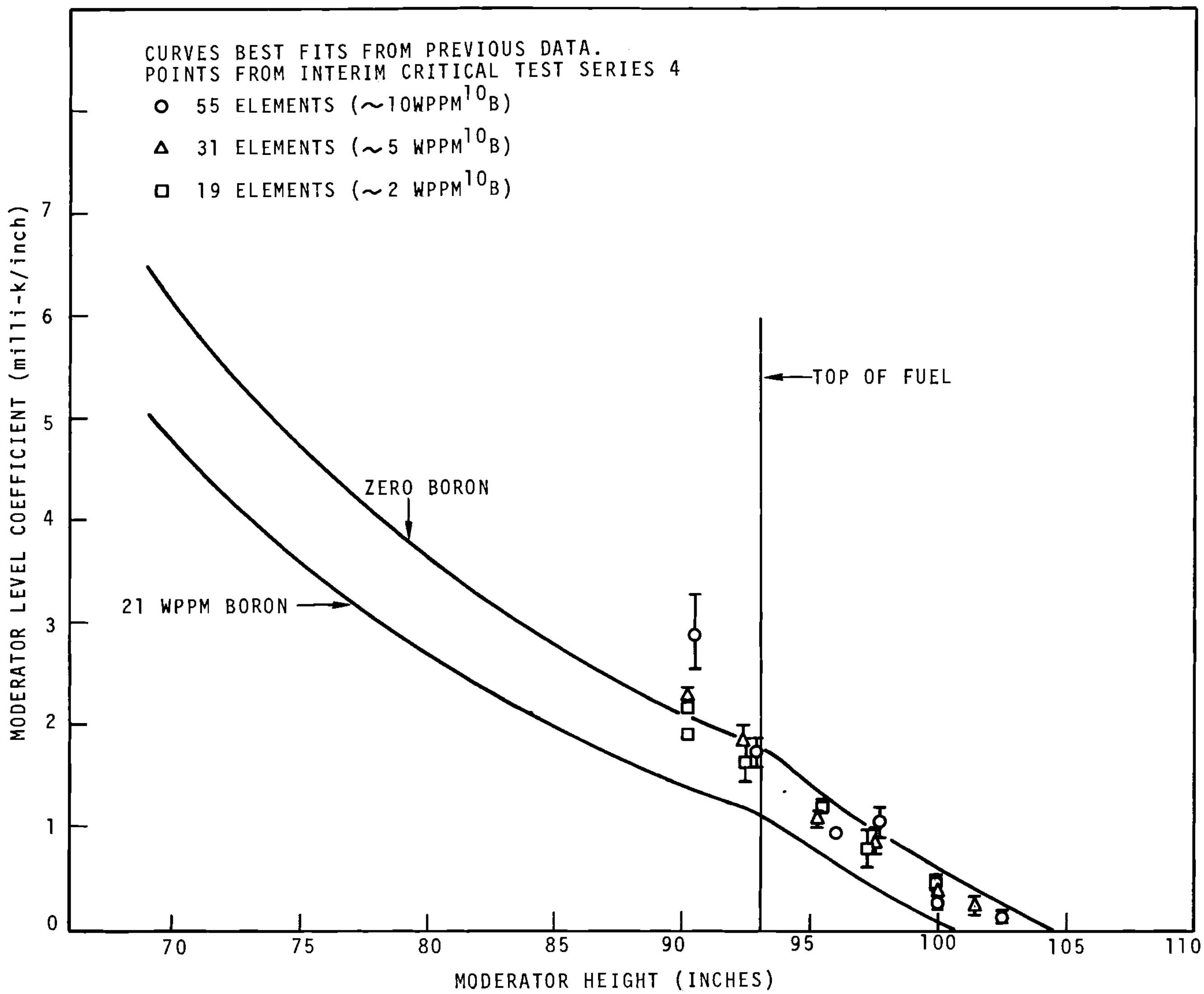


FIGURE 6. MODERATOR LEVEL COEFFICIENT OF REACTIVITY - DATA FROM PRTR BATCH CORE INTERIM CRITICAL TEST SERIES 4

The boron, enriched to 92% ^{10}B, was in the form of anhydrous boric acid dissolved in D_2O. Moderator samples, drawn after each boron addition, were saved for subsequent determinations of their boron content by reactivity comparisons using the Thermal Test Reactor (TTR). The reactivity worths of the boron additions were derived from the change in critical moderator height via the integrated moderator worth curve of Figure 5. The results are shown in Figure 7. The integrated boron worth (IBW) data were fitted by a polynomial of the form $IBW = A + BN + CN^2$ where N is the concentration of ^{10}B in D_2O, in parts per million by weight (wppm). From these data, it can be seen that the excess reactivity of the cold, clean 55-element core was 257 ± 32 milli-k or about $83.

The differential boron worth was determined at a number of boron concentrations, as shown in Figure 7. The horizontal flags on the data points indicate the span of the boron concentration change, with the data point plotted at the midpoint. The vertical flags indicate the uncertainty on the differential worth. The curve drawn through the differential data was based on the derivative of the polynomial fitted to the integrated data.

C. Neutron Flux Distributions

Each time the moderator height at critical was near the 100-inch level, for a new ^{10}B concentration, a set of eighteen gold foils was mounted on a lucite foil bar and inserted into a special thimble (location C-1553) for irradiation. The foils were 0.5 inch diameter and 0.006 inch

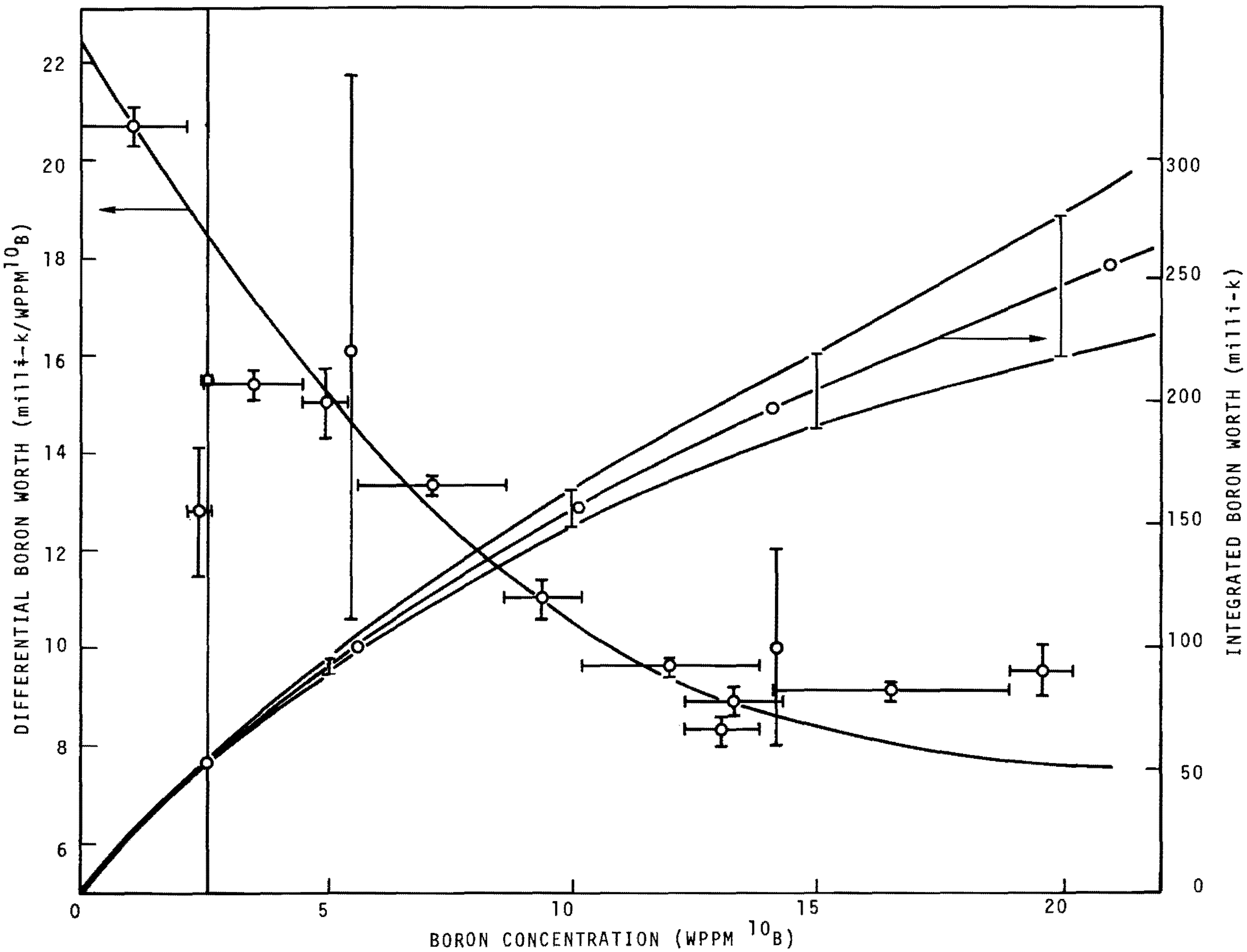


FIGURE 7. REACTIVITY WORTH OF BORON - PRTR BATCH CORE

in thickness, and were irradiated bare. Since the thimble was located in the moderator at a point equidistant from three adjacent fuel channels, the measured flux-shape was assumed to be typical of the moderator. The gold activities in each traverse have been normalized to the average value over the fuel length. The normalized gold activations from each of the irradiations performed at normal moderator heights (near 100 inches) and the coefficients of the cosine function fit to the data over the length of the fuel are shown in Figures 8 through 13. Another irradiation was performed after all fifty-five elements had been loaded but before more boron was added, with the critical moderator level at 72.2 inches. This measurement provided an indication of the power peaking to be expected in the fuel if the reactor were operated at power at low moderator levels. This latter measurement is compared with two of the measurements made at normal moderator heights in Figure 14.

A detailed flux traverse was made in the center cell (PC-1550) of the 55-element core using lutetium oxide foils, both bare and cadmium covered. Data were obtained from foils placed within fuel rods in a cluster (between mating faces of separable rods) and from foils immersed in the moderator, mounted on an aluminum positioning device inserted into a shim hole (see Figures 1 and 15). The cadmium covers used in the fuel rods were 0.030 inches thick, those used in the moderator were 0.020 inches thick.

The measured lutetium activations are shown in Figure 16. Due to the large gas gap between the shroud and process tubes in PC-1550, the flux shape in that region had to be inferred. Cadmium ratios calculated

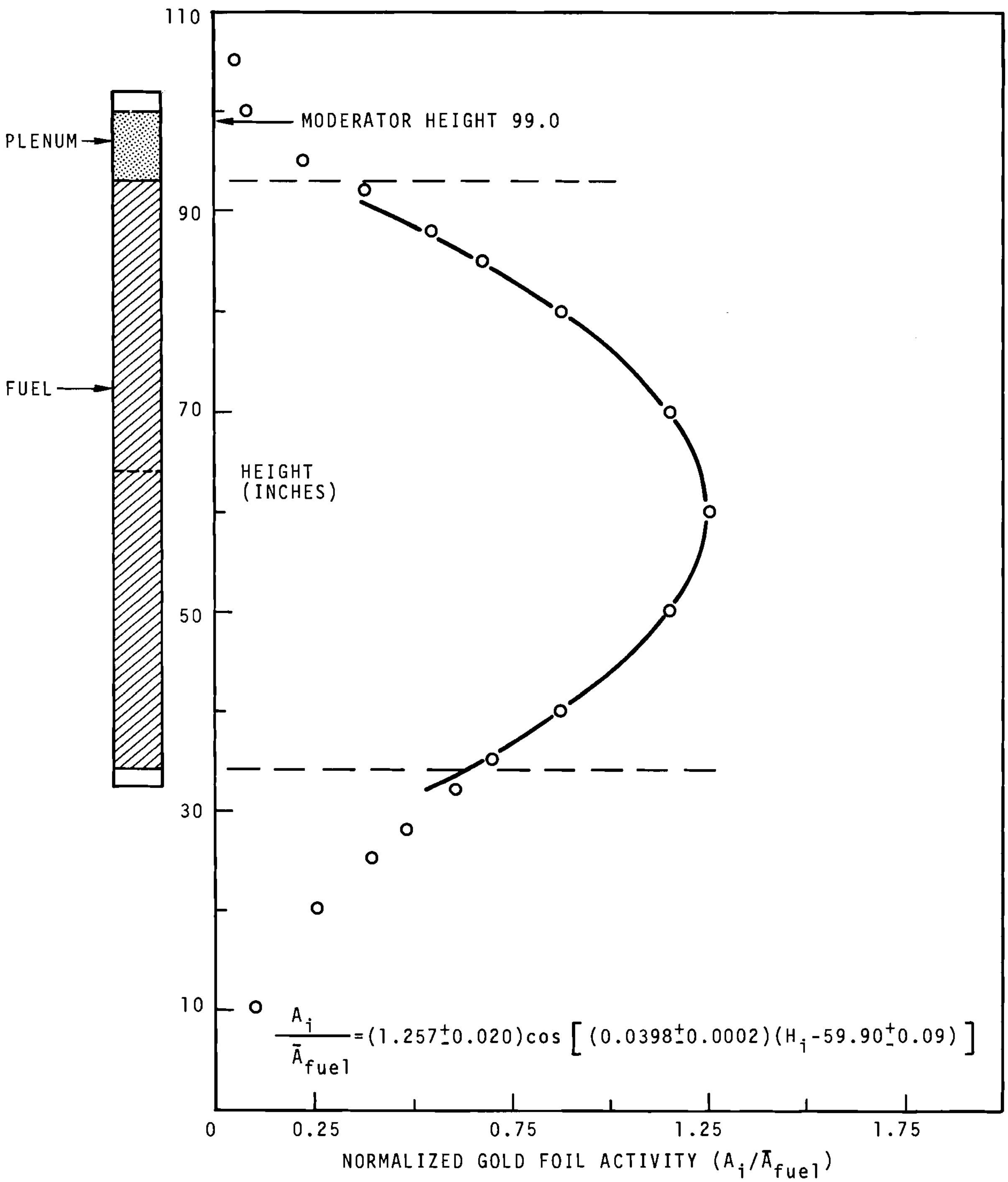


FIGURE 8. AXIAL GOLD FOIL TRAVERSE NINE ELEMENT CORE, ZERO BORON IN MODERATOR

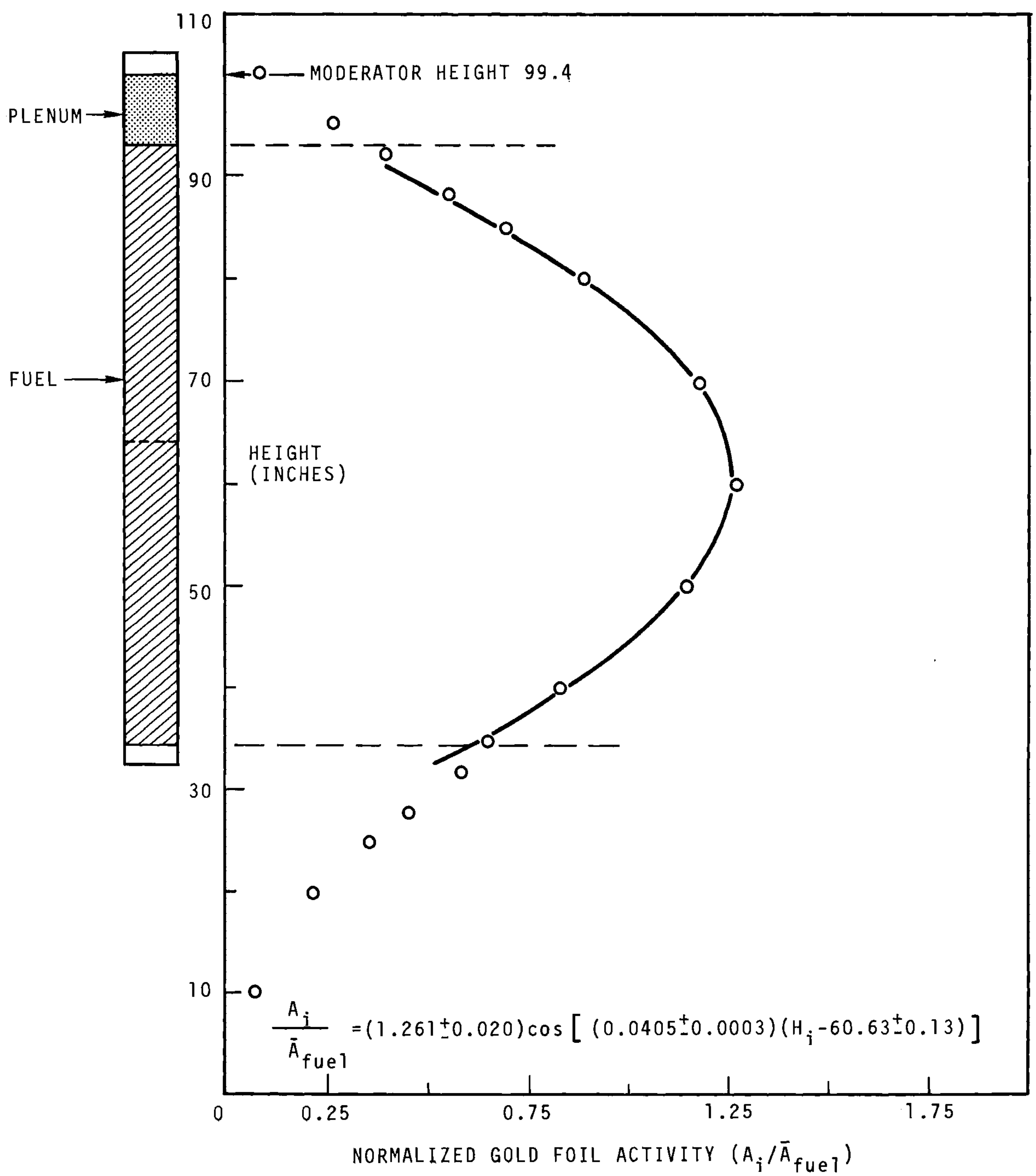


FIGURE 9. AXIAL GOLD FOIL TRAVERSE TWELVE ELEMENT CORE, 2.554 WPPM ^{10}B IN MODERATOR

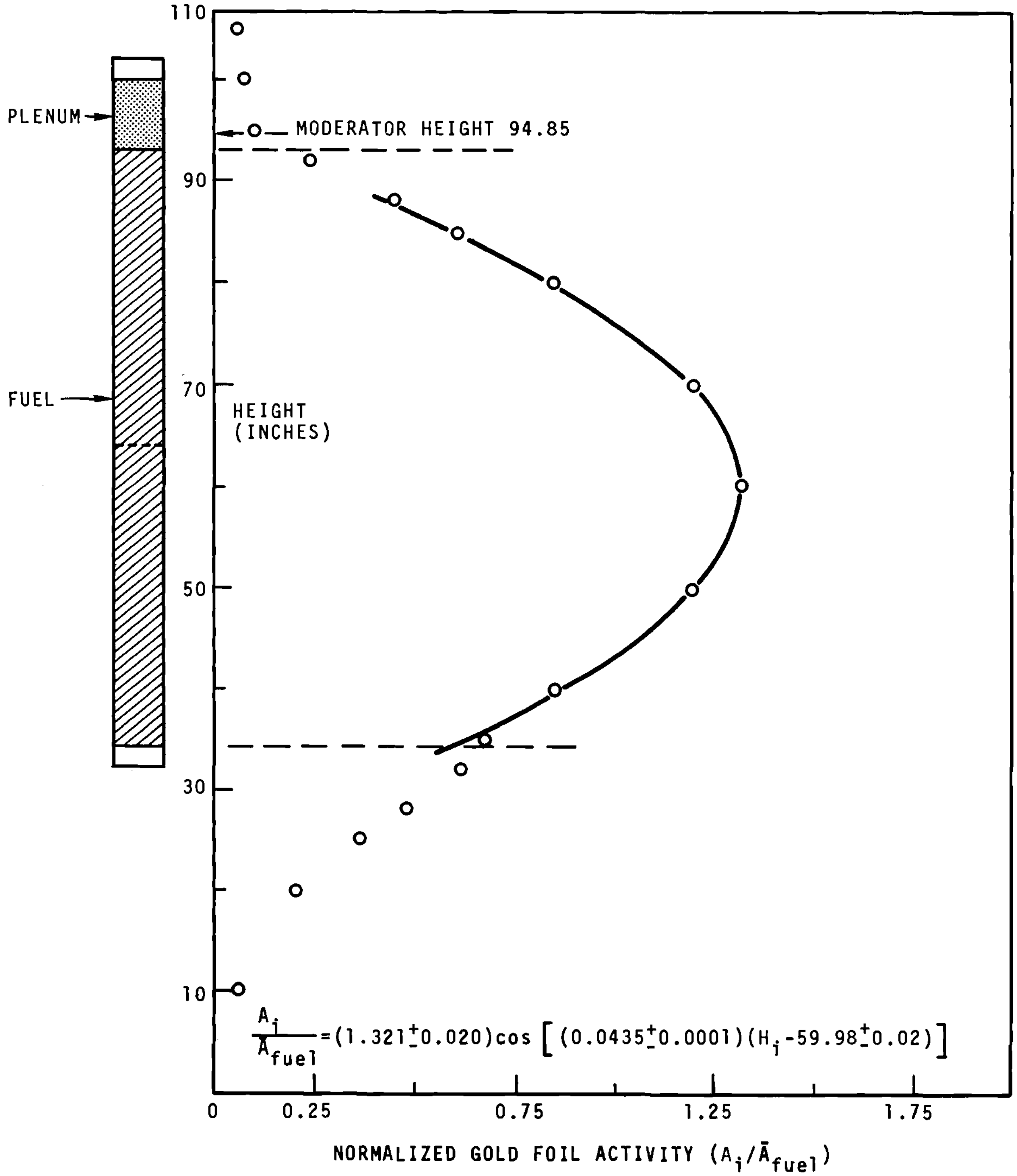


FIGURE 10. AXIAL GOLD FOIL TRAVERSE SIXTEEN ELEMENT CORE, 5.567 WPPM ^{10}B IN MODERATOR

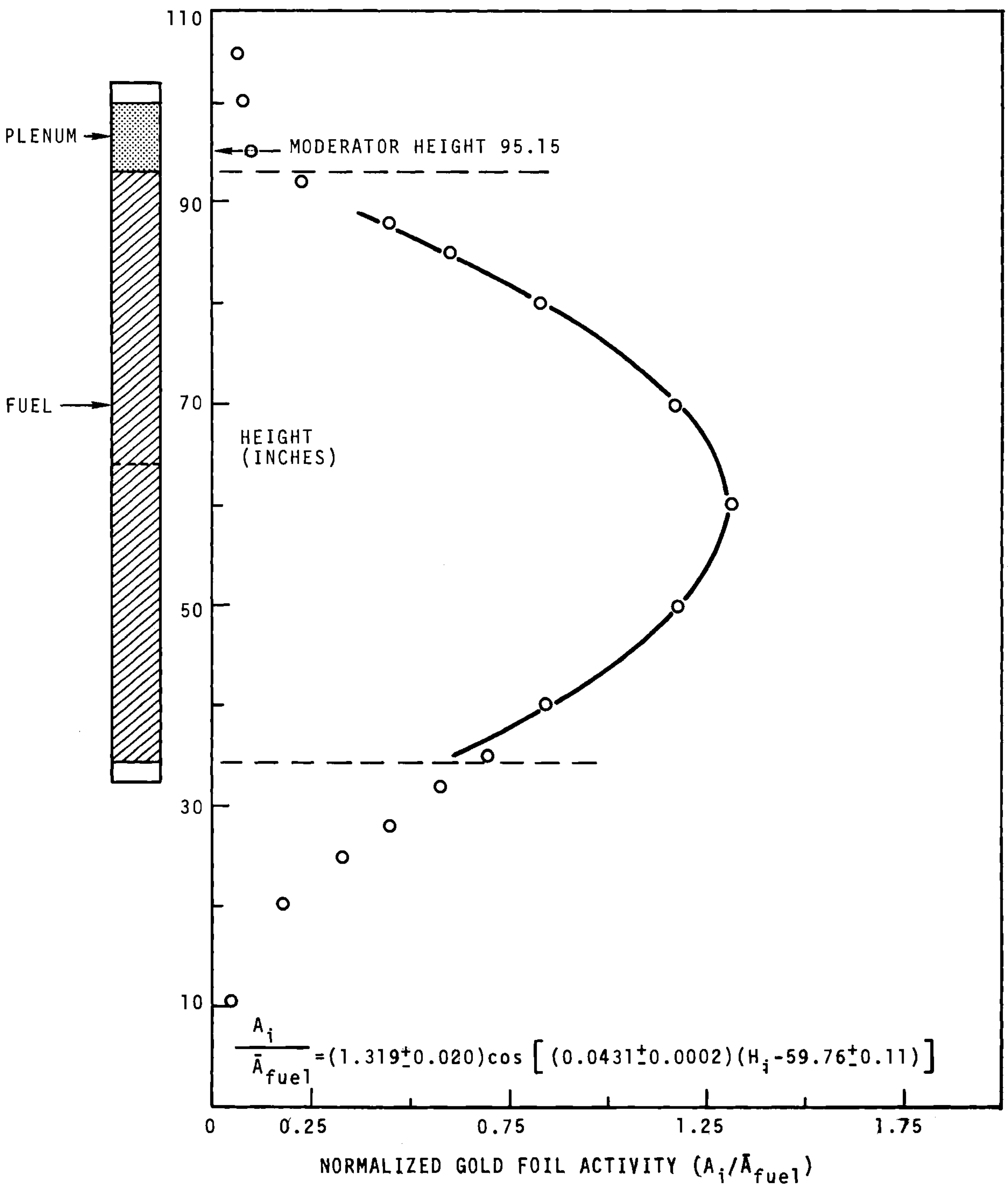


FIGURE 11. AXIAL GOLD FOIL TRAVERSE TWENTY-THREE ELEMENT CORE, 10.162 WPPM ^{10}B IN MODERATOR

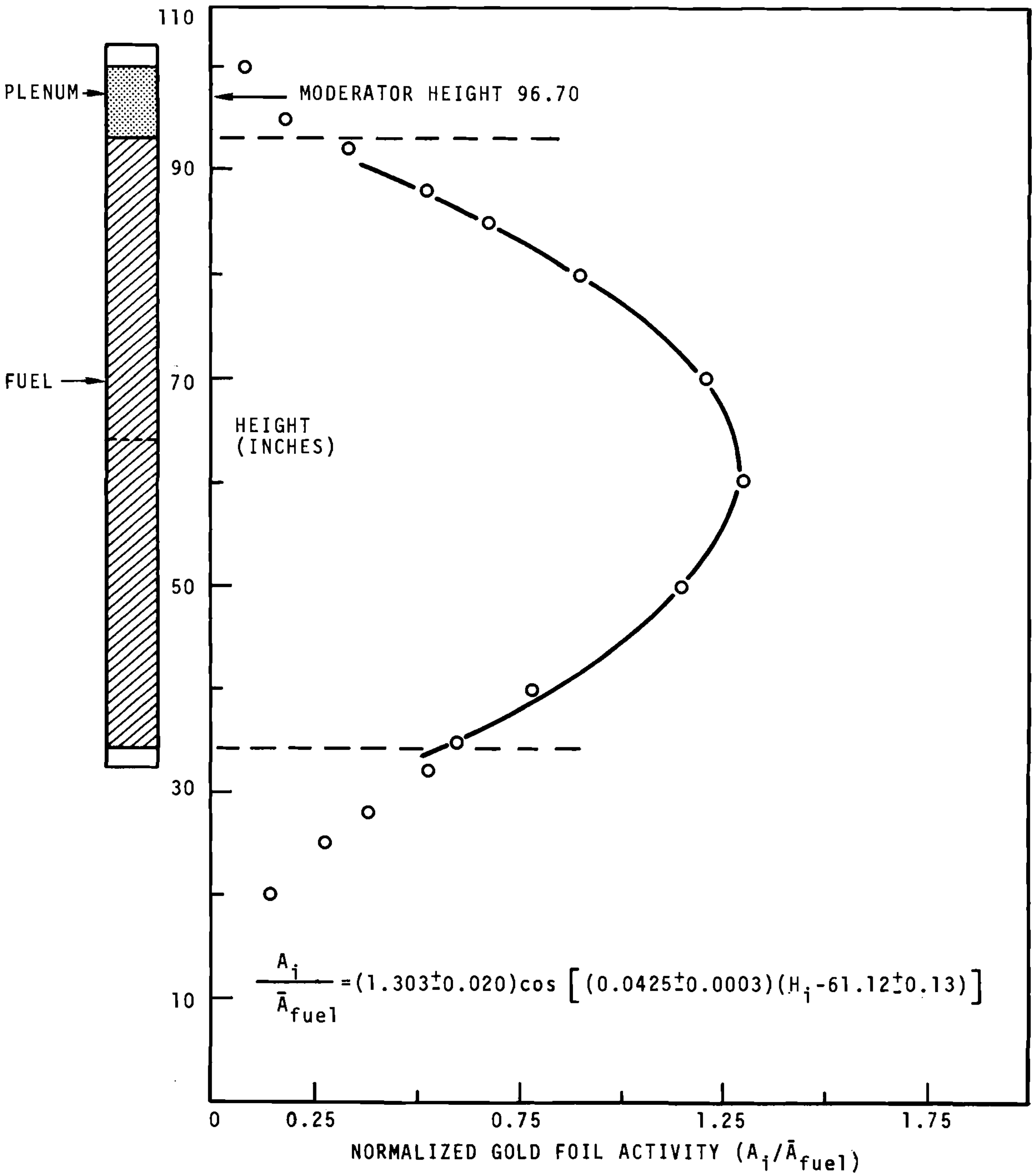


FIGURE 12. AXIAL GOLD FOIL TRAVERSE THIRTY-ONE ELEMENT CORE, 14.135 WPPM ^{10}B IN MODERATOR

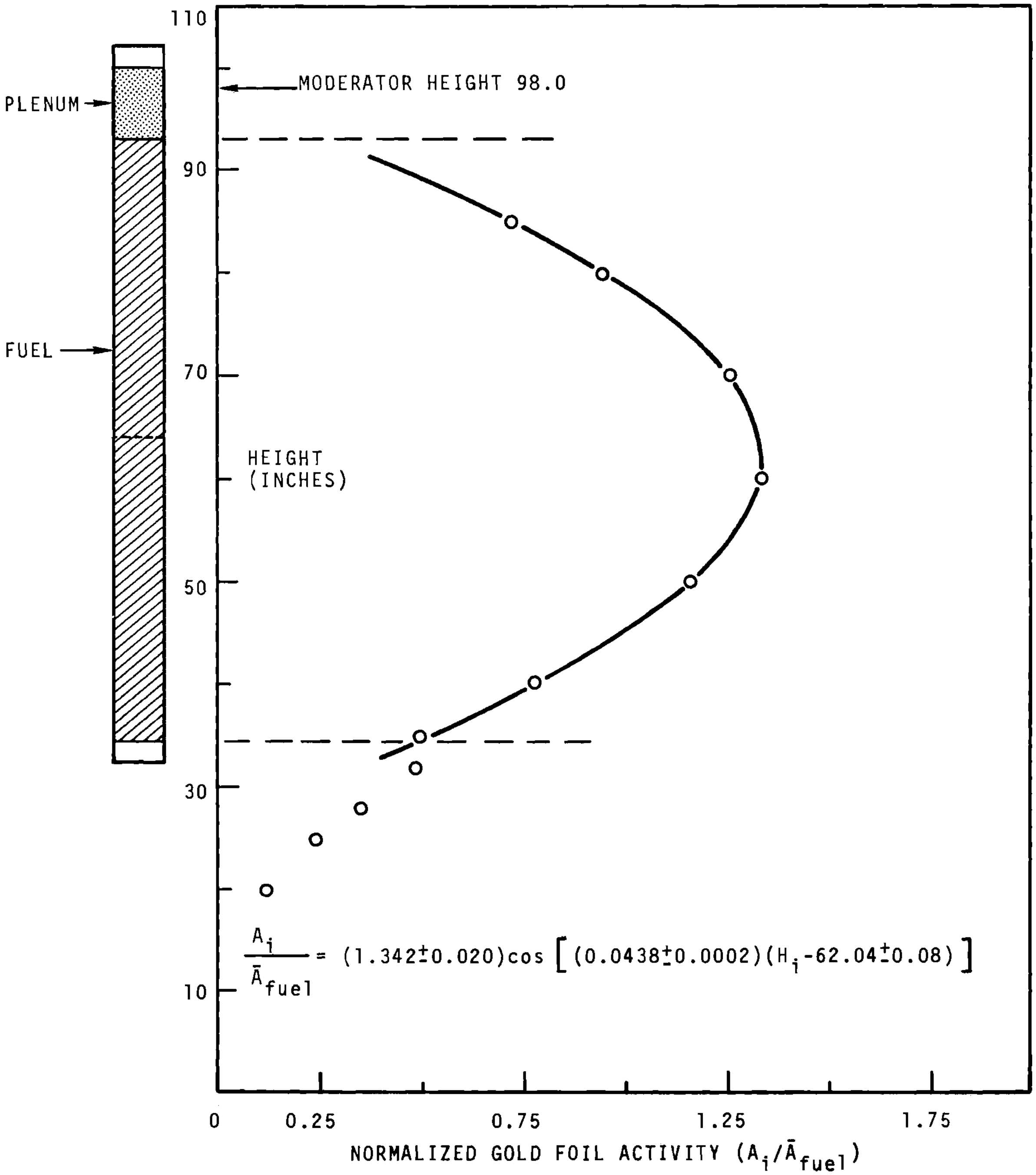


FIGURE 13. AXIAL GOLD FOIL TRAVERSE FIFTY-FIVE ELEMENT CORE, 20.976 WPPM ^{10}B IN MODERATOR

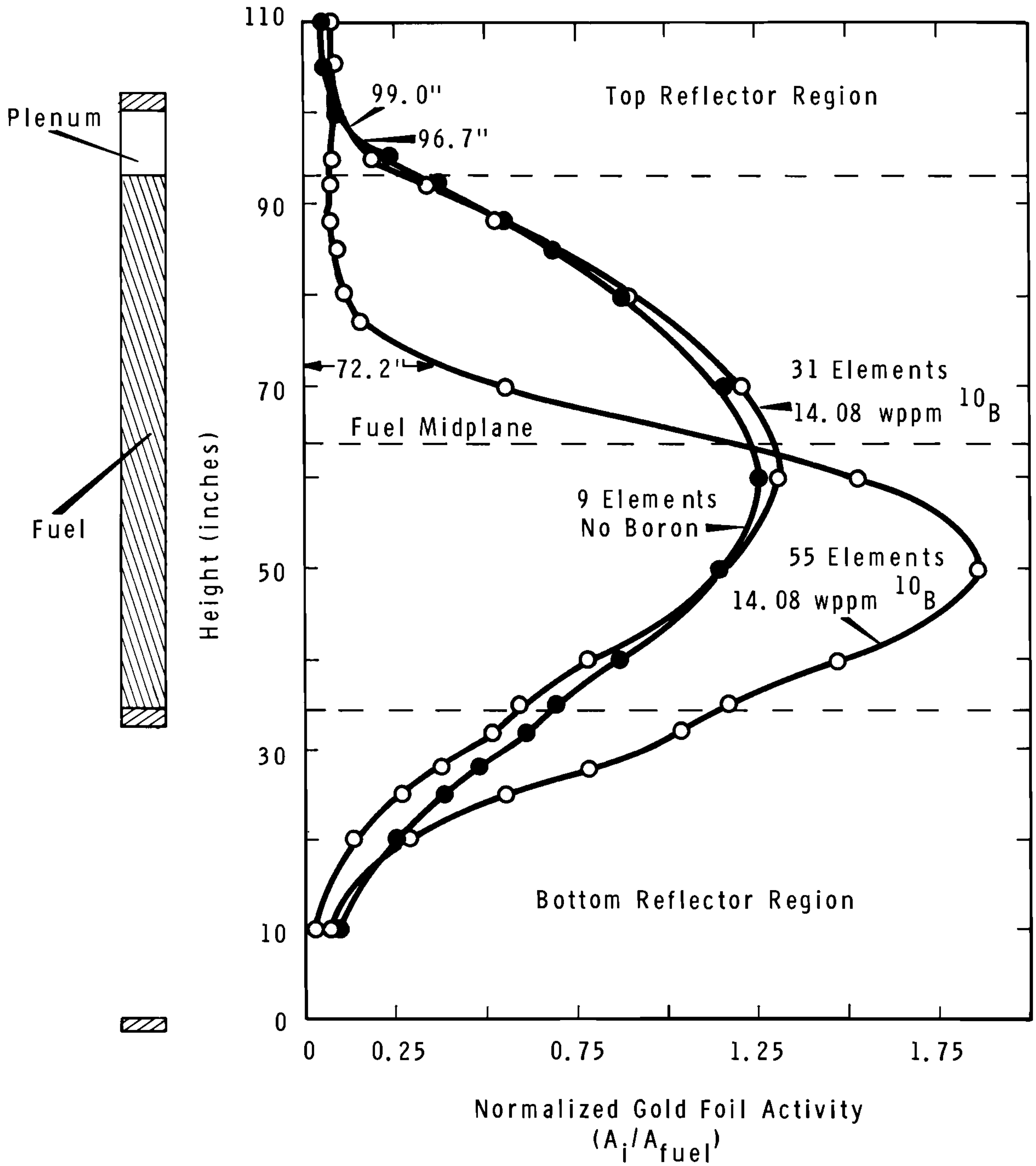


FIGURE 14. COMPARISON OF THERMAL NEUTRON FLUX DISTRIBUTIONS AS A FUNCTION OF MODERATOR LEVEL FOR DIFFERENT CORE SIZES AND ^{10}B CONCENTRATIONS

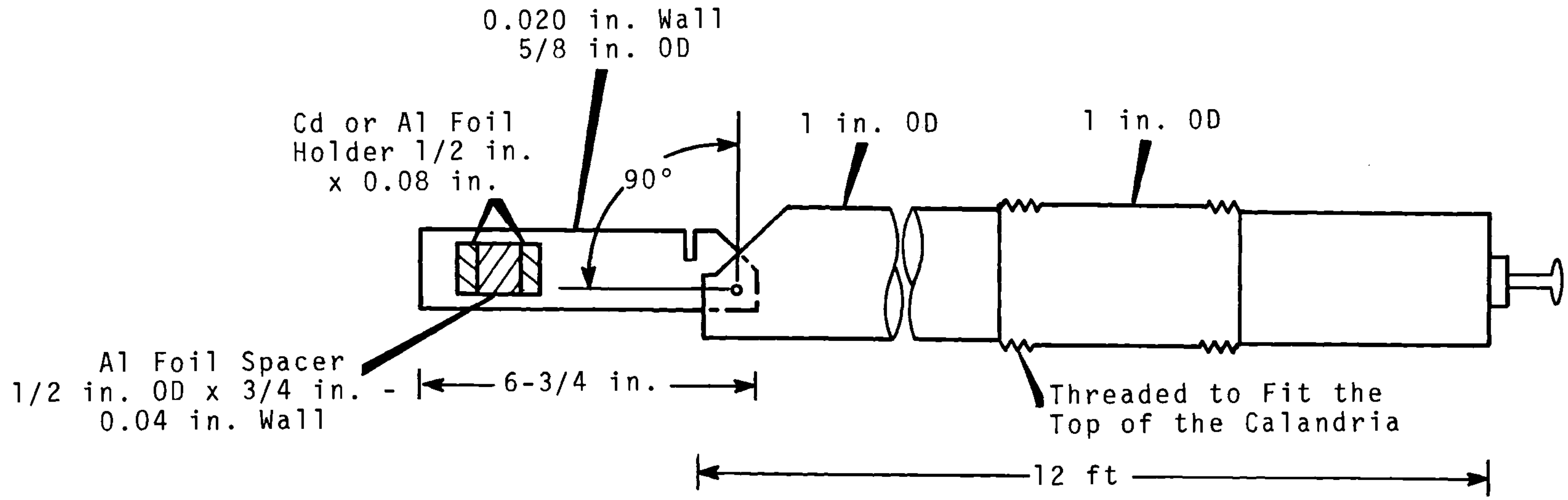


FIGURE 15. CELL TRAVERSE FOIL HOLDER

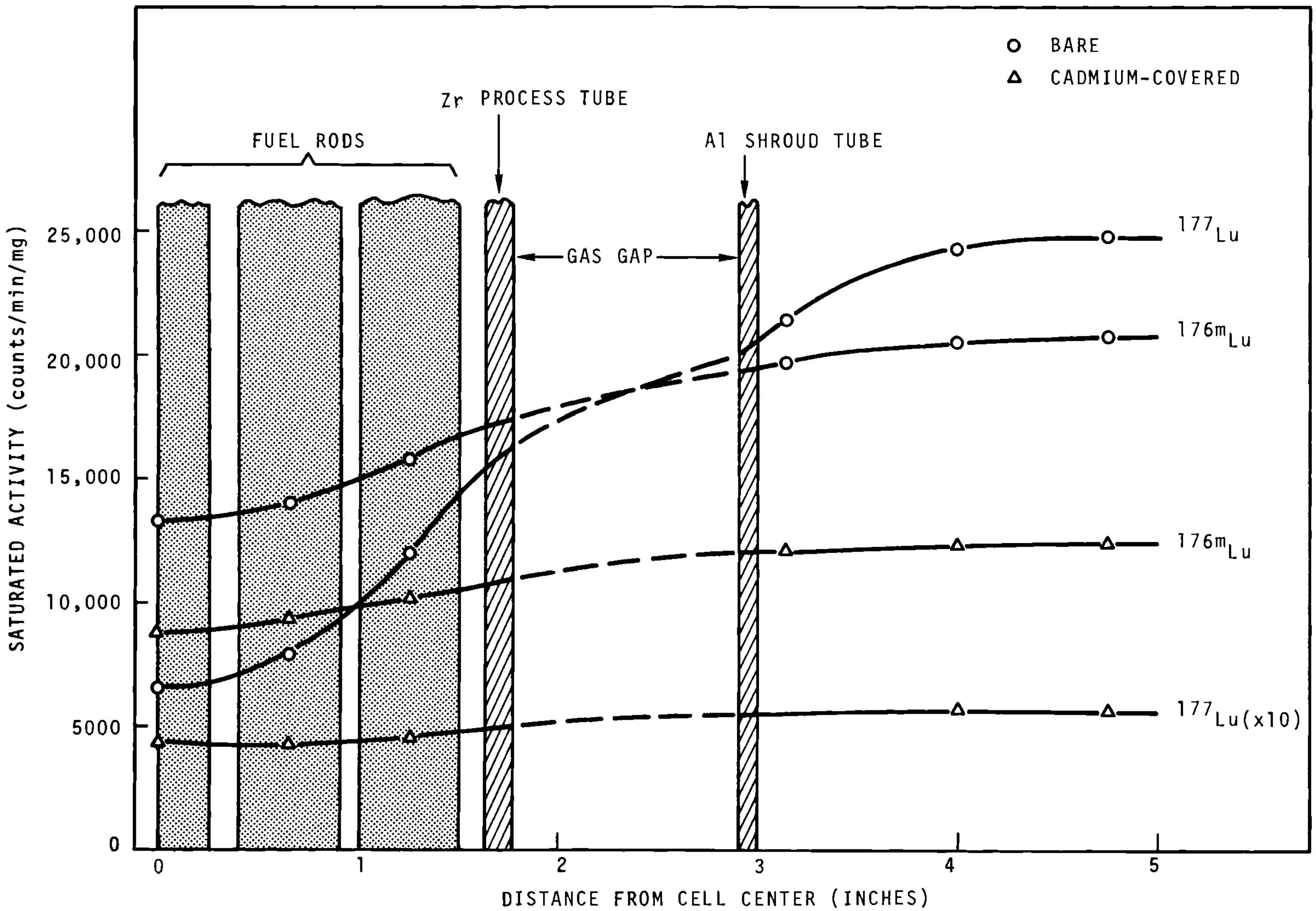


FIGURE 16. LUTETIUM CELL TRAVERSE FIFTY-FIVE ELEMENT CORE, 20.976 WPPM ^{10}B IN MODERATOR

from the lutetium foil data for the fuel and moderator regions are listed in Table II.

Table II

Cadmium Ratios For Lutetium

Radius (in.)	Cadmium Radio ^{175}Lu	^{176}Lu
In Fuel	0.030" Cd	
0.00	1.089 ± 0.003	10.92 ± 0.05
0.64	1.095 ± 0.003	13.78 ± 0.07
1.25	1.128 ± 0.003	19.33 ± 0.09
In Moderator:	0.020" Cd	
3.14	1.191 ± 0.003	24.16 ± 0.13
3.99	1.208 ± 0.003	31.77 ± 0.17
4.86	1.209 ± 0.003	32.03 ± 0.13

D. Reactivity Change Due to Loss of Coolant

The change in critical moderator height associated with the removal of coolant from selected process tubes was determined for a variety of core sizes and boron concentrations. In particular, this effect was measured in the central channel, 1550, for removal of D_20 and H_20 coolant each time the critical moderator height was in the 95-100 inch range during the loading sequence. The effect of removing D_20 coolant from all eighty-five tubes was measured with the 31 element loading and with the 55 element loading. The influence of the coolant in the unfueled peripheral tubes was measured for the 55-element loading by keeping the unfueled tubes full of D_2O and determining the critical heights with the 55 fueled channels either dry, cooled with D_2O or cooled with H_2O. Additional single column coolant removal measurements were made in channels 1651 and 1546. The isotopic composition of the coolant was varied from the nominal primary grade (97% D_2O) to approximately 94% D_2O, to approximately 60% D_2O and finally to H_2O. Within the experimental uncertainties (± 0.1 mk), there was no reactivity difference among 97%, 94% or 60% D_2O, so these results are not reported here.

The results of the coolant worth measurements are listed in Tables III, IV and V. The single channel voiding measurements in 1550 showed a positive coolant reactivity worth under all conditions, with the H_2O consistently larger than the D_2O. The magnitudes of the worths decreased as the number of fueled channels increased and the statistical weight of the test channel decreased. The results are consistent with the fact that channel 1550 is undermoderated, compared with the rest of the channels in the core. Because of the oversize shroud tube (6.00 in. O.D.) the amount of moderator associated with channel 1550 is 34% less than the amount associated with each of the remaining 84 channels which have 4.25 in. O.D. shroud tubes.

Measurements made in single channels in Ring 1 (1651) and Ring 2 (1546) showed some trends which would imply a positive worth for H_2O for the fueled core and a negative worth for D_2O. However, the measurements made with the entire core showed otherwise. The worth of D_2O coolant was slightly positive (+0.6 milli-k) whereas the worth of H_2O coolant was strongly negative (-4.8 milli-k), with the unfueled tubes filled with D_2O. When the unfueled tubes were drained and filled along

Table III

Coolant Worth Measurements in Channel 1550

^{10}B Core (wppm)	Fueled Channels	Reactivity Change in Milli-k Dry	D_2O	H_2O
		0.0	+1.2 ± 0.1	+5.7 ± 0.1
0.0	9	(100.25)	(98.00)	(94.20)
2.6 ± 0.1	12	0.0	+1.4 ± 0.1	+5.6 ± 0.1
		(102.65)	(99.40)	(95.04)
5.6 ± 0.1	16	0.0	+0.9 ± 0.1	+4.2 ± 0.1
		(95.67)	(94.85)	(92.52)
10.2 ± 0.1	23	0.0	+0.4 ± 0.1	+2.7 ± 0.1
		(95.86)	(95.39)	(93.41)
14.1 ± 0.1		0.0	+0.03± 0.10	+3.0 ± 0.1
		(96.93)	(96.90)	(94.87)
20.9 ± 0.1	31	0.0	+0.03± 0.10	+1.2 ± 0.1
		(98.40)	(98.30)	(96.35)

( ) Numbers in parenthesis are critical heights

Table IV

Single Channel Coolant Worth Measurements*

Fuel Channel Number	Radial Position (cm)	Reactivity Change in Milli-k: Dry	D_2O	H_2O
1550	0.0	0.0	+0.03 ± 0.10	+1.20 ± 0.10
		(98.40)	(98.30)	(96.35)
1651	20.32	0.0	-0.15 ± 0.10	+0.30 ± 0.10
		(95.25)	(95.45)	(94.87)
1546	35.20	0.0	-0.08 ± 0.10	+0.12 ± 0.10
		(95.92)	(96.04)	(95.75)

* 55 fueled channels, 20.95 wppm ^{10}B in D_2O moderator.

( ) Numbers in parentheses are critical heights.

Table V

Core Coolant Worth Measurements

^{10}B Conc. (wppm)	Fueled Channels	Reactivity Change in Milli-k: Dry	D_2O	H_2O
12.4 ± 0.1	31	0.0	+17.6 ± 0.1	-----
		(100.40)	(85.92)	-----
18.6 ± 0.1	55	0.0	+7.7 ± 0.1	-----
		(89.77)	(85.29)	-----
*19.6 ± 0.1	55	0.0	+0.6 ± 0.1	-4.8 ± 0.1
		(91.56)	(91.10)	(96.05)

( ) Number in parentheses are critical heights.

* Unfueled channels remained filled with D_2O.

with the fueled tubes, the D_2O worth was strongly positive, showing a marked influence from the unfueled tubes.

E. Moderator Void Worth Measurements

The reactivity effect of void formation in the moderator was measured by determining the change in critical moderator height when the thimble located in C-1352, which was normally dry, was filled with borated D_2O from the moderator system.

The aluminum thimble had an I.D. of 2.124 inches with a wall thickness of 0.063 inches and extended through the calandria vertically. Void measurements of this type were made in the 31 and 55 element cores. Details of the measurements are presented in Table VI. It can be seen from the results that the effect of creating a void in the borated D_2O moderator was negative for all boron concentrations tested.

F. Moderator Temperature Coefficient Measurements

The temperature coefficient of reactivity of the moderator system was measured at several times during the burnup of the Batch Core. The result of the initial measurement has been reported previously.(7) This coefficient was measured by allowing the moderator temperature to increase approximately 60°F (from ∿80°F to ∿140°F) in about two hours. During this time the reactor operated at a constant power in the 7-10 MW range so that the primary coolant temperature remained constant.

The results of the measurements are shown in Figure 17, with temperature coefficient plotted as a function of boron concentration. The temperature coefficient was positive for all the boron concentrations at which it was measured. From the data, it appears that this coefficient would become negative at about 6 wppm ^{10}B.

Table VI

Moderator Void Worth Measurements

# F. E.	^{10}B Conc. (wppm)	Critical Height (inches)	Thimble Condition	Reactivity Change (Milli-k)	Core Void Fraction*	Void Coefficient (mk/% Void)
31	14.14 ± 0.05	95.82 ± 0.05	Filled	—	1.055×10^{-2}	—
31	14.14 ± 0.05	96.70 ± 0.05	Dry	-0.75 ± 0.10	1.055×10^{-2}	-0.71 ± 0.10
55	20.98 ± 0.05	97.24 ± 0.05	Filled	—	6×10^{-3}	—
55	20.98 ± 0.05	97.55 ± 0.05	Dry	-0.14 ± 0.10	6×10^{-3}	-0.23 ± 0.17

* Core void fraction = thimble volume/moderator volume of fueled core.

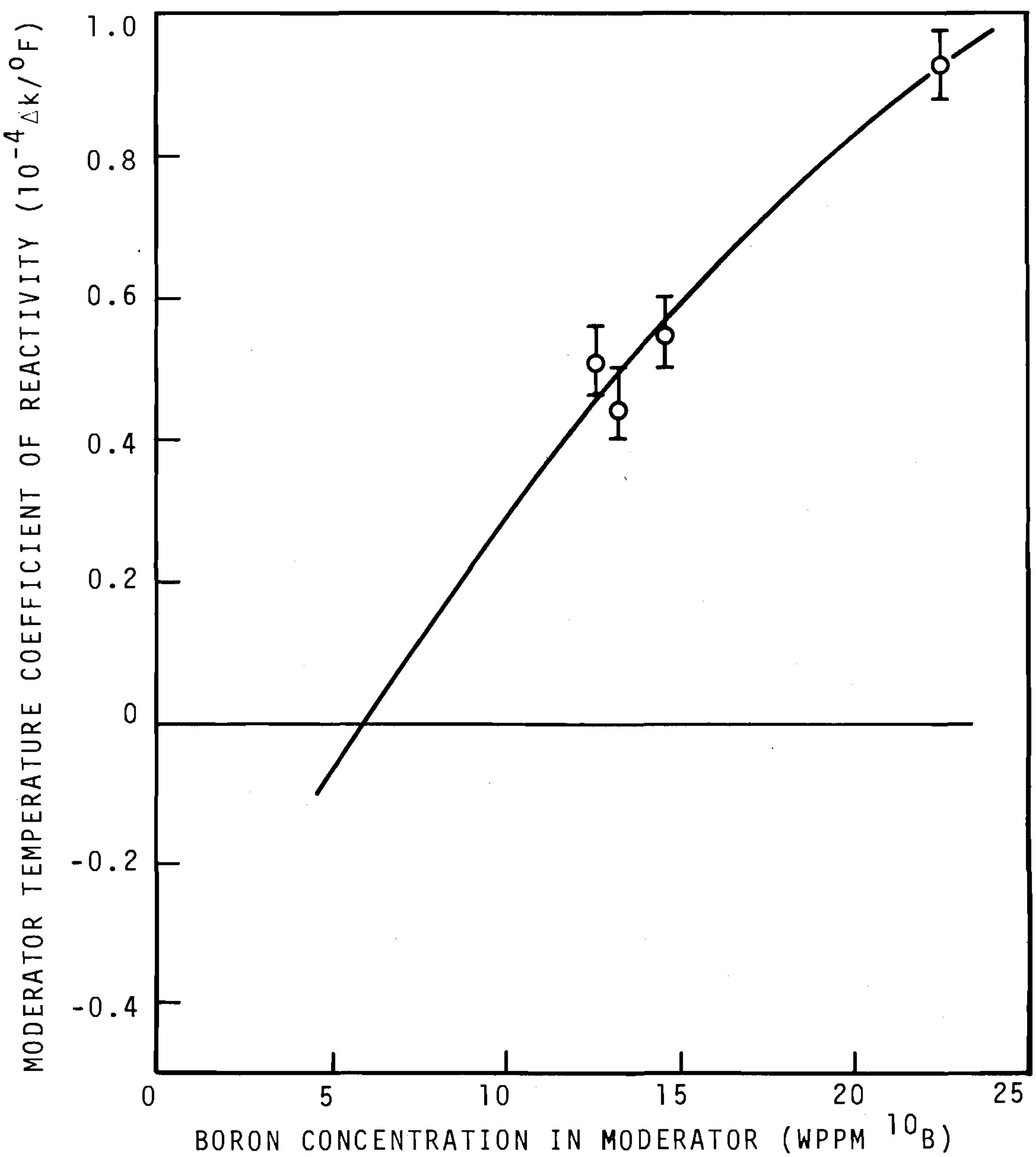


FIGURE 17. MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

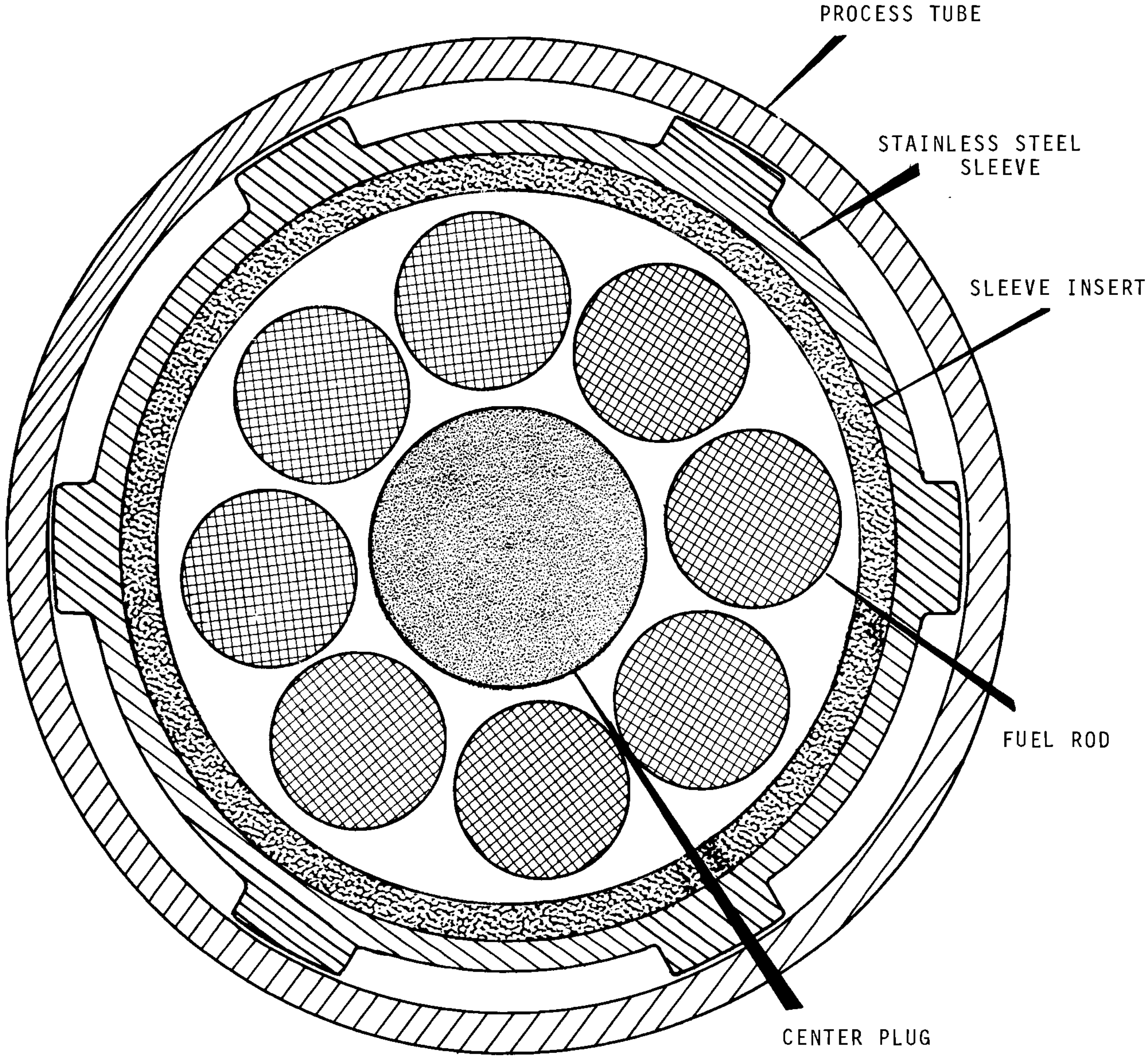
G. FERTF Element Measurements

A series of measurements was performed to determine the characteristics of the several possible configurations of a test assembly designed for use in the Fuel Element Rupture Test Facility (FERTF), which occupied channel 1550 during the Batch Core burnup. The test assembly, with its possible variations, is illustrated in Figure 18. The initial configuration contained six rods spaced equally. This design was later changed to contain eight equally spaced rods as shown in the figure.

The core loading during these measurements contained fifty-four standard Mark I-R 19-rod clusters of UO_2-2 wt% PuO_2 in D_2O coolant, with a nominal 21 wppm ^{10}B in the D_2O moderator.

Measurements were made of the reactivity change caused by a loss of coolant in the FERTF (PC-1550) under a variety of conditions. At the same time, low-power irradiations were conducted and rods from the 6-rod FERTF fuel assembly and from the outer ring of a 19-rod cluster located in Ring 1 were gamma-scanned to obtain relative rod powers in the two assemblies. The results of these measurements are presented in Table VII.

The initial measurements showed that the six-rod assembly was not acceptable for two reasons: first, the reactivity increase resulting from a loss of coolant incident exceeded allowable limits; and second, the rod powers in the assembly would restrict reactor operation. The redesigned assembly had much of the H_2O volume displaced with absorbing materials, such as the sleeve insert and the center plug, and with two additional fuel rods. The reactivity increase resulting from a loss of coolant incident following modification was found to be about 50¢,



PROCESS TUBE	3.25	3.56	ZIRCALOY-2
SLEEVE	2.750	3.165-RIBS 2.980-SLEEVE	304-SS,Zr-2, HASTELLOY-X
SLEEVE INSERT	2.503	2.710	304-SS,Zr-2, HASTELLOY-X
ROD RACK SUPPORT TUBE	0.875	0.973	304-SS
CENTER PLUG	-	0.850	304-SS,Zr-2, HASTELLOY-X

FUEL RODS ARE SPACED EQUALLY ON A CIRCLE OF 2.5 INCH DIAMETER.

FIGURE 18. FERTF TEST ASSEMBLY: 8 ROD ELEMENT

Table VII

Measurements in The FERTF Test Assembly

Fuel Type	No. of Rods	Sleeve	Insert	Center Plug	Measured Power Ratio*	H_2O to Void Worth (Milli-k)
No Fuel	—	—	—	—	—	+8.1 ± 0.2
UO_2	6	304 SS	—	—	0.62 ± 0.05	+4.7 ± 0.2
UO_2-3 wt% PuO_2	6	304 SS	—	—	1.76 ± 0.03	—
UO_2-2 wt% PuO_2	6	304 SS	—	—	1.53 ± 0.01	+1.3 ± 0.2
UO_2-2 wt% PuO_2	8	304 SS	—	304 SS	0.95 ± 0.03	—
UO_2-2 wt% PuO_2	8	304 SS	—	Hastelloy-X	0.95 ± 0.03	+1.4 ± 0.2
UO_2-2 wt% PuO_2	8	304 SS	304 SS	304 SS	0.89 ± 0.04	—
UO_2-2 wt% PuO_2	8	304 SS	304 SS	Hastelloy X	0.90 ± 0.06	—

$$^{*}\text{Power Ratio} = \frac{\text{(Power at maximum point on maximum rod in FERTF element)}}{\text{(Power at maximum point on maximum rod in Ring 1 element)}}$$

within acceptable bounds. Addition of the absorbing sleeve insert and plug reduced the rod powers in the assembly to levels that would not limit reactor operation.

H. Fuel Element Worths-Fuel Substitution Measurements

1. Initial Critical Tests - The reactivity worth of each new fuel element added on the periphery of the core was determined from the change in the critical moderator height following each fuel addition. The measurements were made at various boron concentrations as the reactor was loaded from the initial 9-element, boron-free critical core to the final 55-element, borated core. The results of these measurements are shown in Figure 19, with the reactivity worth of each peripheral element plotted versus the total number of elements in the core. The concentrations of boron in the moderator during the loading steps are also shown. As the loading progressed, two or three elements were charged in one loading increment. The number of elements changed per increment is indicated by the number of points on each horizontal in Figure 19. The reactivity worths plotted are the average worths per element for each incremental fuel addition.

Measurements were made of the reactivity worth of several special fuel elements, relative to the standard UO_2-2 wt% PuO_2 elements. These included: pelleted fuel (UO_2-1.94 wt% PuO_2, density adjusted to match the total plutonium per element of the standard vipac fuel); and elements which had instrumentation installed for monitoring the pressure and temperature of the gases in the gas plenum at the top of two rods in the outer ring of the 19-rod clusters. No differences in the reactivity

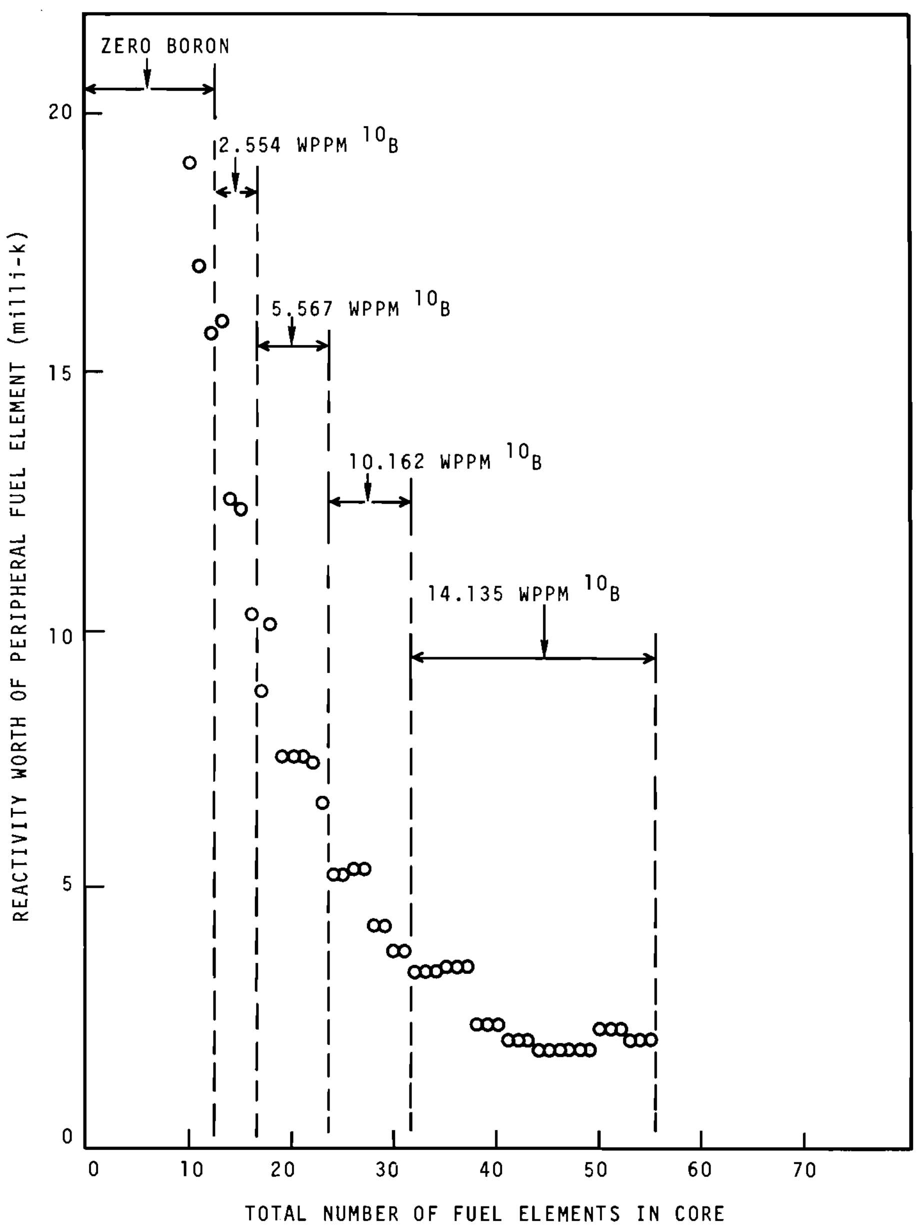


FIGURE 19. REACTIVITY WORTH OF FUEL ELEMENTS

worths were detectable among the various special elements relative to the standard elements.

The worth of a standard UO_2-2 wt.% PuO_2 fuel element was measured relative to a process tube filled with primary grade (~97%) D_2O in PC-1748, to provide a base point for measurements to be made at each of the interim critical tests during the burnup of the batch core. The worth of the fuel element number 6069 was determined to be 6.4 milli-k in the 55-element core with 19.24 wppm ^{10}B in the D_2O moderator.

2. Interim Critical Tests - Throughout the Batch Core Experiment the burnup of the fuel elements in ring three of the core was approximately the same as the average burnup for the 55 elements of the Batch Core. Measurement of the reactivity worth of a ring three element at each burnup step provided an additional means of determining reactivity changes in the Batch Core.

The reactivity worth of FE-6069, a standard UO_2-2 wt% PuO_2 fuel element, relative to the D_2O-filled channel located in PC-1748, was determined during the initial critical tests. It was determined again in series 1, 2, and 3 of the interim critical tests. In addition, the reactivity worth of FE-6506, a standard UO_2-2 wt% PuO_2 fuel element with zero burnup, was measured in the same channel in series 1, 2 and 3 of the interim critical tests. The results of all of these measurements are tabulated in Table VIII and shown in graphical form in Figure 20.

Table VIII

Reactivity Worth of Fuel Element in Ring Three

Batch Core Burnup (MWd/MTM)	^{10}B in Moderator (wppm)	Fuel Element	F. E. Exposure (MWd/MTM)	F. E. Worth (milli-k)
0	19.24 ± 0.05	6069	0	6.4 ± 0.1
1650 ± 16	16.71 ± 0.05	6069	2064 ± 30	5.9 ± 0.1
3560 ± 34	13.75 ± 0.05	6069	4473 ± 65	5.3 ± 0.1
4930 ± 47	11.85 ± 0.05	6069	6210 ± 90	4.9 ± 0.1
1650 ± 16	16.71 ± 0.05	6506	0	7.1 ± 0.1
3560 ± 34	13.75 ± 0.05	6506	0	7.2 ± 0.1
4930 ± 47	11.85 ± 0.05	6506	0	7.2 ± 0.1

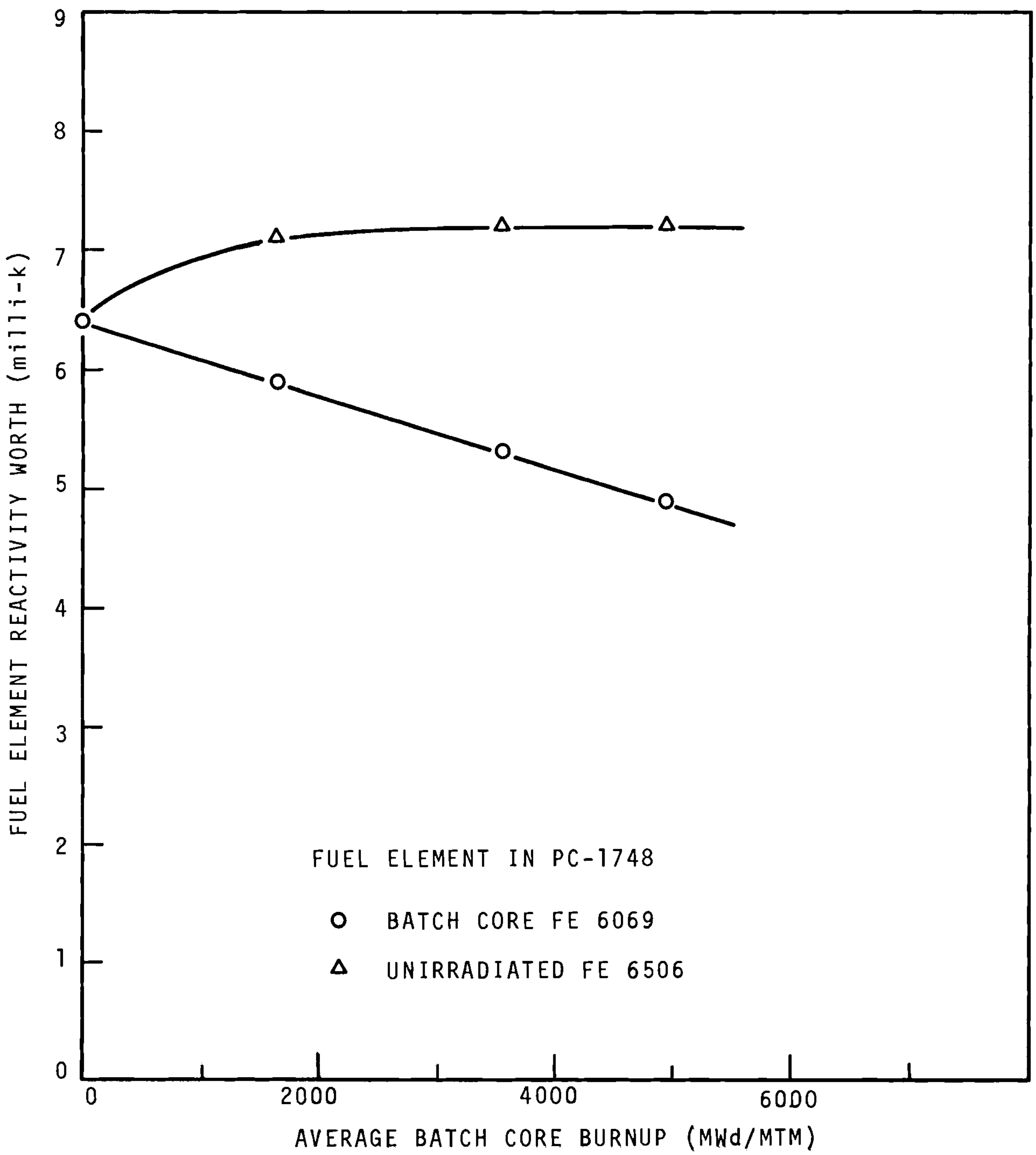


FIGURE 20. REACTIVITY WORTH OF FUEL ELEMENT IN RING THREE

The reactivity worth of FE 6069 decreased continuously due to its burnup. The reactivity worth of the unirradiated element FE 6506 relative to the core increased slightly.

At the completion of interim critical test series 2, a ring three fuel element was removed from the Batch Core for burnup analysis. To replace the removed element, a ring five element was moved into ring three, while a ring eight element was moved into ring five. The reactivity change produced by these fuel element moves was measured as a part of test series 2. The excess reactivity of the 55 element Batch Core was found to increase by 0.5 ± 0.1 milli-k as a result of these moves.

In interim critical test series 3, the reactivity worths of two additional unirradiated fuel elements were measured. One element (FE 6057) was a 19-rod cluster of 2% enriched UO_2 rods originally fabricated for the Carolinas Virginia Tube Reactor (CVTR). The reactivity worth of this element, relative to the D_2O-filled channel located in PC-1748, was measured to be 4.7 ± 0.1 milli-k. This is 35% lower than that of the unirradiated standard Batch Core element, FE-6506.

The other element (FE 6503) was a 19-rod cluster of vipac UO_2-4 wt% PuO_2, identical to a Batch Core element except for its enrichment. The reactivity worth of this element, relative to the D_2O filled channel located in PC-1748, was measured to be 9.2 ± 0.1 milli-k. This is 28% higher than that of the unirradiated standard Batch Core element, FE-6506.

I. Core Excess Reactivity

The method of determining the core excess reactivity for the various critical configurations was to measure the boron concentration and convert it to reactivity using the boron reactivity worth curve of Figure 7. Boron concentrations were adjusted for each configuration to achieve critical moderator heights of about 90, 95, and 100 inches. Critical moderator heights and boron concentrations for the 55 element Batch Core at various stages of burnup are listed in Table IX. Excess reactivity values are normalized to a critical moderator height of 100 inches. The contribution of the moderator level to the inferred excess reactivity was obtained from the moderator level reactivity worth curves of Figure 5. Uncertainties associated with the inferred excess reactivity were propagated from uncertainties in the measured boron concentrations, the moderator level worth curves, and the boron reactivity worth curve.

The reactivity loss for the core average burnup of 5330 ± 50 MWd/MTM was 90 ± 19 milli-k. This corresponds to an average reactivity burnup coefficient of 1.7 ± 0.4 milli-k per 100 MWd/MTM burnup for the 55 element Batch Core.

The ^{10}B concentration versus burnup for the cold, xenon free, 55 element Batch Core is shown in Figure 21. Each value is given for a critical moderator height of 100 inches. The last squares fitting code LEARN was used to linearly extrapolate the cold, xenon free results for each burnup step to a just critical system with zero boron at a 55 element core average burnup of 11,542 ± 840 MWd/MTM.

Table IX

Summary of Interim Critical Test Excess Reactivity Measurements

Interim Critical Test #	# FE	Core Average (a) Burnup, MWd/MTM	Critical Moderator Height, Inches	Boron Concentration, wppm ^{10}B (b)		Inferred Excess Reactivity at 100 inches, milli-k
Initial	55	0	85.30	18.89		255 (17) ± 26
			91.10	19.89		254 (8) ± 29
			93.05	20.18		254 (5) ± 29
			95.70	20.43		253 (2) ± 30
			98.30	20.98		257 (1) ± 32
					Average =	255 ± 13 milli-k
1st	55	1650	90.67	16.32		226 (9) ± 19
			95.75	16.80		224 (3) ± 20
			100.52	17.30		225 ± 21
					Average =	225 ± 12 milli-k
2nd	55	3560	89.80	12.07		189* (11) ± 10
			94.55	13.47		195* (4) ± 13
			100.35	14.06		197 ± 14
					Average =	194* ± 7 milli-k

Table IX (Con't)

Interim Critical Test #	# FE	Core Average (a) Burnup, MWd/MTM	Critical Moderator Height, Inches	Boron Concentration, wppm ^{10}B (b)		Inferred Excess Reactivity at 100 inches, milli-k
3rd	55	4930	89.95	11.29		181 (11) ± 9
			94.94	12.06		182 (4) ± 10
			99.71	12.33		180 ±
					Average =	181 ± 6 milli-k
4th	55	5330	90.33	9.82		165 (1) ± 7
			95.53	10.58		167 (4) ± 8
			99.57	10.94		168 (1) ± 9
					Average =	167 ± 5 milli-k
4th	31	6139	89.94	5.12		108 (13) ± 3
			94.81	5.78		109 (5) ± 3
			99.97	6.12		110 ± 3
					Average =	109 ± 2 milli-k
4th	19	6512	89.98	1.59		50 (15) ± 1.6
			94.93	2.05		50 (6) ± 0.8
			99.43	2.26		49 (1) ± 1.0
					Average =	50 ± 0.5 milli-k

( ) numbers in parentheses are moderator height reactivity contributions required to obtain the 100 inch value.

* these values are questionable due to the moderator sampling procedure

(a) the uncertainty of the core average burnup values is estimated to be ±1%

(b) the experimental uncertainty of the ^{10}B concentration is ± 0.05 wppm ^{10}B

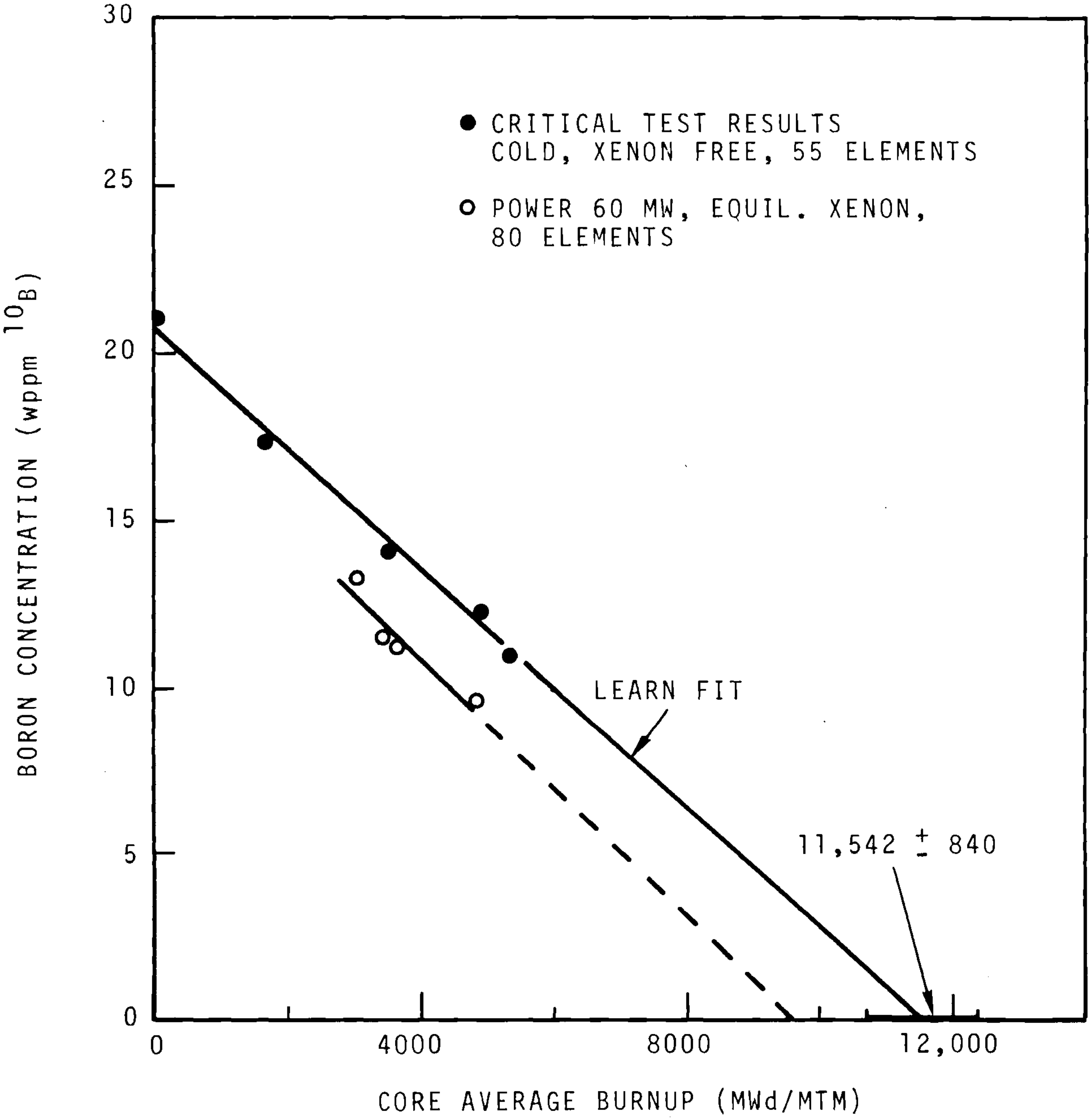


FIGURE 21. MODERATOR ^{10}B CONCENTRATION VERSUS CORE AVERAGE BURNUP FOR 100-INCH MODERATOR LEVEL

Moderator samples were taken periodically during the Batch Core Experiment under equilibrium operating conditions at full power with the full core loading of about 80 fuel elements. The ^{10}B concentrations of some of these samples, adjusted to be equivalent to 60 MW operation at a critical moderator height of 100 inches, are also shown in Figure 21. The extrapolated lifetime for 60 MW equilibrium operation is seen to be approximately 9500 MWd/MTM.

J. Reactivity Worth of Poison Rods in the Moderator

The reactivity worths of several rods containing control rod material were measured in series 2 and 3 of the interim critical tests. Measurements were made of three rods; boron-aluminum, hafnium-aluminum, and aluminum. Each rod was 0.368 inch diameter and 36.875 inches long, clad with zircaloy. The compositions of the rods are listed in Table X. All three were measured both bare and covered with 80-mil cadmium sleeves. Reactivity measurements were made by inserting each rod in turn into the thimble in C-1553 such that it was centered axially in the core and performing an approach to critical to determine critical moderator height. The results of these measurements are listed in Table XI. The reactivity worths of the bare poison rods are listed relative to that of the bare aluminum rod, while the worths of the cadmium-covered poison rods are listed relative to that of the cadmium-covered aluminum rod.

Table X

Composition of Poison Rods

Hafnium rod:

Poison density ($Hf\ O_2$) = 5.11 g/cm^2

Weights:

$Hf\ O_2$	320.4 g
$Zr\ O_2$	8.5
Al	29.4

Form:

Pellets, ∿ 1/4" to 3/4" in length.

Typical agglomerates are of the order of 10-20 microns

Boron rod:

Poison density (B) = 0.0853 g/cm^3

Weights:

B	5.350 g
Al	153.01
O	0.44
Mg	0.10

Form:

Several swaged sections, each ∿ 6" long.

B particle size 0.3-1.5 microns; typical agglomerates are of the order of a few microns.

Table XI

Reactivity Worth of Poison Rods

Poison Rod	Reactivity Worth (milli-k)
Interim critical test series 2:	
B, bare	2.4 ± 0.1
B, Cd-covered	0.2 ± 0.1
Hf 0_2, bare	2.3 ± 0.1
Hf 0_2, Cd-covered	0.6 ± 0.1
Interim critical test series 3:	
B, bare	2.3 ± 0.1
B, Cd-covered	0.1 ± 0.1
Hf 0_2, bare	2.2 ± 0.1
Hf 0_2, Cd-covered	0.6 ± 0.1

In both cases, the reactivity worth of the cadmium sleeve on the aluminum rod was 3.4 ± 0.1 milli-k. The reactivity worths of the poison rods were essentially the same for both interim critical test series at which they were measured. A comparison of the bare and cadmium covered rod worths shows the difference between the resonance absorber (hafnium) and the 1/v absorber (boron).

K. Kinetics Parameters

The value of β/ℓ, the ratio of the effective delayed neutron fraction to the prompt neutron lifetime, was measured several times during the Batch Core Experiment using a sinusoidal reactivity oscillator. The effective multiplication factor was varied sinusoidally about a value of unity by an oscillator assembly mounted in shim hole C-1447 (See Figure 1). A cadmium stator, with a cadmium section rotating relative to it, was located at approximately the vertical mid-plane of the core. The rotating cadmium section was shaped so that its rotation relative to the cadmium stator produced a sinusoidal variation in reactivity of 0.05 milli-k peak-to-peak. The frequency of the reactivity variation was adjustable from 0.1 to 15 cycles per second.

The shapes of the transfer functions were found to be independent of the reactor power for all the power levels used in the measurements. A typical transfer function amplitude response is shown in Figure 22. The points represent the experimental data, while the curve is a fit to the formula

$$|W|^2 = \frac{A_1^2}{\omega^2} \frac{\omega^2 + A_2}{\omega^2 + (\beta/\ell)^2}$$

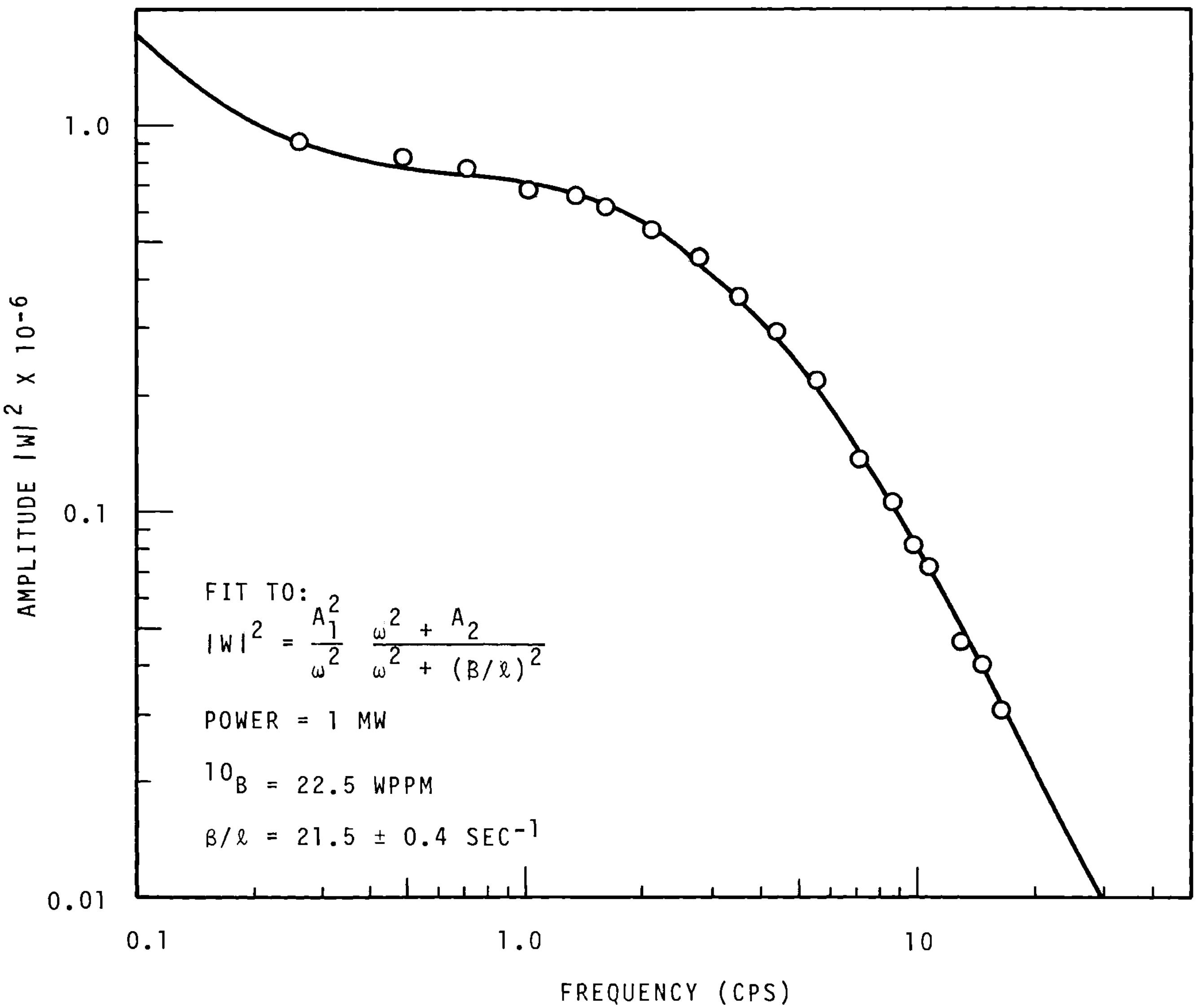


FIGURE 22. PRTR TRANSFER FUNCTION AMPLITUDE RESPONSE

where

A_1, A_2 = constants

ω = frequency in radians/second

β = delayed neutron fraction

ℓ prompt neutron lifetime in seconds

using the least squares fitting code LEARN.

Results of all the β/ℓ measurements made during the Batch Core Experiment are listed in Table XII. These include two measurements made during the power tests[7] with a 77 element core.

Table XII

Summary of β/ℓ Measurements in the Batch Core

Core Average Burnup (MWd/MTM)	Core Size (No. of Elements)	Boron Conc. (wppm ^{10}B)	β/ℓ (sec^{-1})
0	55	20.4 ± 0.1	17.9 ± 0.5
10 ± 1	77 (250 watts)	21.8 ± 0.1	20 ± 1
10 ± 1	77 (1 MW)	22.5 ± 0.1	21.5 ± 0.4
3560 ± 34	55	13.8 ± 0.1	15.1 ± 0.6
5330 ± 50	55	10.6 ± 0.1	14.3 ± 0.3
6139 ± 58	31	5.9 ± 0.1	10.4 ± 0.5
6512 ± 61	19	2.3 ± 0.1	8.0 ± 0.2

For the 31 and 19 element cores, the oscillator was positioned in shim hole C-1245 rather than C-1447 in order to reduce its peak-to-peak worth. The ion chamber was located in C-1352 for all oscillator measurements.

The variation of β/ℓ with core average burnup and ^{10}B concentration is illustrated in Figures 23 and 24, respectively.

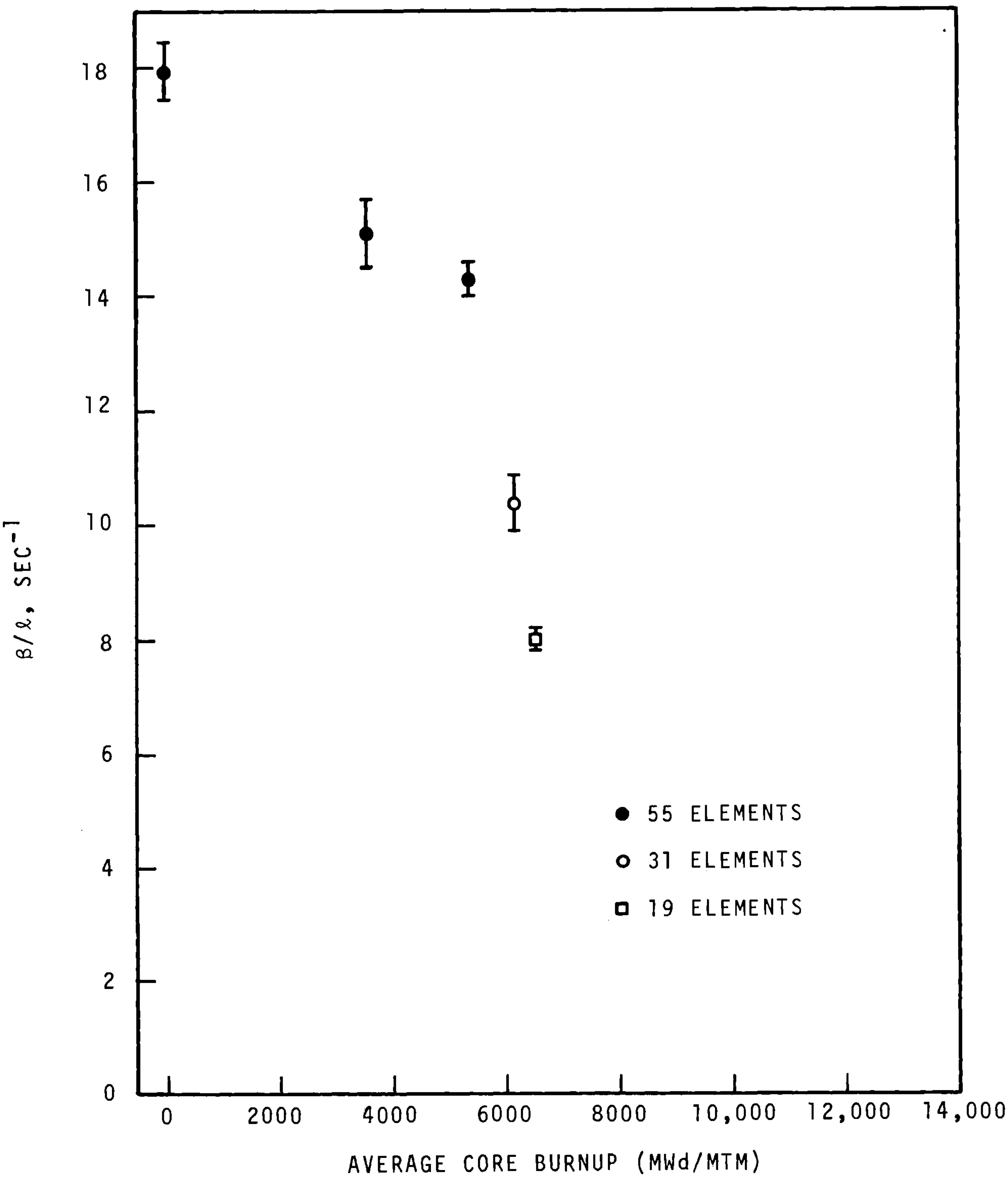


FIGURE 23. DELAYED NEUTRON FRACTION/PROMPT NEUTRON LIFETIME (β/ℓ) VERSUS AVERAGE CORE BURNUP

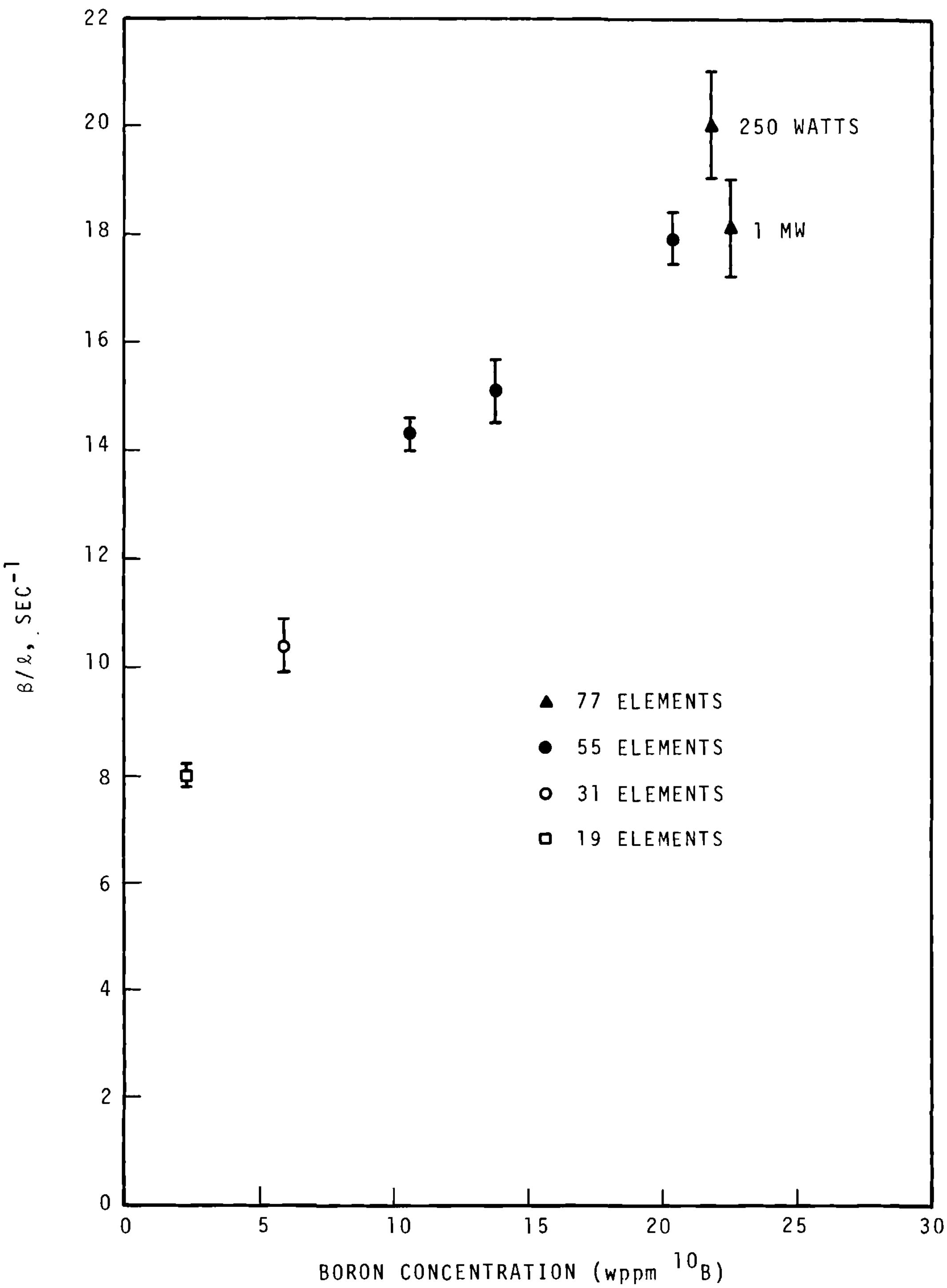


FIGURE 24. DELAYED NEUTRON FRACTION/PROMPT NEUTRON LIFETIME (β/ℓ) VERSUS ^{10}B CONCENTRATION IN MODERATOR

VII. CONCLUSIONS

The results of the Batch Core Critical Tests have, in general, confirmed the predicted characteristics[1] of the PRTR Batch Core. The boron concentration in the D_2O moderator needed to compensate for the initial excess reactivity of the core agreed with the predicted concentration requirements to within a few percent. Measurements of void reactivity worths and kinetics parameters confirmed the data used in safety analyses.

The decrease in excess reactivity with core burnup was in close agreement with calculations. Due to the premature termination of the experiment, the total core reactivity lifetime had to be determined by extrapolation of the available data.

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REFERENCES

1. J. R. Worden, W. L. Purcell, L. C. Schmid, Physics Experiment - High Power Density Core of the PRTR, BNWL-221, Pacific Northwest Laboratory (1966).

2. N. G. Wittenbrock, P. C. Walkup, J. K. Anderson, PRTR Final Safeguards Analysis, HW-61236. Hanford Laboratories (1959).

3. J. P. Keenan, R. W. Sharp, PRTR Fuel Element Quality Control System, BNWL-349, Pacific Northwest Laboratory (1961).

4. J. R. Triplett, et.al., Plutonium Recycle Test Reactor Critical Test Results, HW-61900 BA, Hanford Laboratories (1961).

5. B. H. Duane, Maximum Likelihood Nonlinear Correlated Fields (BNW Program LIKELY), BNWL-390, Pacific Northwest Laboratory (1967).

6. G. D. Seybold, Users Aid: Programs LEARN and LIKELY, BNWL-1057, Pacific Northwest Laboratory (1969).

7. J. W. Kutcher and R. I. Smith, Power Tests with the UO_2-2 wt% PuO_2 Batch Core in the PRTR, BNWL-1383, Pacific Northwest Laboratory (1970).

8. J. H. Lauby and J. W. Kutcher, "Measurements of ^{10}B Concentrations Using Reactivity Techniques," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1966, BNWL-400, Pacific Northwest Laboratory (1967), p. 16.

APPENDIX I

Summary

Of

Experimental Data

And

Loading Sequence Information

Core Load & Mult. No.	Core Description No. F.E.	Core Description Conc. B-10	Critical Height	Period No. Pa-	Rising Period (sec)	ΔH (in.)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	Level Coefficient (Ave HT)	Remarks
1	4	0	Subcritical							Charged Channels 1550, 1451, 1449, 1548
2	5	0	Subcritical							Charged Channel 1649
3	6	0	Subcritical							Charged Channel 1552
4	7	0	Subcritical							Charged Channel 1651
5	8	0	Subcritical							Charged Channel 1350
6	9	0	Subcritical							Charged Channel 1447
7	9	0	97.75	7-1	74.80	0.48	0.4328	0.9016	97.99	Removed Large BF_3 and Fission Chamber from C-1352
				7-2	42.78	0.70	0.6267	0.8954	98.10	1550 Filled (D_2O)
				7-3	27.73	0.90	0.8089	0.8988	98.20	
8	9	0	99.0	8-1	190.35		0.2128			Axial Traverse (0.006" Au) installed BF_3 in 1352
9	9	0	100.15	9-1	52.699	1.00	0.5488	0.5488	100.65	1550 Drained to -15'6"
				9-2						Clad Temp (110-160)°F
				9-3	164.33	0.80	0.2395	0.2994		Data Questionable Due to Heating of Fuel in Voided (136-225)°F
				9-4						Channel 1550
										Clad Temp. Reached 200°F
				9-5	35.01	1.46	0.7076	0.4847	100.88	Clad Temp. (163-185)°F
				9-6	61.612	0.82	0.4947	0.6033	100.56	Clad Temp. (83-85)°F
				9-7	27.602	1.45	0.8110	0.5593	100.875	Clad Temp. (87-128)°F
				9-8	163.28	0.55	0.2407	0.4376	100.425	Clad Temp. (125-160)°F
				9-9	95.528	0.70	0.3630	0.5186	100.50	Clad Temp. (145-218)°F
				9-10	66.876	1.00	0.4679	0.4679	100.65	Clad Temp. (188-237)°F
				9-11	44.563	1.35	0.6110	0.4526	100.825	Clad Temp. (220-262)°F
10	9	0	100.25	10-1	68.718	0.802	0.4592	0.5726	100.651	1550 Dry. Clad Temp. (85-86)°F
				10-2	111.84	0.600	0.3228	0.5380	100.55	Clad Temp. (82-90)°F
				10-3	30.770	1.393	0.7628	0.5476	100.95	Clad Temp. (85-87)°F
11	9	0	94.32	11-1	46.916	0.301	0.5914	1.9647	94.47	1550 Filled with H_2O
				11-2	31.554	0.410	0.7519	1.8339	94.525	
				11-4	122.55	0.168	0.3011	1.0454	94.464	
12	9	0	100.25	12-1	30.575	1.25	0.7656	0.6125	100.875	1550 Drained to -15'6" (H_2O)
				12-2	60.525	0.75	0.5007	0.6676	100.625	
				12-3	136.99	0.50	0.2763	0.5526	100.50	
13	9	0	97.99	13-1	50.904	0.6575	0.5613	0.8536	98.318	1550 Filled (96.99% D_2O), (78-78)°F
				13-2	114.53	0.371	0.3170	0.8544	98.175	(80-81)°F
				13-3	34.77	0.829	0.7104	0.8569	98.404	(80-82)°F
14	9	0	100.10	14-1	72.378	0.80	0.4351	0.5439	100.50	1550 Drained to -15'6" (D_2O), (83-85)°F
				14-2	25.27	1.53	0.8400	0.5490	100.86	Clad Temp. (79-82)°F
				14-3	213.47	0.52	0.1889	0.3633	100.36	Clad Temp. (79-82)°F
				14-4	71.05	0.70	0.4409	0.6298	100.45	
				14-5	134.32	0.60	0.2745	0.4575	100.40	Clad Temp. (88-95)°F
15	9	0	97.42	15-1	35.638	0.677	0.7003	1.0344	97.759	1550 Filled (D_2O),
				15-2	51.541	0.530	0.5568	1.0543	97.685	Removed BF_3 Tubes from C-1352
				15-3						and M-1450.
										Scaler Malfunction
				15-4	92.44	0.353	0.3718	1.0532	97.596	

Core Load & Mult. No.	Core No. F.E.	Description Conc. B-10	Critical Height	Period No. Pa-	Rising Period (sec)	ΔH (in.)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	Level Coefficient Ave HT)	Remarks
16	10	0	87.00	16-1	65.652	0.270	0.4738	1.7544	87.135	Charged Channel 1453
				16-2						Scram
				16-3	100.100	0.200	0.3507	1.7535	87.100	
				16-4						Scram
				16-5						Scram
				16-6						Scaler Malfunction
				16-7						Scaler Malfunction
				16-8						Scram
				16-9	61.399	0.43	0.4959	1.1532	87.22	
				16-10						Scram
				16-11	87.00					Scram
17	11	0	81.187	17-1	44.021	0.176	0.6175	3.4589	81.224	Charged Channel 1653
				17-2	71.162	0.133	0.4482	3.3699	81.253	
				17-3	105.270	0.095	0.3378	3.5557	81.272	
				17-4	48.009	0.173	0.5828	3.3687	81.273	
18	12	0	77.00	18-1						Charged Channel 1647, Scram
				18-2						Scram
				18-3	93.292	0.100	0.3693	3.6930	77.050	
				18-4						Scram
				18-5	166.63	0.058	0.2368	4.0827	77.029	
				18-6	83.304	0.100	0.4010	4.0100	77.050	
				18-7						Scram
				18-8	85.732	0.100	0.3927	3.9270	77.050	
19	13	0	73.5	19-1						Charged Channel 1750, Scram
				19-2						Scrammed on all Period Runs
				19-3	142.9	0.028	0.2673	9.546	73.515	From Spurious Noise
				19-4						
20	12	1.231	79.1							Discharged Channel 1750 Added Boron, Sample C-1
21	12	1.1	85.245	21-1	86.029	0.153	0.3918	2.5607	85.322	Added Boron, No Sample
				21-2						Scaler Malfunction
				21-3	102.580	0.135	0.3444	2.551	85.313	
				21-4	64.686	0.185	0.4786	2.5731	85.338	
22	12	2.160	94.500	22-1	62.204	0.355	0.4914	1.384	94.678	Added Boron, Sample C-2
				22-2	115.86	0.226	0.3142	1.3902	94.613	
				22-3	81.582	0.305	0.4069	1.334	94.653	
23	12	2.638	102.390	23-1	274.72	0.605	0.1574	0.2601	102.693	Added Boron, Sample C-3
				23-2	168.2	0.918	0.2350	0.2559	102.849	
				23-3	83.541	1.61	0.4001	0.2485	103.196	
24	12	2.638	Subcritical							1550 Drained to -15'6" (D_2O)
25	12	2.554	102.600	25-1	71.954	1.60	0.4447	0.2779	103.40	Added 800 lbs. new D_2O to Mod. C-4
				25-2	81.432	1.40	0.4075	0.2910	103.30	1550 as above
				25-3	57.466	1.80	0.5183	0.2879	103.50	
				25-4	87.639	1.10	0.3865	0.3514	103.15	
26	12	2.554	102.49	26-1	172.76	0.961	0.2301	0.2394	102.97	1550 Dry
				26-2	94.358	1.50	0.3662	0.2441	103.29	
				29-3	72.177	1.86	0.4437	0.2385	103.42	

Core Load & Mult. No.	Core Description No. F.E.	Conc. B-10	Critical Height	Period No. Pa-	Rising Period (sec)	ΔH (in.)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	Ave HT)	Remarks
27	12	2.554	95.04	27-1	143.924	0.155	0.2658	1.7148	95.118	1550 Filled with H_2O
				27-2	56.380	0.360	0.5249	1.458	95.220	
				27-3	48.291	0.410	0.5805	1.4158	95.245	
				27-4	90.818	0.26	0.3766	1.4484	95.170	
				27-5	137.84	0.198	0.2749	1.3893	95.139	
28	12	2.554	102.20	28-1	83.834	1.80	0.3991	0.7217	103.10	1550 Drained to -15'6" (H_2O)
				28-2	110.42	1.50	0.3258	0.2172	102.95	
				28-3	68.558	2.093	0.4598	0.2196	103.25	
				28-4	57.97	2.4	0.5153	0.2147	103.40	
29	12	2.554	99.40	29-1	80.785	0.80	0.4098	0.5122	99.80	1550 Filled with 98.10% D_2O
				29-2	62.34	1.01	0.4907	0.4858	99.905	
				29-3	126.01	0.60	0.2947	0.4956	99.70	
				29-4	66.653	1.00	0.4688	0.4688	99.90	
				29-5	91.58	0.80	0.3743	0.4678	99.80	
				29-6	132.94	0.60	0.2827	0.4711	99.70	
30	12	2.554		30-1	45.3					Axial Traverse (0.006" Au)
31	13	2.554	88.96	31-1	67.129	0.190	0.4665	2.4553	89.055	Charged Channel 1750
				31-2	102.81	0.140	0.3437	2.4550	89.03	
				31-3	116.14	0.134	0.3136	2.3417	89.027	
32	14	2.554	83.42	32-1	107.46	0.130	0.3325	2.5577	83.485	Charged Channel 1752
				32-2	32.451	0.290	0.7396	2.5503	83.565	
				32-3	62.998	0.190	0.4872	2.5642	83.515	
33	15	2.554	79.543	33-1	74.511	0.125	0.4339	3.4712	79.606	Charged Channel 1554
				33-2	209.15	0.064	0.1971	3.0796	79.575	
				33-3	43.585	0.208	0.6194	2.9778	79.647	
				33-4	121.60	0.103	0.3028	2.9398	79.595	
34	16	2.554	76.788	34-1	50.87	0.130	0.3614	4.3184	76.853	Charged Channel 1352
				34-2	97.602	0.082	0.3572	4.3560	76.829	
				34-3	79.856	0.100	0.4132	4.1320	76.838	
35	16	4.510	86.246	35-1	134.87	0.132	0.2796	2.118	86.312	Added Boron, Sample C-5
				35-2	73.485	0.207	0.4382	2.1169	86.350	
				35-3	97.322	0.164	0.3580	2.182	86.328	
36	16	5.430	93.230	36-1	55.704	0.320	0.5290	1.653	93.39	Added Boron, Sample C-6
				36-2	75.279	0.260	0.4307	1.656	93.36	
				36-3	102.58	0.205	0.3442	1.679	93.33	
37	16	5.567	94.850	37-1	46.043	0.500	0.5982	1.1964	95.100	Added Boron, Sample C-7
				37-2	76.757	0.358	0.4248	1.1865	95.02	
				37-3	120.5	0.259	0.3049	1.1772	94.97	
38	16	5.567	95.620	38-1	45.299	0.580	0.6044	1.0421	95.910	1550 Drained to -15'6" (D_2O 97.9%)
				38-2	85.208	0.380	0.3943	1.0376	95.810	
				38-3	135.88	0.275	0.2780	1.0109	95.758	
39	16	5.567	95.67	39-1	54.753	0.498	0.5350	1.074	95.919	1550 Dry
				39-2	114.44	0.298	0.3171	1.0640	95.819	
				39-3	74.57	0.389	0.4336	1.1146	95.865	
40	16	5.567	92.515	40-1	35.501	0.456	0.7015	1.5383	92.743	1550 Filled with H_2O
				40-2	54.405	0.355	0.5372	1.5132	92.693	
				40-3	89.707	0.245	0.3799	1.5506	92.638	

Core Load & Mult. No.	Core Description No. F.E.	Conc. B-10	Critical Height	Period No. Pa-	Rising Period (sec)	Δ H (in.)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	Ave HT)	Remarks
41	16	5.567	95.54	41-1	56.532	0.460	0.5238	1.1387	95.77	1550 Drained to -15'6" (H_2O)
				41-2	82.278	0.360	0.4044	1.1233	95.72	
				41-3	124.22	0.260	0.2979	1.1457	95.67	
42	16	5.567	95.52	42-3	127.31	0.28	0.2923	1.0439	95.66	1550 Dry
				42-2	44.857	0.58	0.6082	1.0486	95.81	
				42-1	70.851	0.43	0.4494	1.0451	95.735	
43	16	5.567		43-1	41.05		0.6426			Axial Traverse (0.006" Au), 1550 Filled (D_2O
44	17	5.567	89.385	44-1	58.787	0.263	0.5104	1.9406	89.517	Charged Channel 1348
				44-2	88.47	0.195	0.3838	1.8682	89.483	
				44-3	45.391	0.306	0.6036	1.9725	89.538	
45	18	5.567	84.740	45-1	73.008	0.195	0.4401	2.2569	84.838	Charged Channel 1546
				45-2	37.215	0.289	0.6820	2.3598	84.885	
				45-3	195.66	0.086	0.2081	2.4197	84.783	
46	21	5.567	77.356	46-1	74.882	0.105	0.4323	4.1171	77.409	Charged Channels 1748, 1249, 1251
				46-2	110.00	0.078	0.3267	4.1884	77.395	
				46-3	58.575	0.122	0.5166	4.2344	77.417	
47	22	5.567	75.484	47-3	84.476	0.086	0.3968	4.6140	75.527	Charged Channel 1354
				47-2	85.58	0.081	0.3931	4.8531	75.525	
				47-1	52.752	0.116	0.5482	4.7259	75.542	
48	23	5.567	73.916	48-1	73.897	0.132	0.4363	3.3053	73.982	Charged Channel 1455
				48-2	151.08	0.074	0.2558	3.4568	73.953	
				48-3	46.563	0.172	0.5940	3.4535	74.002	
49	23	8.554	85.410	49-1	47.901	0.260	0.5834	2.2438	85.540	Added Boron, Sample C-8
				49-2	65.031	0.215	0.4767	2.2172	85.518	
				49-3	82.462	0.180	0.4037	2.2427	85.500	
50	23	10.560	Subcritical							Added Boron, Sample C-9
51	23	10.162	95.390	51-1	47.777	0.643	0.582	0.9085	95.71	Removed Boron, Sample C-10
				51-2	87.801	0.430	0.3858	0.8972	95.60	
				51-3	108.18	0.363	0.3307	0.9110	95.57	
52	23	10.162	95.850	52-1	65.596	0.520	0.4737	0.9109	96.110	1550 Drained to -15'6" (D_2O)
				52-2	41.362	0.700	0.6394	0.9134	96.200	
				52-3	137.39	0.300	0.2755	0.9183	96.000	
53	23	10.162	95.860	53-2	47.456	0.640	0.5867	0.9167	96.180	1550 Dry
				53-1	91.056	0.410	0.3757	0.9163	96.065	
				53-3	67.021	0.500	0.4668	0.9336	96.110	
54	23	10.162	93.410	54-1	61.077	0.398	0.4973	1.2495	93.609	1550 Filled with H_2O
				54-2	47.876	0.474	0.5834	1.2308	93.647	
				54-3	82.789	0.340	0.4025	1.1838	93.580	
55	23	10.162	95.800	55-1	70.377	0.501	0.4513	0.9008	96.050	1550 Drained to -15'6" (H_2O)
				55-2	42.099	0.693	0.6324	0.9126	96.147	
				55-3	113.12	0.343	0.3198	0.9324	95.972	
56	23	10.162	95.15	56-1	57.153		0.5199			Axial Traverse (0.006" Au), 1550 Filled (D_2O)
57	25	10.162	88.303	57-1	32.714	0.377	0.7356	1.9511	88.491	Charged Channels 1346, 1445
				57-2	56.527	0.267	0.5237	1.9614	88.436	
				57-3	148.63	0.152	0.2591	1.7046	88.379	

Core Load & Mult. No.	Core Description No. F.E.	Conc. B-10	Critical Height	Period No. Pa-	Rising Period (sec)	ΔH (in)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	Ave HT)	Remarks
58	27	10.162	83.375	58-3	89.703	0.158	0.3798	2.4037	83.454	Charged Channels 1645, 1746
				58-1	69.656	0.189	0.4546	2.4052	83.470	
				58-2	43.081	0.260	0.6235	2.3980	83.505	
59	29	10.162	80.270	59-1	45.927	0.185	0.5990	3.2378	80.363	Charged Channels 1849, 1851
				59-2	60.495	0.150	0.5005	3.3366	80.345	
				59-3	82.504	0.125	0.4035	3.2280	80.333	
60	31	10.162	77.331	60-1	81.238	0.110	0.4079	3.7081	77.386	Charged Channels 1754, 1655
				60-2	51.863	0.154	0.5541	3.5980	77.408	
				60-3	109.13	0.080	0.3286	4.1275	77.371	
61	31	13.830	93.400	61-1	46.141	0.450	0.5972	1.3271	93.625	Added Boron, Sample C-11
				61-2	69.88	0.343	0.4536	1.3224	93.572	
				61-3	87.523	0.293	0.3867	1.3198	93.547	
62	31	13.830	Subcritical							Primary System and 1550 Dry
63	31	12.315	100.25	63-3	57.901	2.20	0.5454	0.2343	101.35	As Above, Removed Boron, Sample C-12
				63-2	79.189	1.75	0.4154	0.2374	101.125	
				63-1	124.53	1.27	0.2972	0.2341	100.875	
64	31	12.315	85.89	64-1	62.529	0.224	0.4894	2.1848	86.002	Primary System and 1550 Filled, D_2O
				64-2	39.363	0.312	0.6591	2.4125	86.046	
				64-3	98.541	0.170	0.3545	2.0853	85.975	
65	31	14.328	100.83	65-1	198.18	1.549	0.2059	0.1328	101.605	Added Boron, Sample C-13
				65-2	98.741	2.69	0.3539	0.1316	102.175	
66	31	14.135	96.83	66-1	122.82	0.47	0.3003	0.6389	99.06	Removed Boron, Sample C-14
				66-2	86.102	0.61	0.3912	016413	99.13	
				66-3	52.037	0.87	0.5528	0.6331	99.26	
67	31	14.135	96.925	67-3	61.793	0.935	0.4933	0.5276	97.393	1550 Dry
				67-1	80.931	0.775	0.4090	0.5277	97.313	
				67-2	43.431	1.175	0.6202	0.5278	97.513	
68	31	14.135	94.87	68-1	49.271	0.52	0.5726	1.1012	95.130	1550 Filled with H_2O
				68-4	97.502	0.318	0.3572	1.123	95.029	
				68-5	67.427	0.43	0.4648	1.084	95.085	
69	31	14.135	96.7	69-1	45.519					Axial Traverse (0.006" Au), 1550 Filled (D_2O)
70	31	14.135	96.7	70-3	103.560	0.50	0.3416	0.6832	96.950	Plastic Liner in C-1352, Dry
				70-2	64.469	0.69	0.4793	0.6946	97.045	
				70-1	45.293	0.90	0.6041	0.6712	97.150	
71	31	14.135	95.82	71-3	42.362	0.82	0.6298	0.7680	96.230	Plastic Liner in C-1352,
				71-1	98.163	0.47	0.3555	0.7564	96.055	Filled with Borated Moderator
				71-2	60.638	0.65	0.4996	0.7686	96.145	
72	34	14.135	88.83	72-1	46.804	0.288	0.5917	2.0545	88.974	C-1352 Dry, Charged Channels 1556, 1253, 1247
				72-2	63.252	0.228	0.4855	2.1294	88.944	
				72-3	94.267	0.176	0.3664	2.0818	88.918	
73	37	14.135	83.63	73-1	84.251	0.180	0.3973	2.2072	83.720	Charged Channels, 1853, 1847, 1544
				73-2	64.743	0.208	0.4779	2.2975	83.734	
				73-3	116.38	0.138	0.3129	2.2673	83.699	
74	40	14.135	80.806	74-1	58.87	0.194	0.5096	2.6268	80.903	Charged Channels 1150, 1356, 1756
				74-2	29.314	0.284	0.7830	2.7570	80.948	
				74-3	100.96	0.124	0.3481	2.8073	80.868	
				74-4	91.693	0.144	0.3737	2.5951	80.878	

Core Load & Mult. No.	Core Description No. F.E.	Conc. B-10	Critical Height	Period No. Pa-	Rising Period (sec)	ΔH (in)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	(Ave HT)	Remarks
75	43	14.135	78.615	75-2	43.584	0.223	0.6188	2.7749	78.727	Charged Channels 1950, 1744, 1344
				75-1	58.923	0.190	0.5093	2.6805	78.710	
				75-3	88.329	0.138	0.3840	2.7826	78.684	
76	46	14.135	76.81	76-1	34.619	0.200	0.7115	3.56	76.910	Charged Channels 1245, 1152, 1457
				76-2	55.01	0.150	0.5330	3.55	76.885	
				76-3	86.584	0.115	0.3896	3.54	76.887	
77	49	14.135	75.328	77-1	128.64	0.075	0.2898	3.8640	75.366	Charged Channels 1643, 1948, 1855
				77-2	71.926	0.107	0.4444	4.1532	75.382	
				77-3	57.235	0.138	0.5192	3.7623	75.397	
78	52	14.135	73.600	78-3	222.04	0.050	0.1876	3.7520	73.625	Charged Channels 1255, 1148, 1443
				78-2	92.584	0.100	0.3711	3.7110	73.650	
				78-1	52.290	0.150	0.5510	3.6733	76.677	
79	55	14.135	72.217	79-3	170.07	0.053	0.2328	4.3925	72.244	Charged Channels 1657, 1952, 1845
				79-2	95.009	0.083	0.3641	4.3867	72.259	
				79-1	47.505	0.133	0.5861	4.4067	72.284	
80	55	14.135	72.20	80-1						Axial Traverse (0.006" Au)
				80-2	56.654		0.5228			
81	55	16.217	77.6	81-1	55.371	0.125	0.5307	4.2456	77.663	Au, Lu Normalization. Added Boron, Sample C-15
82	55	18.889	85.303	82-1	70.085	0.230	0.4524	1.9670	85.420	Added Boron, Sample C-16 1550 Filled with D_2O
				82-2	35.903	0.342	0.6961	2.0354	85.476	
				82-3	133.35	0.140	0.2818	2.0129	85.375	
83	55	18.889	89.765	83-3	204.34	0.085	0.2008	2.3624	89.808	Primary System and 1550 Dry
				83-2	106.27	0.135	0.3350	2.4815	89.833	
				83-1	51.644	0.235	0.5553	2.3630	89.883	
84	55	18.889	85.27	84-1	67.183	0.240	0.4658	1.9408	85.390	Repeat of 82
				84-2	184.84	0.110	0.2177	1.o791	85.325	
				84-3	39.116	0.320	0.6612	2.0663	85.430	
85	55	20.176	93.05	85-3	82.127	0.32	0.4046	1.2644	93.230	Added Boron, Sample C-17
				85-2	39.899	0.52	0.6532	1.2562	93.310	Primary System and 1550 full D_2O
				85-1	58.497	0.42	0.5116	1.2181	93.26	
86	55	20.176	93.98	86-1	52.415	0.52	0.550	1.0577	94.240	H_2O Reflector Drained Primary and 1550 Full, D_2O
				86-2	67.798	0.43	0.4657	1.0830	94.195	
				86-3	92.671	0.35	0.3707	1.0591	94.155	
87	55	20.176	93.7							1550 Dry, Reflector Partially Filled with H_2O
88	55	20.815	98.45	88-3	68.859	4.60	0.4579	0.09954	100.75	Added Boron, Sample C-18
				88-2	82.588	4.00	0.4029	0.1007	100.45	1550 Dry, Partial Reflector
				88-1	112.84	3.20	0.3202	0.1001	100.05	
89	55	20.815	97.22	89-3	86.07	1.15	0.3911	0.3401	97.79	1550 Filled with H_2O, Partial Reflector
				89-2	71.88	1.28	0.445	0.3472	97.86	
				89-1	42.084	1.85	0.6321	0.3417	98.15	
90	55	20.815	100.275	90-1	99.629	2.995	0.3515	0.1174	101.773	1550 Filled with D_2O (86%)
				90-2	131.95	2.195	0.2841	0.1294	101.373	
				90-3	82.456	3.995	0.4034	0.1010	102.273	
91	55	20.815	97.55							1550 Dry, Reflector Filled (H_2O
92	55	21.002	Subcritical							Added Boron, Sample C-19

Core Load & Mult. No.	Core No. F.E.	Description Conc. B-10	Critical Height	Period No.	Rising Period (sec)	ΔH (in)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	Level Coefficient (Ave HT)	Remarks
93	55	20.976	98.4	93-1	72.438	1.7	0.4896	0.3060	99.20	Removed Boron, Sample C-20, 1550 Dry,
				93-2	49.15	1.85	0.5733	0.3099	99.375	Reflector Full
				93-3	43.377	2.1	0.6204	0.2954	99.45	
94	55	20.976	96.35	94-1	28.511	1.70	0.7949	0.4676	97.200	1550 Filled with H_2O
				94-2	70.955	0.95	0.4485	0.4721	96.825	
				94-3	113.82	0.70	0.3181	0.4544	96.700	
				94-4	36.658	1.45	0.6874	0.4741	97.075	
95	55	20.976	97.66	95-1	56.051	2.44	0.5263	0.2156	98.88	1550 Filled with D_2O
				95-2	46.181	2.84	0.5965	0.2100	99.08	
				95-3	80.659	1.94	0.4098	0.2112	98.63	
96	55	20.976	98.3	96-1	51.705	2.03	0.5548	0.2733	99.32	Lutetium Foil Traverse (Bare), T/C Removed
97	55	20.976	99.1	97-1	66.831	2.47	0.4674	0.1892		Lutetium Foil Traverse (CD Cov.) T/C Removed
98	55	20.976	101.6							Noise Measurement
99	55	20.976	97.55	99-3	92.598	1.35	0.3709	0.2747	98.22	C-1352 w/Liner, Dry
				99-2	50.557	2.05	0.5629	0.2746	98.57	
				99-1	68.925	1.15	0.4576	0.2773	98.37	
100	55	20.976	97.24	100-3	87.555	1.14	0.3863	0.3388	97.81	C-1352 w/Liner, Filled with Borated Moderator
				100-2	46.644	1.76	0.5927	0.3367	98.12	
				100-1	67.859	1.36	0.4625	0.3401	97.92	
101	55	20.976	97.6	101-1	72.654	1.6	0.4412	0.2757	98.4	C-1352 w/Liner, Dry
				101-2	59.679	1.848	0.5047	0.2731	98.524	
				101-3	52.147	2.045	0.5518	0.2698	98.622	
				101-4	92.663	1.40	0.3707	0.2648	98.30	
102	55	20.976	97.58	102-1	41.087	2.47	0.6415	0.2597	98.81	Replaced Std. F.E. in 1548 with 6519, Fission Gas Pressure Element
				102-2	45.208	2.27	0.6045	0.2662	98.71	
				102-3	55.58	1.98	0.5292	0.2672	98.57	
103	55	20.976	98.0	103-1	55.499	1.55	0.5297	0.3417	98.77	Replaced Std. F.E. in 1245 with F.E. 6520, Fission Gas Pressure Element
				103-2	44.383	1.89	0.6116	0.3235	98.895	
				103-3	93.279	0.97	0.3689	0.3803	98.485	
104	55	20.976	Subcritical							Removed Liner from C-1352, Installed Noise chamber in C-1352, Installed Oscillator in C-1447
105	55	13.739	99.5	105-1	95.232	1.925	0.3634	0.1888	100.46	Removed Log N Chamber from PT-1053 Removed Boron, Sample C-21
				105-2	45.514	4.5	0.6020	0.1338	101.75	
							(OSC Min. Worth)			
				105-3	70.157	4.5	0.4520	0.1004	101.75	
106	55	Not Meas.	Subcritical							Added Boron, Sample C-22 IX-3 Saturated with Boron
107	55	Not. Meas.	95							Boron Removal via BIX to obtain 95" Critical Height
108	55	22.891	95.30	108-1	54.175	0.750	0.5382	0.7176	95.67	55 Channels Cooled, ∿97% D_2O. Sample, C-23
				108-2	43.698	0.870	0.6175	0.7097	95.73	
				108-3	87.972	0.545	0.3850	0.4064	95.57	

Core Load & Mult. No.	Core Description No. F.E.	Conc. B-10	Critical Height	Period No. Pa-	Rising Period (sec)	ΔH (in)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	(Ave HT)	Remarks
109	55	22.891	95.32	109-1	45.141	0.765	0.6051	0.7910	95.70	Channel 1651 Dry
				109-2	60.396	0.625	0.5007	0.8001	95.63	
				109-3	78.463	0.533	0.4179	0.7841	95.59	
110	55	22.891	95.51	110-1	58.687	0.678	0.5104	0.7528	95.85	Channel 1651 Filled with ∿94% D_2O
				110-2	86.925	0.513	0.3884	0.7571	95.77	
				110-3	117.560	0.418	0.3104	0.7426	95.72	
111	55	22.891	95.37	111-1	103.230	0.463	0.3423	0.7393	95.60	Channel 1651 Filled with ∿60% D_2O
				111-2	54.757	0.720	0.5344	0.7422	95.73	
				111-3	71.405	0.605	0.4465	0.7380	95.67	
112	55	22.891	94.87	112-1	29.349	0.90	0.7821	0.8690	95.30	Channel 1651 Filled with H_2O (∿3% D_2O)
				112-2	39.658	0.74	0.6556	0.8859	95.22	
				112-3	57.066	0.60	0.5200	0.8667	95.15	
113	55	22.891	95.45	113-1	44.335	0.850	0.6120	0.7200	95.87	Channel 1651 Filled with ∿97% D_2O. Repeat of (108).
				113-2	53.857	0.750	0.5403	0.7204	95.82	
				113-3	84.623	0.550	0.3959	0.7198	95.72	
114	55	22.891	95.51	114-1	58.028	0.690	0.5143	0.7454	95.86	Zr-Nb Samples charged into PT- 1641
				114-2	39.478	0.895	0.6574	0.7345	95.96	
				114-3	80.70	0.550	0.4096	0.7447	95.79	
115	55	Not meas.	Subcritical							Charged Oscillator into C-1447 Chamber into C-1352, Sample C-24
116	55	Not meas.	>105							Boron Removal
117	55		95							Reduced Oscillator Worth, Moved OSC to C-1644
118	55		>103							Added Boron
119	55	20.430	100.1							Removed Boron to obtain ∿100" Critical Level for Sample C-25
				119-1	59.918		0.5034	Level @	107.16(min)	Oscillator Measurement
				119-2	64.590		0.4785		107.14(max)	Oscillator worth meas. (∿0.802¢)
120	55	20.430	95.9	120-1	22.196	1.77	0.9103	0.5143	96.78	Base Measurement, channel 1546 Filled with Primary Grade D_2O
				120-2	31.625	1.45	0.7495	0.5169	96.625	
				120-3	50.397	1.10	0.5641	0.5128	96.45	
121	55	20.430	96.01	121-1	42.203	0.99	0.6310	0.6374	96.51	Channel 1546 Dry.
				121-2	58.171	0.79	0.5134	0.6499	96.41	
				121-3	89.432	0.59	0.3804	0.6447	96.31	
122	55	20.430	96.12	122-1	66.124	0.83	0.4709	0.5673	96.54	Channel 1546 Filled with 93.58% D_2O
				122-2	47.725	1.03	0.5841	0.5671	96.64	
				122-3	90.309	0.67	0.3777	0.5637	96.45	
123	55	20.430	96.13	123-1	74.809	0.735	0.4322	0.5880	96.50	Channel 1546 Filled with 58% D_2O
				123-2	55.823	0.895	0.5277	0.5896	96.58	
				123-3	91.089	0.635	0.3754	0.5912	96.45	
124	55	20.430	95.75	124-1	61.315	0.70	0.4956	0.7080	96.10	Channel 1546 Filled with H_2O
				124-2	51.565	0.80	0.5558	0.6948	96.15	
				124-3	88.988	0.55	0.3818	0.6942	96.03	
125	55	20.430	95.70	125-1	32.328	1.20	0.7401	0.6167	96.30	Channel 1546 Filled with Primary Grade D_2O
				125-2	51.298	0.90	0.5577	0.6196	96.15	
				125-3	94.68	0.60	0.3649	0.6081	96.00	

Core Load & Mult. No.	Core Description No. F.E.	Core Description Conc. B-10	Critical Height	Period No. Pa-	Rising Period (sec)	ΔH (in)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	Level Coefficient (Ave HT)	Remarks
126	55	20.430	96.90	126-1	72.394	1.10	0.4423	0.4021	97.45	Base Measurements
				126-2	46.466	1.50	0.5942	0.3961	97.65	
				126-3	102.150	0.91	0.3450	0.3791	97.36	
				126-4	43.418	1.60	0.6200	0.3875	97.70	
127	55	20.430	96.87	127-1	39.676	1.65	0.6554	0.3972	97.70	Charged F.E. 6700 (Pelleted Element)
				127-2	67.735	1.13	0.4631	0.4098	97.44	in place of Std. Element (F.E. 6062)
				127-3	92.545	0.93	0.3711	0.3990	97.34	in Channel 1649
				127-4	39.946	1.63	0.6527	0.4004	97.69	
128	55	20.430	96.94	128-1	68.789	1.06	0.4582	0.4323	97.47	Replaced F.E. 6062 into 1649
				128-2	42.26	1.51	0.6305	0.4175	97.70	Removed 7600. Installed F.E. 6701 in
				128-3	115.91	0.76	0.3138	0.4129	97.32	1756 in place of F.E. 6518.
129	0	20.430								Core defueled for installation of temporary process tube plugs.
130	55	20.705	96.62	130-1	47.184	0.68	0.5884	0.8653	96.96	Central 55 Channels Void 29 peripheral
				130-2	77.803	0.48	0.4204	0.8758	96.86	Tubes Filled with Primary Grade D_2O
				130-3	126.05	0.34	0.2943	0.8656	96.79	C-26
131	55	20.705	Subcritical							Boron Removal, C-27
132	55	20.182	96.05	132-6	86.312	0.93	0.3903	0.4197	96.52	Central 55 Channels Filled with H_2O
				132-7	43.146	1.48	0.6224	0.4205	96.79	29 Peripheral Tubes Filled with
				132-8	53.638	1.28	0.5417	0.4232	96.69	Primary Grade D_2O
133	55	19.894	91.58	133-1	28.176	0.44	0.8002	1.8186	91.80	Central 55 Channels Void 29 Peripheral
				133-2	63.902	0.27	0.4820	1.7851	91.72	Tubes Filled with Primary Grade D_2O
				133-3	115.98	0.17	0.3136	1.8447	91.67	C-28 not same as C-27
				133-4	Thrown out					
134	55	19.894	91.10	134-1	66.422	0.30	0.4694	1.5647	91.25	All Channels Filled with Primary
				134-3	26.014	0.55	0.8362	1.5204	91.32	Grade D_2O
				134-4	99.103	0.23	0.3528	1.5339	91.22	

The following measurements were performed to evaluate the various possible configurations of the FERTF Test Assembly. Measurements were made by approach-to-critical only.

Core Load & Mult. No.	F.E. Number	Concentration 10-B (wppm)	Critical Height (in.)	Status of PC-1550	Remarks
135	54	19.894	110 +	H_2O coolant, no fuel	54 HPD elements surrounding PC-1550 in loadings 135-156
136	54	21.252	Subcritical	" " " "	
137	54	18.864	98.75	" " " "	
138	54	18.864	90.0	Coolant void, no fuel	
139	54	18.864	91.1	H_2O coolant, 6-rod FERTF Assembly, UO_2-2 wt% PuO_2	
140	54	18.864	90.03	Coolant void , 6 rod FERTF Assembly, UO_2-2 wt% PuO_2	
141	54	18.864	99.18	H_2O coolant, no fuel, repeat of Mult. 137	
142	54	-	-	-	Not completed
143	54	19.342	94.61	H_2O coolant, 6 rod FERTF, UO_2-2 wt% PuO_2	
144	54	19.342	92.98	H_2O coolant, 6 rod FERTF, UO_2-3 wt% PuO_2	
147	54	19.342	110 +	H_2O coolant, 6 rod FERTF, UO_2 (with Au pins)	
148	54	19.371	98.53	H_2O coolant, 6 rod FERTF, UO_2 (with Au pins)	
149	54	19.371	92.75	Coolant void, 6 rod FERTF, UO_2 (with Au pins)	
150	54	19.371	98.13	H_2O coolant, 6 rod FERTF, UO_2	
152	54	19.371	93.25	Coolant void, 8 rod FERTF, UO_2-2 wt% PuO_2, Al Plug	
153	54	19.371	94.18	H_2O coolant, 8 rod FERTF, UO_2-2 wt% PuO_2, Al Plug	
154	54	-	-		Not Completed
155	54	19.212	93.94	Coolant void, 8 rod FERTF, UO_2-2 wt% PuO_2, 304-SS Plug	
156	54	19.212	95.18	H_2O coolant, 8 rod FERTF, UO_2-2 wt% PuO_2, 304-SS Plug	
157	54	19.212	92.51	19-rod HPD in D_2O, in PC-1550	6-rod FERTF in PC-1849, coolant void, UO_2-2 wt% PuO_2
158	54	19.212	93.20	19-rod HPD in D_2O, in PC-1550	6-rod FERTF in PC-1849, H_2O coolant UO_2-2 wt% PuO_2
159	54	19.212	92.38	19-rod HPD in D_2O, in PC-1550	6-rod FERTF in PC-1847, H_2O coolant UO_2-2 wt% PuO_2

Additional FERTF Assembly tests:

14-4-2	57	18.089	92.75	H_2O coolant, 8-rod FERTF, 304-SS sleeve, no insert, UO_2-2 wt% PuO_2 Hastelloy Plug	
14-4-4	57	18.089	98.08	H_2O coolant, No fuel assembly	
14-4-5	57	18.089	98.95	D_2O coolant, No fuel assembly	
14-4-6	57	18.089	91.41	Coolant void,	8 rod FERTF, 304-SS Sleeve, no insert UO_2-2 wt% PuO_2 Hastelloy Plug
14-4-7	57	18.089	92.40	H_2O coolant	8 rod FERTF, 304-SS Sleeve, no insert UO_2-2 wt% PuO_2, Hastelloy Plug
14-4-8	57	18.089	93.55	H_2O coolant	8 rod FERTF, 304-SS Sleeve, 304-SS insert UO_2-2 wt% PuO_2, 304-SS Plug
14-4-9	57	18.089	94.12	H_2O coolant	8-rod FERTF, 304-SS sleeve, 304-SS insert UO_2-2 wt% PuO_2, Hastelloy plug
14-5-2	57	18.089	92.20	H_2O coolant,	8 rod FERTF, 304-SS sleeve, no insert, UO_2-2 wt% PuO_2, 304-SS Plug

The following measurements were performed during Interim Critical Test Series 1, 2, and 3. Measurements were made by approach-to-critical only.

Mult No.	Core Loading Number	^{10}B Conc. (wppm)	Critical Height (in.)	Remarks
Series 1				
1-1	1-1	-	∿ 85.5	Not Completed
1-2	1-1	-	∿ 89.1	Not Completed
1-3	1-1	-	∿ 89.5	Not Completed
1-4	1-1	-	∿ 91.0	Not Completed
1-5	1-1	16.32	90.67	Measurement of Batch Core excess reactivity
1-6	1-1	16.80	95.75	Measurement of Batch Core excess reactivity
1-7	1-1	17.30	100.52	Measurement of Batch Core excess reactivity
1-8	1-2	-	>100	Not Completed
1-9	1-2	-	∿ 99	Not Completed
1-10	1-2	16.71	98.01	No FE in PC-1748
1-11	1-3	16.71	91.19	Unirradiated FE 6506 in PC-1748
1-12	1-4	16.71	91.93	Original Batch Core configuration
1-13	1-5	17.15	92.35	Poison holder in C-1553
1-14	1-6	17.15	92.53	1 poison section
1-15	1-7	17.15	92.72	3 poison sections
1-16	1-8	17.15	92.93	5 poison sections
1-17	1-9	17.15	92.39	Poison holder empty
1-18	1-9	17.15	100.06	Noise at 100" critical
1-19	1-10	17.64	∿109.3	Subcritical noise
Series 2				
2-1	2-1	12.07	89.80	Measurement of Batch Core excess reactivity
2-2	2-1	13.47	94.55	Measurement of Batch Core excess reactivity
2-3	2-1	14.06	100.38	Measurement of Batch Core excess reactivity
2-4	2-2	13.15	99.95	No FE in PC-1748
2-5	2-3	13.15	91.95	Unirradiated FE 6506 in PC-1748
2-6	2-4	13.15	92.84	Fuel element "shuffle"
2-7	2-5	13.15	93.30	Original Batch Core configuration
2-8	2-6	-	∿96	Oscillator installed in core
2-9	2-6	13.75	98.95	Oscillator measurement performed
2-10	2-7	-	-	Scrammed out
2-11	2-7	-	-	Scrammed out
2-12	2-7	13.78	94.85	Bare Al "poison" rod
2-13	2-8	13.78	95.21	Bare boron poison rod (0.0056 g/cm^3)
2-14	2-9	13.78	97.75	Bare boron poison rod (0.0853 g/cm^3)
2-15	2-10	13.78	95.64	Bare Hafnium poison rod (0.52 g/cm^3)
2-16	2-11	13.78	97.60	Bare Hafnium poison rod (5.11 g/cm^3)
2-17	2-12	13.86	96.54	Cd-covered hafnium poison rod
2-18	2-13	13.86	95.50	Cd-covered boron poison rod
2-19	2-14	13.86	94.98	Cd-covered aluminum "poison" rod
2-20	2-15	14.22	99.38	Shuffled Batch Core configuration
Series 3				
3-1	3-1	11.29	89.95	Measurement of Batch Core excess reactivity
3-2	3-1	12.06	94.94	Measurement of Batch Core excess reactivity
3-3	3-1	12.33	99.71	Measurement of Batch Core excess reactivity
3-4	3-2	12.33	>100	Not completed
3-5	3-2	11.85	99.35	No FE in PC-1748
3-6	3-3	11.85	92.15	Unirradiated FE 6506 in PC-1748
3-7	3-4	11.85	90.80	UO_2-4 wt% PuO_2 FE in PC-1748
3-8	3-5	11.85	93.95	CVTR element in PC-1748
3-9	3-6	11.85	93.79	Original Batch Core configuration
3-10	3-7	11.85	98.05	Cd-covered hafnium poison rod
3-11	3-8	11.85	97.38	Cd-covered boron poison rod
3-12	3-9	11.85	97.12	Cd-covered aluminum "poison" rod
3-13	3-10	11.85	95.70	Bare hafnium poison rod
3-14	3-11	11.85	95.85	Bare boron poison rod
3-15	3-12	11.85	93.82	Bare aluminum "poison" rod

The following measurements were performed during Interim Critical Test Series 4.

Core Load & Mult. No.	Core Description No. F.E.	Core Description Conc. B-10 (wppm)	Critical Height (in)	Period No. P –	Rising Period (sec)	ΔH (in.)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	Level Coefficient (Ave Ht)	Remarks
4-1	55	9.82	90.33	4-1-1	142.33	0.095	0.262	2.76	90.378	Measurement of Batch Core excess reactivity
				4-1-2	59.345	0.17	0.499	2.93	90.415	
				4-1-3	32.935	0.26	0.724	2.78	90.46	
				4-1-4	74.639	0.12	0.426	3.55	90.39	
4-2	55	10.58	95.53	4-2-1	179.99	0.23	0.218	0.95	95.645	Measurement of Batch Core excess reactivity
				4-2-2	119.09	0.315	0.301	0.96	95.668	
				4-2-3	51.94	0.57	0.545	0.96	95.815	
				4-2-4	34.96	0.75	0.699	0.93	95.905	
4-3	55	10.94	99.57	4-3-1	–	0.43	–	–	–	Inconclusive data
				4-3-2	103.55	1.23	0.335	0.27	100.185	
				4-3-3	62.75	1.93	0.480	0.25	100.535	
				4-3-4	42.49	2.73	0.620	0.23	100.935	
4-4	55	10.63	101.41	4-4-1	202.56	1.59	0.197	0.12	102.205	Oscillator in C-1447
				4-4-2	141.06	1.59	0.264	0.17	102.205	
4-5	31	?	?	–	–	–	–	––	––	Oscillator in C-1447
4-6	31	?	∿101.2	–	–	–	–	–	––	Oscillator in C-1447
4-7	31	?	101.50	4-7-1	117.93	1.50	0.304	0.20	102.25	Oscillator in C-1447
				4-7-2	222.07	1.50	0.183	0.12	102.25	
4-8	31	?	Scram	–	–	–	–	–	–	Oscillator in C-1245
4-9	31	5.94	100.95	4-9-1	110.92	0.855	0.318	0.37	101.378	Oscillator in C-1245
				4-9-2	105.86	0.86	0.330	0.38	101.38	
4-10	31	6.12	99.99	4-10-1	119.24	0.83	0.301	0.36	100.405	Measurement of 31-element core excess reactivity
				4-10-2	62.84	1.50	0.480	0.32	100.74	
				4-10-3	43.28	2.06	0.614	0.30	101.02	
4-11	31	5.78	94.91	4-11-1	56.90	0.49	0.513	1.05	95.055	Measurement of 31-element core excess reactivity
				4-11-2	61.68	0.43	0.485	1.13	95.025	
				4-11-3	100.61	0.315	0.343	1.09	94.968	
				4-11-4	37.58	0.608	0.669	1.10	95.114	
4-12	31	5.12	89.94	4-12-1	97.92	0.16	0.344	2.15	90.02	Measurement of 31-element core excess reactivity
				4-12-2	85.88	0.16	0.385	2.41	90.02	
				4-12-3	40.70	0.273	0.637	2.34	90.077	
				4-12-4	35.27	0.30	0.694	2.31	90.09	
				4-12-5	30.74	0.329	0.752	2.29	90.105	
4-13	19	?	?	–	–	–	–	–	–	
4-14	19	2.26	99.43	4-14-1	100.79	0.70	0.342	0.49	99.78	Measurement of 19-element core excess reactivity
				4-14-2	57.06	1.09	0.512	0.47	99.975	
				4-14-3	29.19	1.64	0.775	0.47	100.25	

Core Load & Mult. No.	Core Description No. F.E.	Conc. 10-B (wppm)	Critical Height (in)	Period No. P -	Rising Period (sec)	ΔH (in.)	Δk_{eff} (milli-k)	Level Coefficient (mk/in)	(Ave Ht)	Remarks
4-15	19	2.05	94.93	4-15-1	65.99	0.39	0.464	1.19	95.125	Measurement of 19-element core excess reactivity
				4-15-2	54.34	0.438	0.529	1.21	95.149	
				4-15-3	36.25	0.57	0.583	1.02	95.215	
4-16	19	1.59	89.98	4-16-1	56.99	0.23	0.513	2.23	90.095	Measurement of 19-element core excess reactivity
				4-16-2	128.14	0.12	0.285	2.38	90.04	
				4-16-3	35.22	0.31	0.695	2.24	90.135	
4-17	19	1.59	89.90	4-17-1	91.50	0.196	0.368	1.88	89.998	Oscillator support pedestal removed
				4-17-2	55.93	0.275	0.519	1.88	90.038	
				4-17-3	38.41	0.345	0.659	1.91	90.073	
4-18	19	1.59	91.25	-	-	-	-	-	-	Oscillator in C-1245
4-19	19	?	Scram	-	-	-	-	-	-	Oscillator in C-1245
4-20	19	?	Scram	-	-	-	-	-	-	Oscillator in C-1245
4-21	19	?	>104	-	-	-	-	-	-	Oscillator in C-1245
4-22	19	2.31	103.57	4-22-1	137.48	1.445	0.270	0.19	104.293	Oscillator in C-1245
				4-22-2	132.29	1.425	0.278	0.19	104.283	

APPENDIX II

Analysis of ^{10}B Concentrations in D_2O Moderator By Reactivity Measurements

A precise knowledge of boron concentration in the moderator during a given set of experiments is essential to the analysis of the experiment. The Thermal Test Reactor was found to have sufficient sensitivity in the central thermal column to give the desired accuracy using 400 gram samples of borated D_2O.[8] Precise standard samples were prepared using ^{10}B enriched boric acid and 99.8% D_2O. The reactivity worths vs concentration of ^{10}B in the standards were fitted to a polynomial which included a second-order term to account for spatial self-shielding, using program LEARN.[5,6] This equation was solved for the unknown concentrations from ^{10}B worths measured in the TTR.

Flux shape and self-shielding considerations led to the choice of cylindrical sample containers of thin polyethylene. The samples were 6.4 cm in diameter and 11.4 cm high. Bottles were weighed empty to obtain a set of six having nearly equal weights to use for the standard solutions. Those to be used for standards plus one of the heaviest bottles were tested in the TTR to see if there were detectable differences in polyethylene worth. No significant variations were detected.

Two bottles with a large weight difference were filled with pure D_2O and their reactivity difference measured to obtain a correction for bottle weight differences. This value was -0.16 ± 0.03 ¢/gram of polyethylene as compared to the empty worth of -0.128 ± 0.027 ¢/gram. The full bottle worth correction was not required to be known very well because the total spread in bottle weights was approximately 2 grams.

The standard solutions contained 0.995, 5.08, 9.98, 15.0, 19.9 and 24.9 wppm ^{10}B in D_2O. Reactor periods were run with alternately pure and borated D_2O samples in the TTR. Differences from the base period reactivities with pure D_2O were corrected for bottle weights and reactor drift. The points calculated using the fitted polynomial agreed with the ^{10}B worths of the standard samples used in the fitting program to within 1.5%. The standard curve is shown in Figure A-II-1.

As a further test on the validity of the standards, five of them were analyzed by the Analytical Chemistry Laboratory using a mass spectrograph and isotopic dilution methods. Four of these results agreed within 0.5% and one within 1.5% of the quoted concentrations. The same analysis on 99.8% D_2O yielded 0.14 wppm ^{10}B with approximately 50% uncertainty.

Moderator samples taken during the critical tests were measured in exactly the same way as were the standard samples, with one exception. For the measurement of the higher concentrations, the base period measurements used the 19.9 wppm standard. The use of this standard for base periods gave more precise comparative reactivity differences because there was a smaller difference between the base and unknown reactivities. The accuracy of the results is then based on the known reactivity worth of the 19.9 wppm standard. The measurements are indicated in Figure A-II-1, and the results are listed in Table A-II-I.

These reactivity techniques have proven to be a very sensitive and precise method to determine the effective ^{10}B concentration in borated D_2O samples. The ^{10}B concentration of an unknown sample can be determined to ± 0.05 wppm within the range of the standard curve.

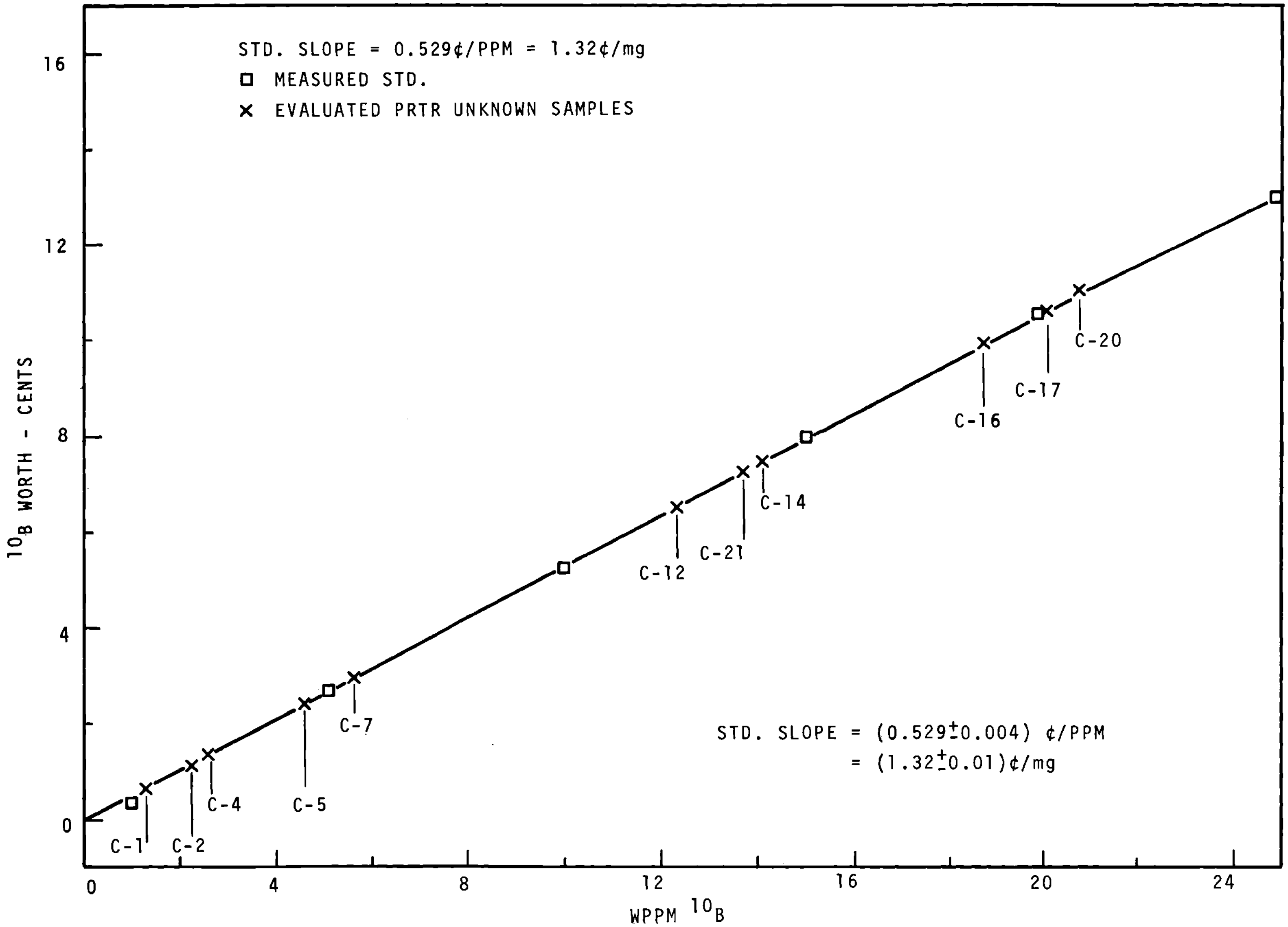

FIGURE A-II-1. REACTIVITY WORTH OF ^{10}B - D_2O SOLUTION: 400 GRAM SAMPLES IN THERMAL TEST REACTOR

Table A-II-I

Critical Test Boron Results

Sample No.	Measured wppm	Loading #	Reactor Condition # Clusters	Reactor Condition Critical Height (in.)	Remarks
C-0	-0.008	A-7	9	97.725	Initial Critical
1	1.231	A-20	12	79.1	
2	2.160	A-22	12	94.500	
3	2.638	A-23	12	102.300	
3 (rerun)	2.628	A-23	12	102.300	
4	2.554	A-25	12	102.600	
5	4.510	A-35	16	86.246	
6	5.430	A-36	16	93.230	
7	5.567	A-37	16	94.850	
8	8.554	A-49	23	85.410	
8 (rerun)	8.613	A-49	23	85.410	
9	10.560	A-50	23	>110	
10	10.162	A-51	23	95.390	
11	13.830	A-61	31	93.400	
12	12.315	A-63	31	100.300	Primary Voided
13	14.328	A-65	31	101.50	Primary Filled
14	14.135	A-66	31	96.90	
15	16.217	A-81	55	77.6	
16	18.889	A-82	55	85.303	
17	20.176	A-85	55	93.05	
18	20.903	A-88	55	99.65	
18 (rerun)	20.726	A-88	55	99.65	
19	21.002	A-92	55	>108	
20	20.976	A-93	55	98.4	
21	13.739	A-105	55	99.5	Oscillator & Chambers in
22	not measured	A-106	55	>110	
23	22.891	A-108	55	95.45	
24	not measured	A-115	55		Oscillator & Chambers in
25	20.430	A-119	55	100.1	Oscillator & Chambers out
26	20.705	A-131	55	>110	
27	20.182	A-132	55	96.05	
28	19.894	A-133	55	91.56	Coolant H_2O to void
29	21.252	A-136	55	>110	H_2O-1550
30	18.864	A-137	55	98.75	
31	19.342	A-143	55	94.67	2 wt% FERTF in 1550
32	19.371	A-148	55	98.525	UO_2 FERTF in 1550
33	19.212	A-155	55	95.175	2 wt% FERTF in 1550
34	20.093	A-158	55	93.2	
35	18.089	A-159	55	92.4	

All measurements are ± 0.050 wppm ^{10}B.

APPENDIX III

Descriptions of Special Equipment

The locations of the various pieces of special equipment in the core are shown in Figure 1. The neutron source thimble and basket are shown in Figure A-III-1, where all vertical distances are measured from the bottom plate of the calandria. Since the bottom of the core is at a height of 34.5 in., the source is seen to be well below the core.

The moderator level probe is shown in Figure A-III-2. During the critical tests this probe was installed in the moderator in access opening P-1443.

The flux chamber thimble located in C-1352 was the same as the neutron source thimble shown in Figure A-III-1. The chambers in this thimble were removed after initial criticality, but the thimble remained throughout the critical testing program.

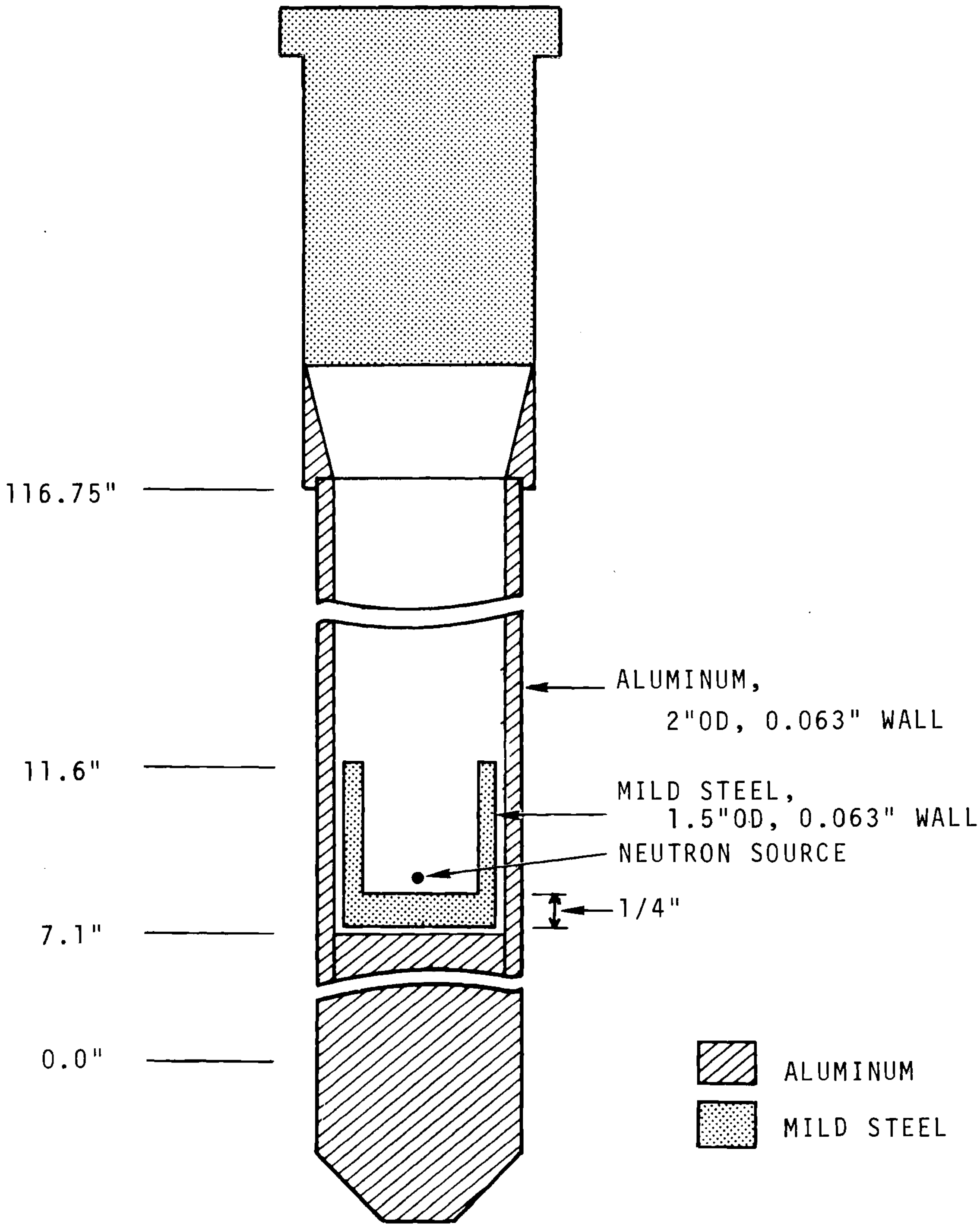

FIGURE A-III-1. NEUTRON SOURCE THIMBLE AND SOURCE BASKET

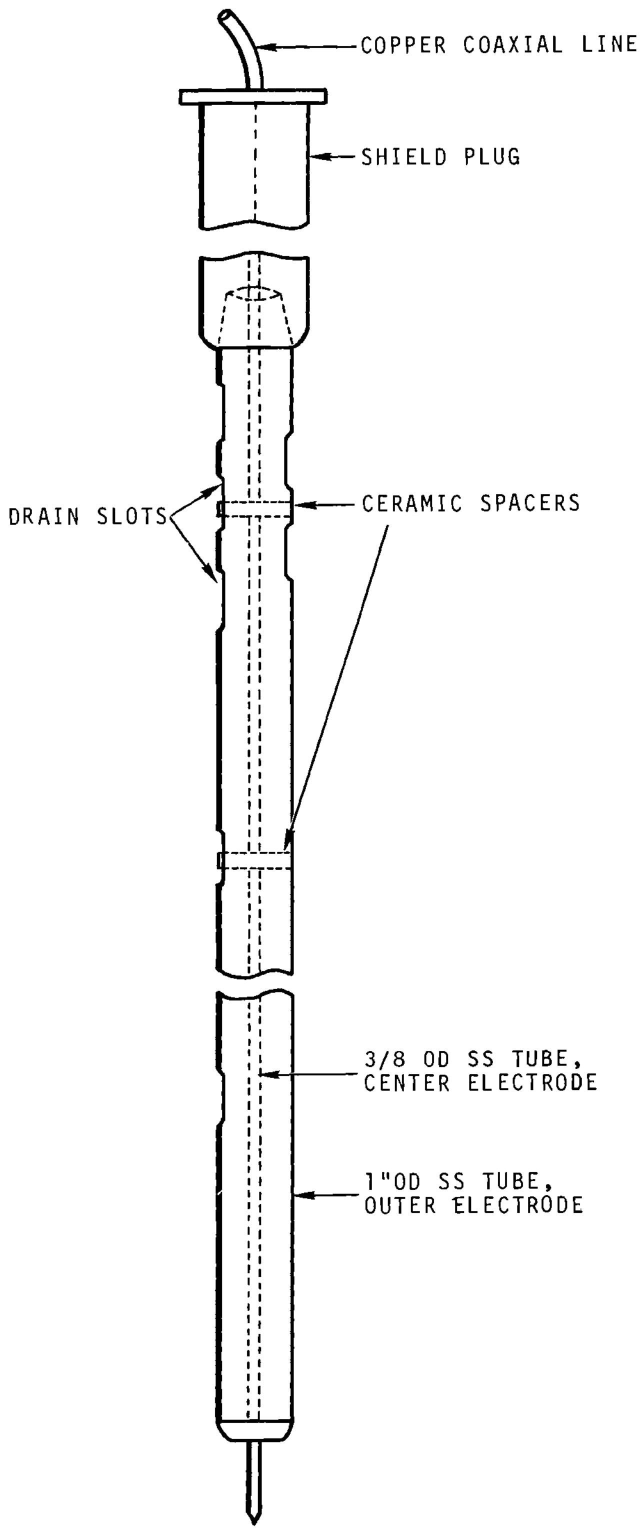

FIGURE A-III-2. MODERATOR LEVEL PROBE

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