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POWER TESTS WITH THE UO_2–2 wt% PuO_2 BATCH CORE IN THE PRTR

J. W. Kutcher and R. I. Smith

May 1970

AEC RESEARCH & DEVELOPMENT REPORT

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POWER TESTS WITH THE UO_2-2 wt% PuO_2 BATCH CORE IN THE PRTR

by

J.W. Kutcher and R.I. Smith

Reactor Physics Department

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ABSTRACT

As a part of the BATCH CORE EXPERIMENT in the Plutonium Recycle Test Reactor (PRTR), a series of power tests was conducted when the reactor loading was completed, at various powers up to 60 MW. These tests were designed primarily to verify predicted operational characteristics of the PRTR with the BATCH CORE loading, but also included those physics tests which required power operation.

The tests included flux traverses, operation of the traveling flux wire system, transient response of the reactor to the addition and removal of boron from the D_2O moderator, determination of reactor power, temperature coefficients of reactivity, kinetics parameters, and the growth and decay of fission product poisoning. Results of these tests are 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 full burnup experiment-the BATCH CORE EXPERIMENT[1]-was initiated in the Plutonium Recycle Test Reactor (PRTR). The basic fuel loading has remained essentially unperturbed throughout the irradiation 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[2] to determine the physics parameters of the new core. These tests were completed with a basic core of fifty-five elements which is referred to as the BATCH CORE.

A series of Power Tests was initiated when the reactor loading was completed. These tests, which were conducted at various power levels up to 60 MW, were primarily designed to verify predicted operational characteristics of the PRTR with the Batch Core loading, but also included those physics tests which required power operation.

The tests included the following:

- Measurement of the reactivity worth of peripheral fuel elements added to the basic BATCH CORE. The total worth of twenty-three peripheral fuel elements was 20.4 $\pm$ 0.1 milli-k.
- Traverses of the peripheral flux chambers to determine the flux distribution radially through the H_2O reflector. The difference between H_2O and D_2O coolant for the chambers was measured.
- Operation of the traveling flux wire system and its use to determine axial flux peaking as a function of critical moderator level over the range from 86 to 100 inches. The reactivity change induced by operation of the flux wire at its maximum speed was found to be less than 0.2 milli-k. The axial flux peaking was found to vary from 1.47 at a moderator level of 86 inches to 1.31 at a moderator level of 100 inches.
- Measurement of the transient response of the reactor to the addition and removal of boron from the D_2O moderator by the chemical shim system. In all cases an oscillatory reaction was observed due to the recirculation time of the moderator and the fact that complete mixing is not immediate.
- Detailed examination of the heat removal by each of the reactor coolant systems in order to determine total reactor power. The results indicate that 93.5 $\pm$ 1.2% of the total reactor is captured in the primary coolant system, while 4.8 ± 1.0% is captured in the moderator system, and 1.7 ± 0.5% in the reflector, shields, and FERTF.

- Measurement of the temperature coefficients of reactivity associated with the (fuel + coolant) system and the moderator system. The temperature coefficient of reactivity for the primary coolant system, over the range from 178°F to 356°F, was determined to be -0.025 ± 0.001 milli-k/°F. The temperature coefficient of reactivity for the moderator, containing 22.5 wppm ^{10}B, over the range from 92.5°F to 141°F, was determined to be + 0.101 ± 0.003 milli-k/°F.
- Measurement of β/ℓ, the ratio of the effective delayed neutron fraction to the neutron lifetime, with a sinusoidal reactivity oscillator. With 77 fuel elements in the core, 22.5 wppm ^{10}B in the moderator, and a reactor power of 1 MW, $\beta/\ell = 21.5 \pm 0.4\ sec^{-1}$.
- Reactivity measurements of the growth and decay of fission product poisoning. Equilibrium fission product poisoning at a power of 47 MW was determined to be 21.0 ± 0.1 milli-k. The shutdown xenon transient after equilibrium operation at 51 MW was determined to be 27.0 ± 0.1 milli-k.

Details of the performance of these tests and more complete results are presented in this report.

II. REACTOR DESCRIPTION

A. Reactor Type

The PRTR[3] is a vertical pressure tube reactor, heavy water moderated and cooled. The reactor is designed for diversified programs of fuel irradiation and testing. Heavy water as coolant and moderator has many safety advantages in 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; one central tube 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. In addition, there are thirteen 1.5-inch O.D. flux monitoring channels and 18 channels originally used for mechanical shims. The core plan and tube index is shown in Figure 1.

Each fuel channel contains a pressure tube of Zircaloy-2 inside the calandria tubes. This tube contains the fuel element and heavy water coolant. The nominal purity of the coolant is 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 calandria tube contains helium gas as a thermal barrier.

The 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 core essentially without radial reflection. This space is a dump

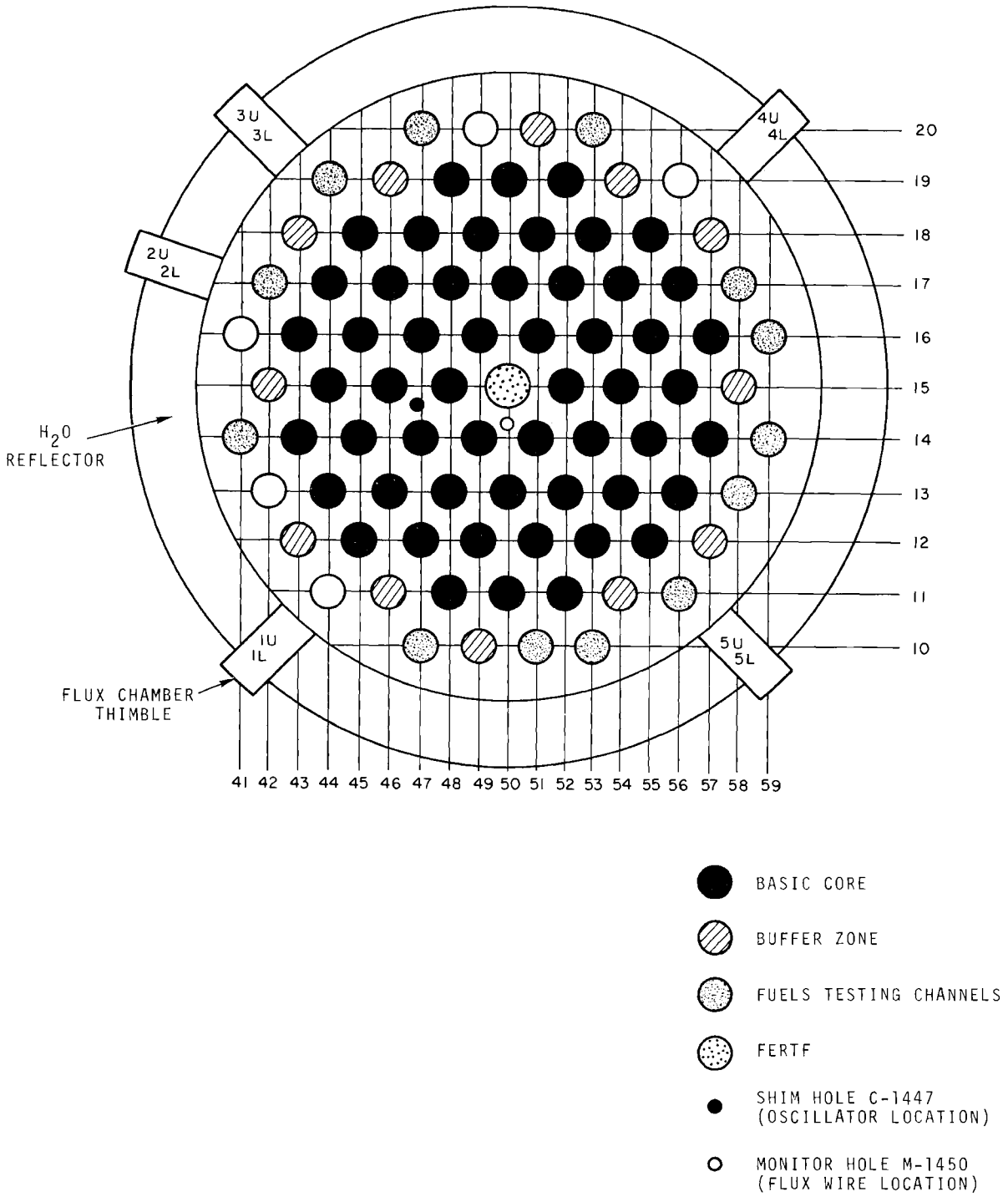

FIGURE 1. PRTR BATCH CORE CONFIGURATION

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. The heavy water moderator has a free surface at an annular weir, extending around the base of the calandria inside the dump chamber. The helium atomsphere 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.

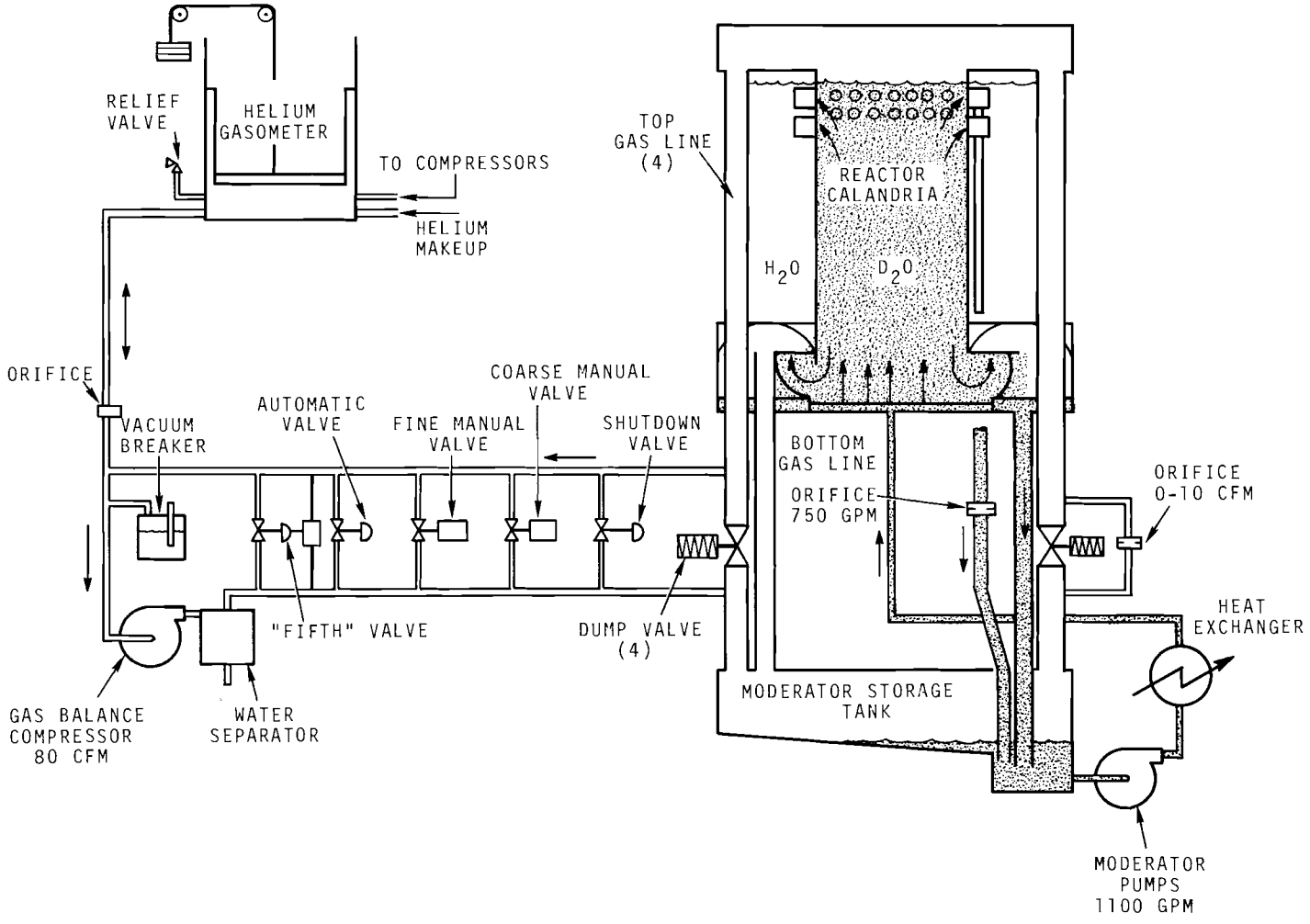

FIGURE 2. PRTR MODERATOR LEVEL CONTROL SYSTEM

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.[4] 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 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 ~29.440 kg natural uranium and ~599.6 grams plutonium.

F. Fuel Element Rupture Test Facility

For normal power operation of the PRTR, the central channel of the reactor is connected to the Fuel Element Rupture Test Facility (FERTF).[5] This facility is comprised of a pressurized light-water loop, entirely separate from the primary coolant system. It is normally used for the testing of experimental fuel elements. During the power tests reported here, the FERTF contained a 19-rod cluster of pelleted UO_2-1.94 wt% PuO_2, of the same dimensions as the standard elements. The enrichment of this element was chosen to match the macroscopic fission cross section of a standard UO_2-2 wt% PuO_2 element by compensating for its higher density.

III. EXPERIMENTAL RESULTS

A. Completion of Fuel Loading

The Batch Core Critical Tests[2] were all performed with the basic fifty-five element Batch Core. For power operation in the PRTR, however, approximately eighty fuel elements are charged. The loading of twenty-three peripheral elements was performed under test conditions to determine their incremental reactivity worth.

The peripheral fuel elements were charged in groups of three, as indicated in Table I. After charging the elements in each group, a standard approach to critical and measurement of the moderator level coefficient of reactivity were made. The results of these measurements are also listed in Table I.

The total reactivity worth of the twenty-three peripheral fuel elements was 20.4 ± 0.1 milli-k, compared with an excess reactivity of 263 milli-k for the basic fifty-five element Batch Core.

B. Radial Flux Distributions Through Reflector

Three of the neutron detectors used for reactor control were traversed along the thimbles which penetrate the shield and the annular reflector. Flux levels were recorded at incremental positions. The chambers used in the traverses were:

1. Startup fission chambers in side hole 1L (^{235}U-lined fission chamber).
2. Log N chamber #3 in side hole 3L (compensated ion chamber).
3. Linear chamber #1 in side hole 3U (uncompensated ion chamber).

In the case of the startup chambers, two traverses were made, one with H_2O coolant and one with D_2O coolant.

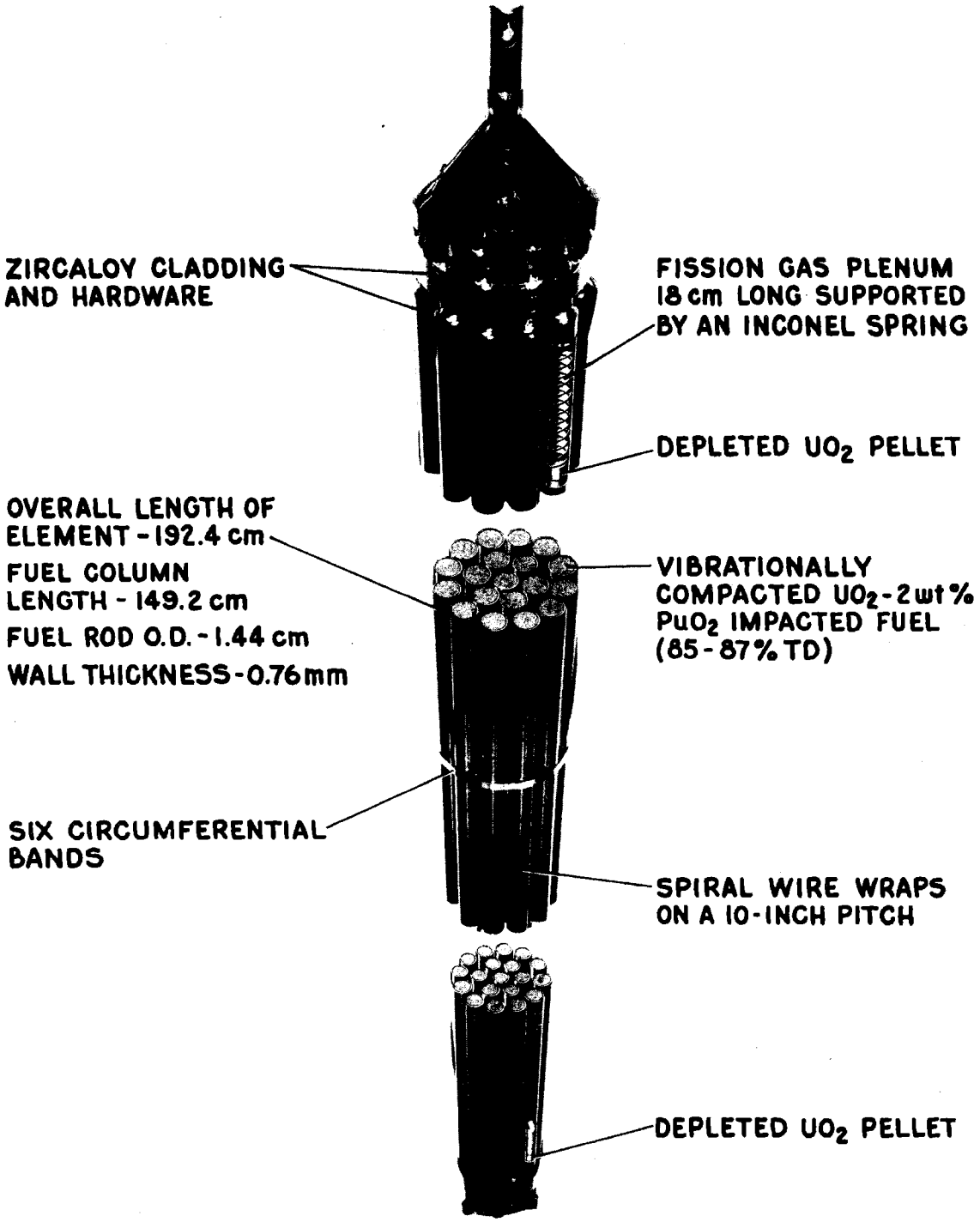

FIGURE 3

PRTR High Power Density Nineteen-Rod Cluster Fuel Element

TABLE I

REACTIVITY WORTH OF PERIPHERAL FUEL ELEMENTS

Group	Fuel Element No.	Fuel Type	Process Channel	Total No. FE in Core	Critical Height (inches)	Cumulative Reactivity Worth (milli-k)
Batch Core		UO_2-2 wt% PuO_2		55	91.15	——
1	6513	UO_2-2 wt% PuO_2	1558			
	6514	UO_2-2 wt% PuO_2	1954			
	6515	UO_2-2 wt% PuO_2	1946	58	88.80	4.4 ± 0.1
2	6512	UO_2-2 wt% PuO_2	1154			
	6511	UO_2-2 wt% PuO_2	1146			
	6510	UO_2-2 wt% PuO_2	1542	61	86.25	9.8 ± 0.1
3	6508	UO_2-2 wt% PuO_2	1243			
	6014	UO_2-2 wt% PuO_2	2051			
	6509	UO_2-2 wt% PuO_2	1843	64	84.86	13.2 ± 0.1
4	6013	UO_2-2 wt% PuO_2	1049			
	6056	UO_2-2 wt% PuO_2	1857			
	6012	UO_2-2 wt% PuO_2	1257	67	83.65	16.3 ± 0.1
5	6003	UO_2-2 wt% PuO_2	1742			
	6005	UO_2-2 wt% PuO_2	1358			
	6501	EBWR*	1051	70	82.55	19.3 ± 0.1

TABLE I (Continued)

Group	Fuel Element No.	Fuel Type	Process Channel	Total No. FE in Core	Critical Height (inches)	Cumulative Reactivity Worth (milli-k)
6	5226	UO_2-1 wt% PuO_2	1047			
	5227	UO_2-1 wt% PuO_2	1659			
	5224	UO_2-1 wt% PuO_2	1944	73	82.35	19.9 ± 0.1
7	5118	UO_2-0.5 wt% PuO_2	1156			
	1037	UO_2	1441			
	5142	UO_2-0.5 wt% PuO_2	1758			
	6006	UO_2-2 wt% PuO_2	2047			
	5186	UO_2-0.5 wt% PuO_2	2053	78	82.17	20.4 ± 0.1

*EBWR element is a 21-rod cluster of UO_2-1.5 wt% PuO_2

There are ten flux-chamber thimbles located at two elevations around the periphery of the reactor core (Figure 1). The thimbles or ports are constructed perpendicular to the reactor tank walls and pass through the biological and thermal shields and the annular H_2O reflector. A detailed cross section of a thimble is shown in Figure 4. The neutron detectors are enclosed in an aluminum housing which protects them from the H_2O circulated through the ports for cooling.

The results of the flux traverses are presented in Figures 5,6, and 7. The position coordinate represents distance between the reactor tank wall and detector housing (d distance in Figure 4). In Figure 5, note that the active length of the startup chamber is 6 inches, rather than the 11 1/2 inches indicated in Figure 4 for the log and linear chambers. The effect of D_2O coolant in reducing the thermal neutron "hump" can be clearly seen.

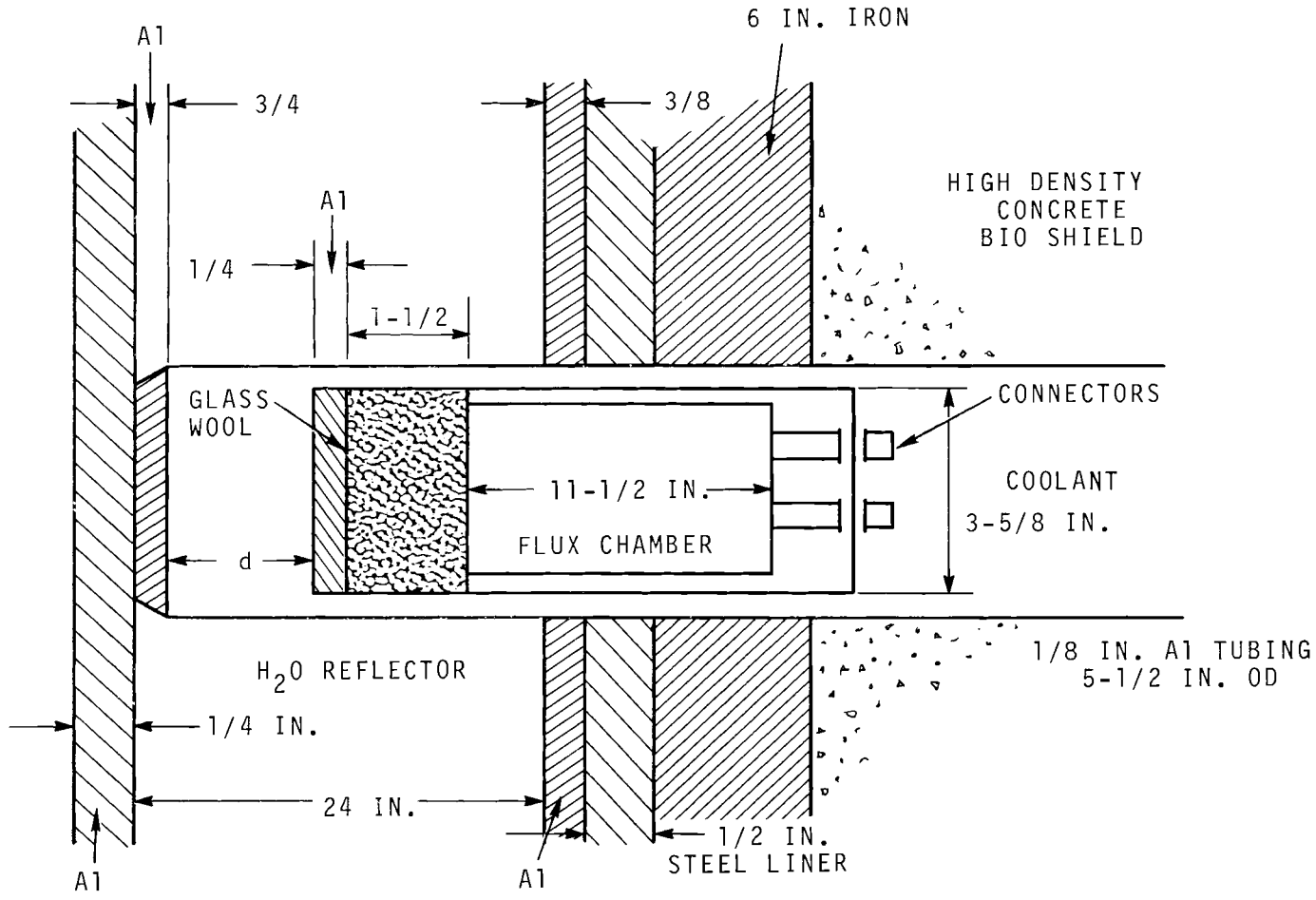

FIGURE 4. FLUX MONITOR THIMBLE

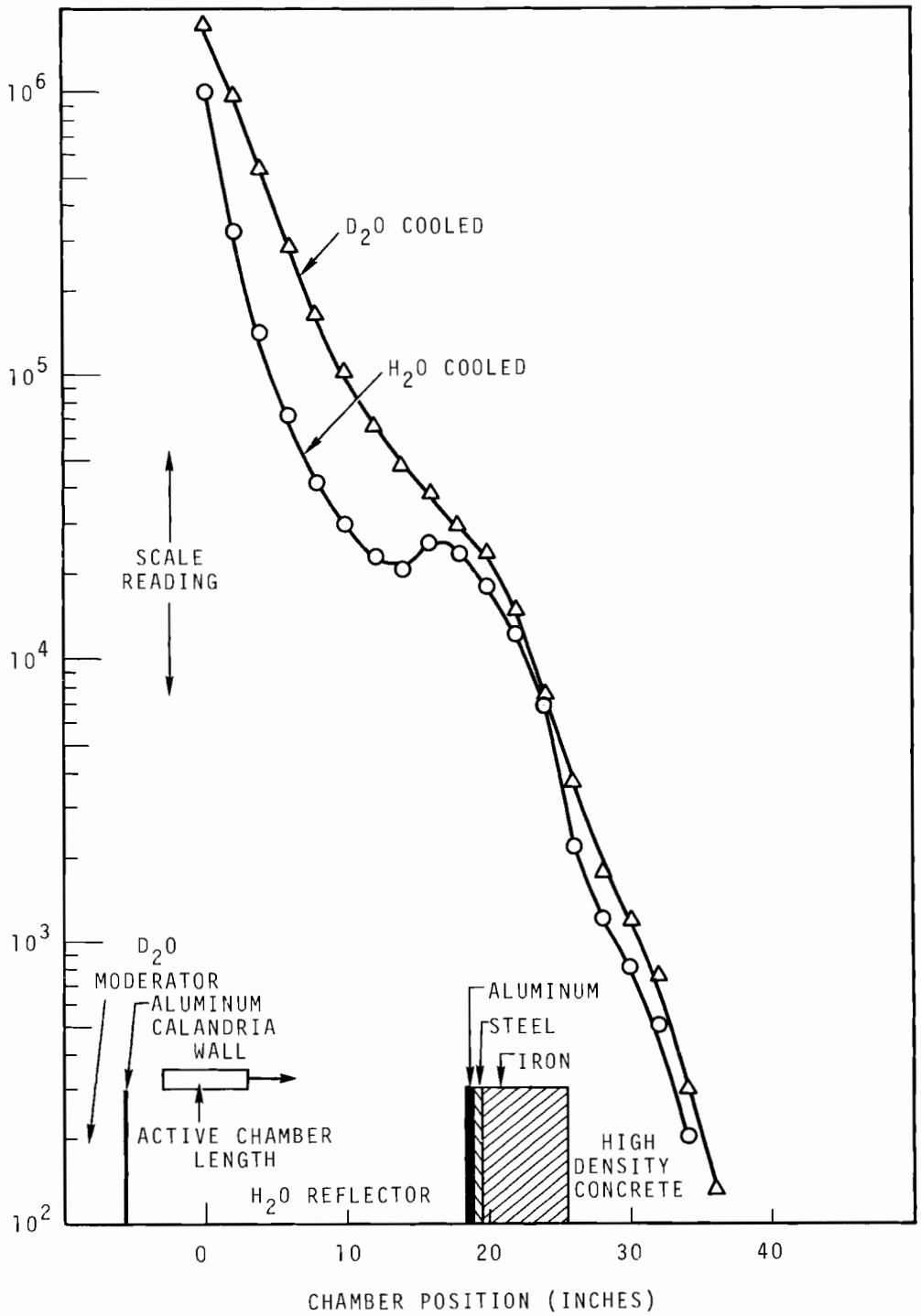

FIGURE 5. CHAMBER TRAVERSE STARTUP #1

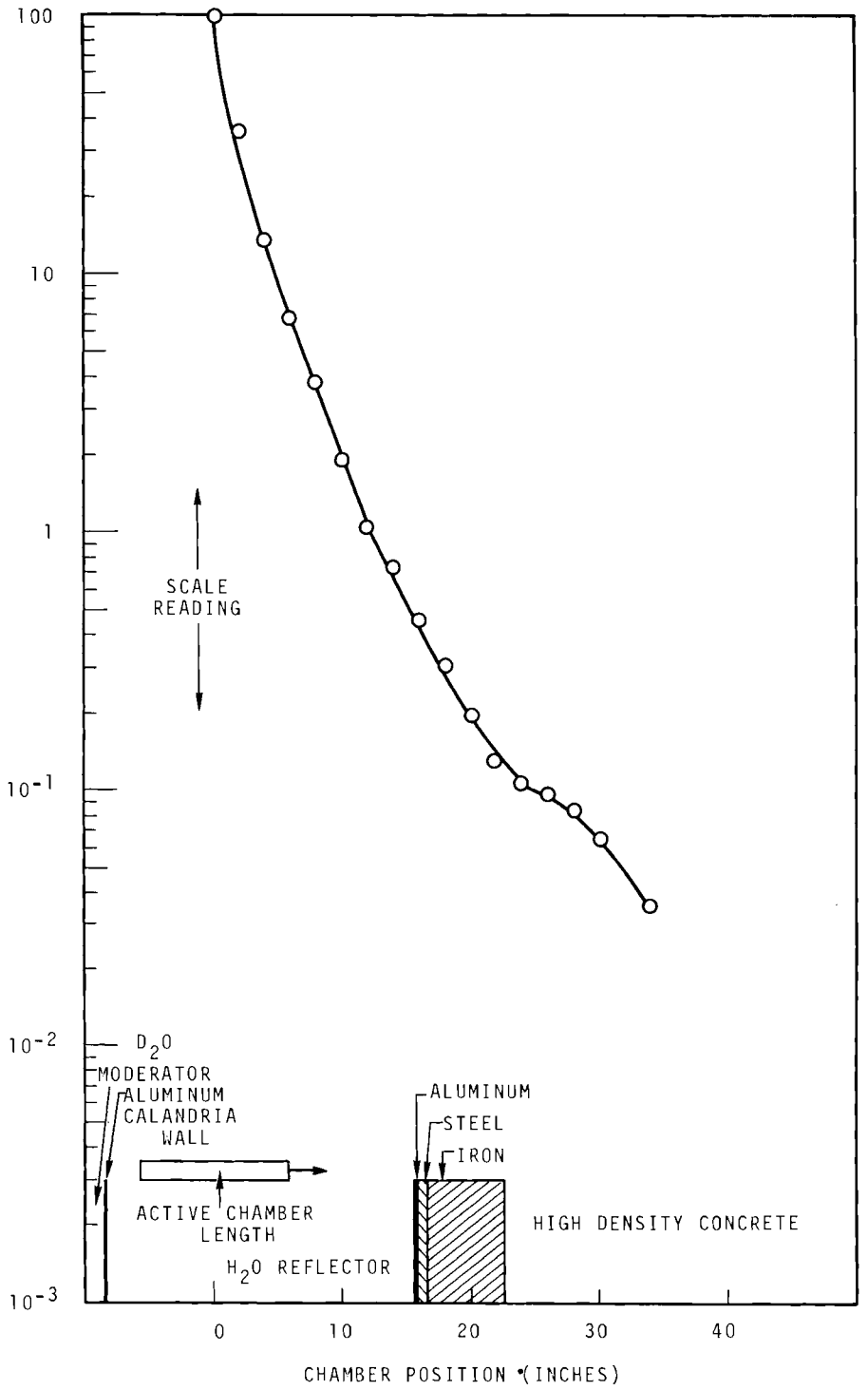

FIGURE 6. CHAMBER TRAVERSE LINEAR #1

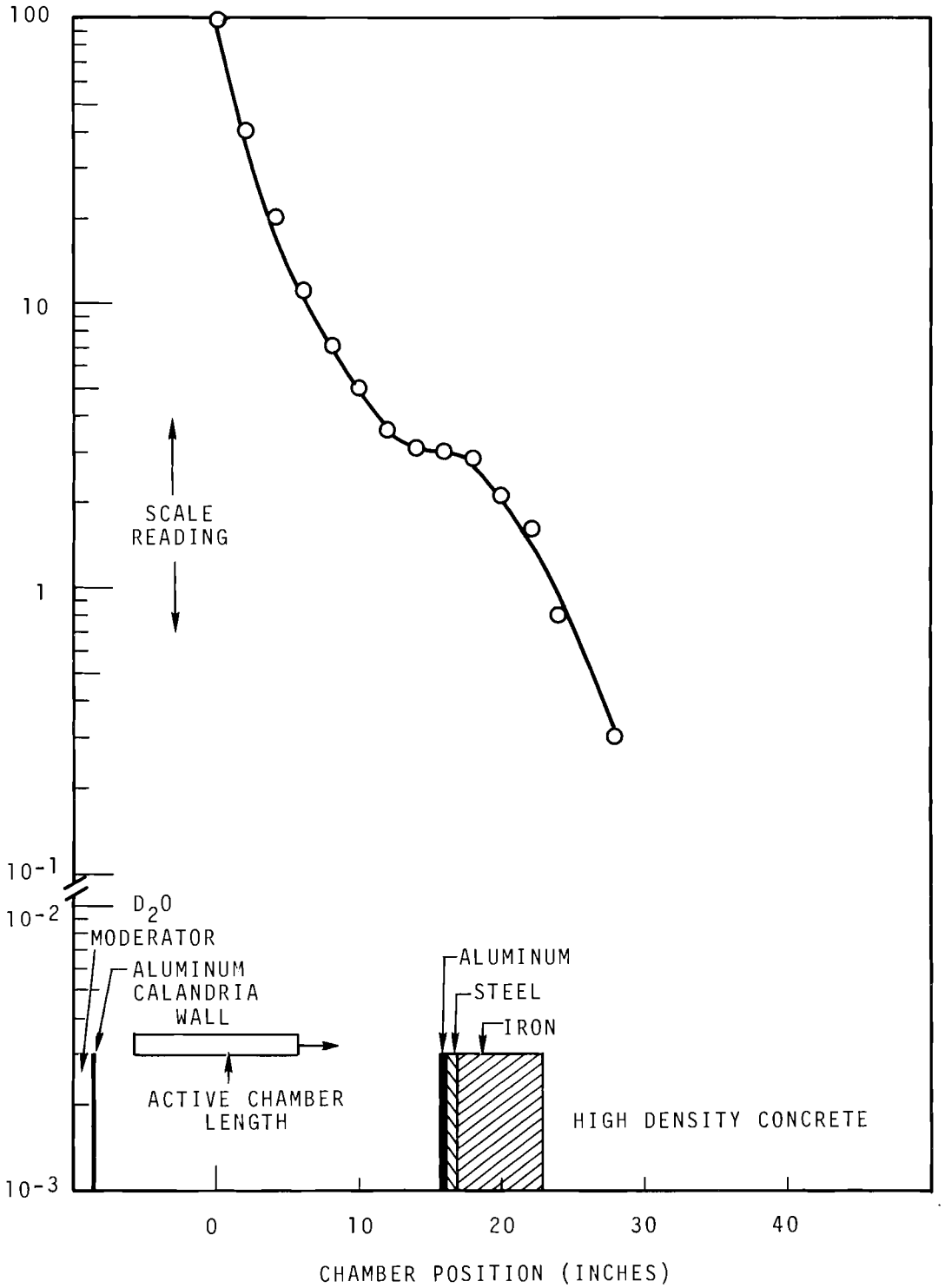

FIGURE 7. CHAMBER TRAVERSE LOG N #3

C. Axial Flux Distributions

The axial distribution of neutron flux at selected radial locations in the PRTR core is measured by activation of a traveling wire. This wire is made of titanium, which becomes activated with a half-life of 5.8 minutes. The wire is inserted into the core in one of several vertical monitor tubes, and irradiated for a period of time. It is then removed rapidly from the core and scanned as it passes in front of a NaI(Tl) detector. The short half-life enables the wire to be re-irradiated after a brief "cooling-off" period.

The magnitude of the reactivity changes associated with insertion and removal of the traveling wire were determined, in order to establish operational limits on rate-of-travel of the in-core segment. At a power level of 20 MW with 22.5 wppm ^{10}B in the moderator the reactor was placed on automatic control. The flux wire was inserted at maximum speed (2 ft/sec) into its position of maximum reactivity worth, left in that position for about three minutes, and then withdrawn at maximum speed (2 ft/sec). The automatic controller maintained reactor equilibrium by varying the moderator level. The critical moderator level as a function of time is shown in Figure 8. The maximum change in critical level was 0.30 ± 0.02 inches, corresponding to a reactivity change of 0.17 ± 0.02 milli-k, which did not create any control difficulties for reactor operation.

Once the conditions for its operation had been established, the traveling flux wire was used to measure axial flux distributions for various critical moderator heights ranging from 86 inches to 100 inches at a reactor power of 35 MW. The flux shapes for these two heights are shown in Figure 9. The difference between the thick bottom reflector and the thin top reflector may be clearly seen.

Values for axial buckling and top and bottom reflector savings have been derived from these data by fitting with a cosine function using the least squares fitting code LEARN.[6,7] At moderator level = 100 inches, the axial buckling B_z^2 was 2.66 ± 0.04m^{-2}. The bottom reflector savings was 22.75 ± 1.01 cm, while the top reflector savings was 21.11 ± 1.02 cm. At moderator level = 86 inches, the axial buckling B_z^2 was 3.60 ± 0.05 m^{-2}. The bottom reflector savings was 23.03 ± 0.78 cm. Top reflector savings was not defined in this case, because the moderator level was below the top of the fuel. However, one may infer an extrapolation length above the moderator level of 4.62 ± 0.81 cm.

In establishing reactor operating limits, the ratio of peak-to-average flux in the axial direction is a very important parameter. This is due to the fact that reactor operating limits are based on maximum specific rod power, such that no centerline melting shall occur. Since what is being monitored during operation is tube power, it is necessary to know the appropriate fuel element peaking factors. The axial peaking factor is a function of three variables: fuel burnup, boron concentration in the moderator, and moderator height, which is the most important. Since these quantities are in fact time dependent variables, it is essential that the axial flux distribution be measured regularly.

The ratio of peak-to-average flux in the axial direction, as a function of moderator height, is plotted in Figure 10. These measurements were made at boron concentrations between 17 and 19 wppm ^{10}B. (Boron was injected between measurements to raise the critical moderator height.)

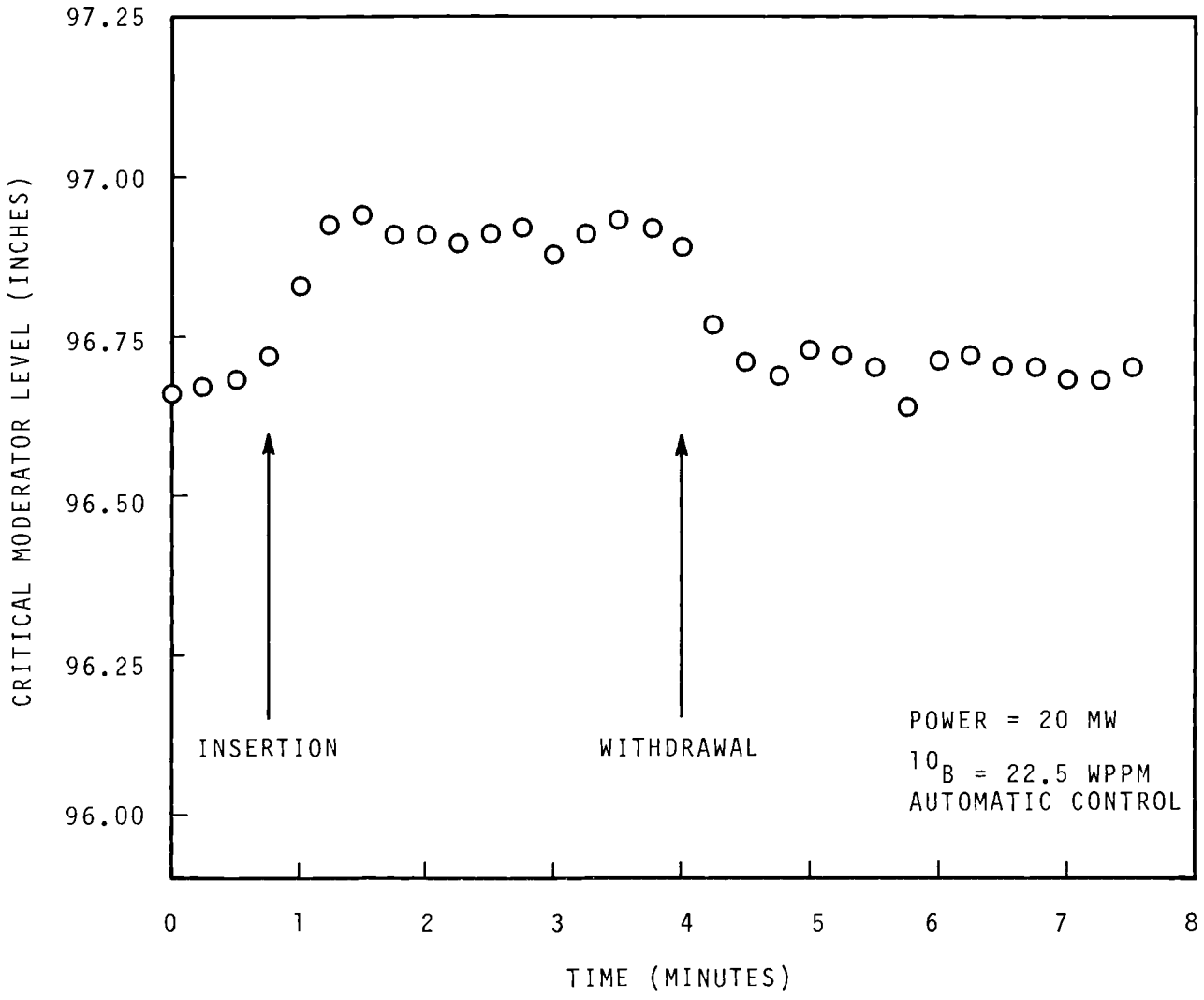

FIGURE 8. EFFECT OF TRAVELING FLUX WIRE ON CRITICAL MODERATOR LEVEL

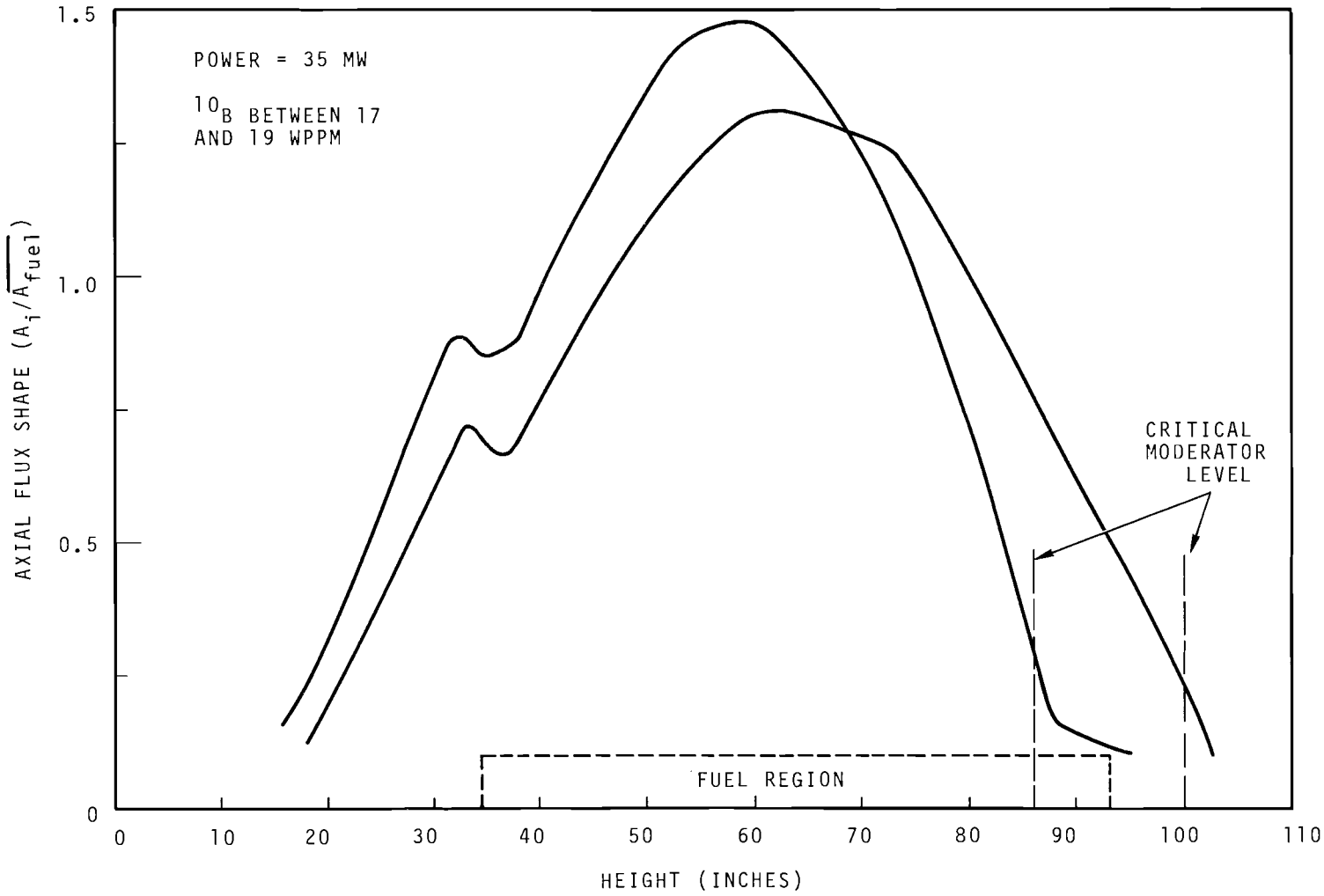

FIGURE 9. AXIAL FLUX SHAPE AS A FUNCTION OF CRITICAL MODERATOR LEVEL

D. Chemical Shim System

The chemical shim system can change the reactivity status of the reactor in three ways: (1) injection of concentrated boric acid solution into the moderator, which causes a reactivity decrease; (2) flow through the boron ion exchanger (BIX) unit, which removes boron from the moderator and causes a reactivity increase; and (3) addition of clean D_2O to the moderator, which dilutes the concentration of boron in the moderator and causes a reactivity increase. The effect of each of these operations has been determined, with approximately 20 wppm ^{10}B in the moderator.

The boron addition system is capable of injecting a concentrated boron solution at a maximum rate of about 100 lb/hour. The normal concentration of the solution (2.5g ^{10}B/liter D_2O) is such that injection of 10 pounds of solution will increase the moderator boron concentration by 1 wppm ^{10}B. Injection of this concentrated solution at the maximum rate would decrease reactivity at the rate of about 1.[illegible] milli-k/minute, at a nominal boron concentration of 20 wppm ^{10}B. Tests were run at injection rate settings of 5 lb/hour, 10 lb/hour, and 20 lb/hour to determine the reactor response, on automatic control, at a power level of 20 MW.

The results of the injection test at 10 lb/hour are presented in Figure 11, showing the response of the automatic controller to the change in reactivity. An oscillatory reaction may be clearly seen. This is due to the recirculation time of the moderator and the fact that complete mixing is not immediate. As it is gradually dispersed throughout the moderator, the injected boron solution circulates as a decreasing "slug" of negative reactivity. The response of the automatic controller which is shown follows this damped oscillation.

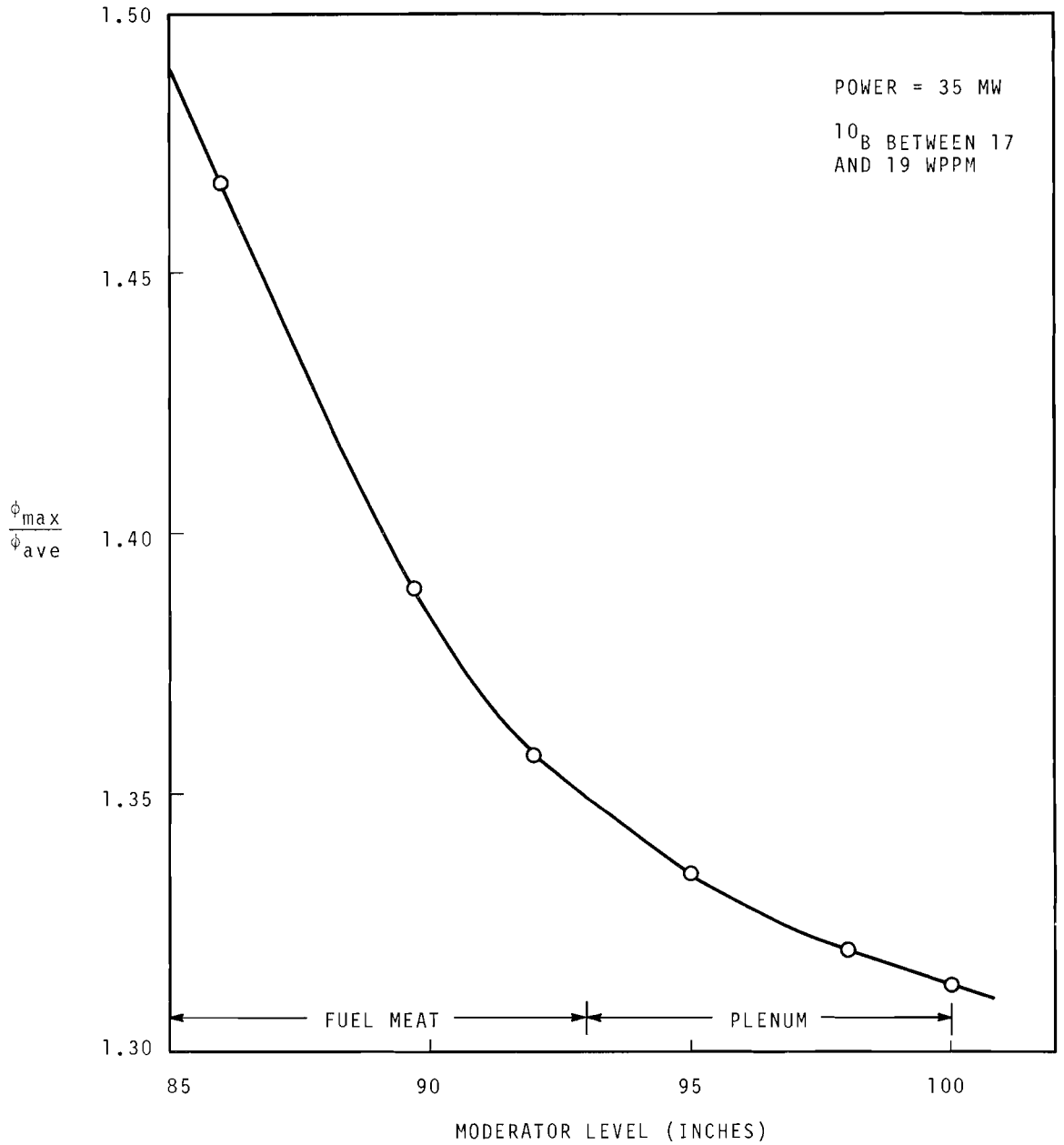

FIGURE 10. AXIAL FLUX PEAKING VERSUS MODERATOR LEVEL

The BIX system, when operating at a full flow (200 lb of moderator/minute). is capable of removing boron from the moderator at a rate equivalent to a reactivity change of about 0.5 milli-k/minute, at a nominal boron concentration of 20 wppm ^{10}B. Limits have been established, however, to restrict this flow to 62 lb/min in order to reduce the rate of reactivity addition to about 0.2 milli-k/minute. Tests were run at BIX flow rate settings 10 lb/min., 20 lb/min., 30 lb/min., and 50 lb/min. to determine the reactor response, on automatic control, at a power level of 20 MW. The results of the test at 20 lb/min. are presented in Figure 12, showing the response of the automatic controller to the change in reactivity.

Again the oscillatory nature can be seen. In this case the moderator passing through the BIX becomes a positive "slug" of reactivity, as it contains no boron when returned to the moderator.

Clean D_2O was added to the moderator for periods of about 4 minutes at rate settings of 10 lb/min., 30 lb/min., and 33 lb/min. to determine the reactor response, on automatic control, at a power level of 20 MW. The results of the test at 30 lb/min. are presented in Figure 13, showing the response of the automatic controller to the change in reactivity. Once again the oscillatory nature of the reactor response can be seen.

These tests were all conducted with approximately 20 wppm ^{10}B in the moderator. The results obtained may not be directly applicable to operation of the chemical shim system at lower boron concentrations because the differential reactivity worth of boron is a function of boron concentrations, becoming larger at lower concentrations. Boron injection will clearly produce a larger effect at lower concentrations. The effect of BIX operation is not so obvious, however, because the boron removal rate will be smaller at lower concentrations while boron worth is larger.

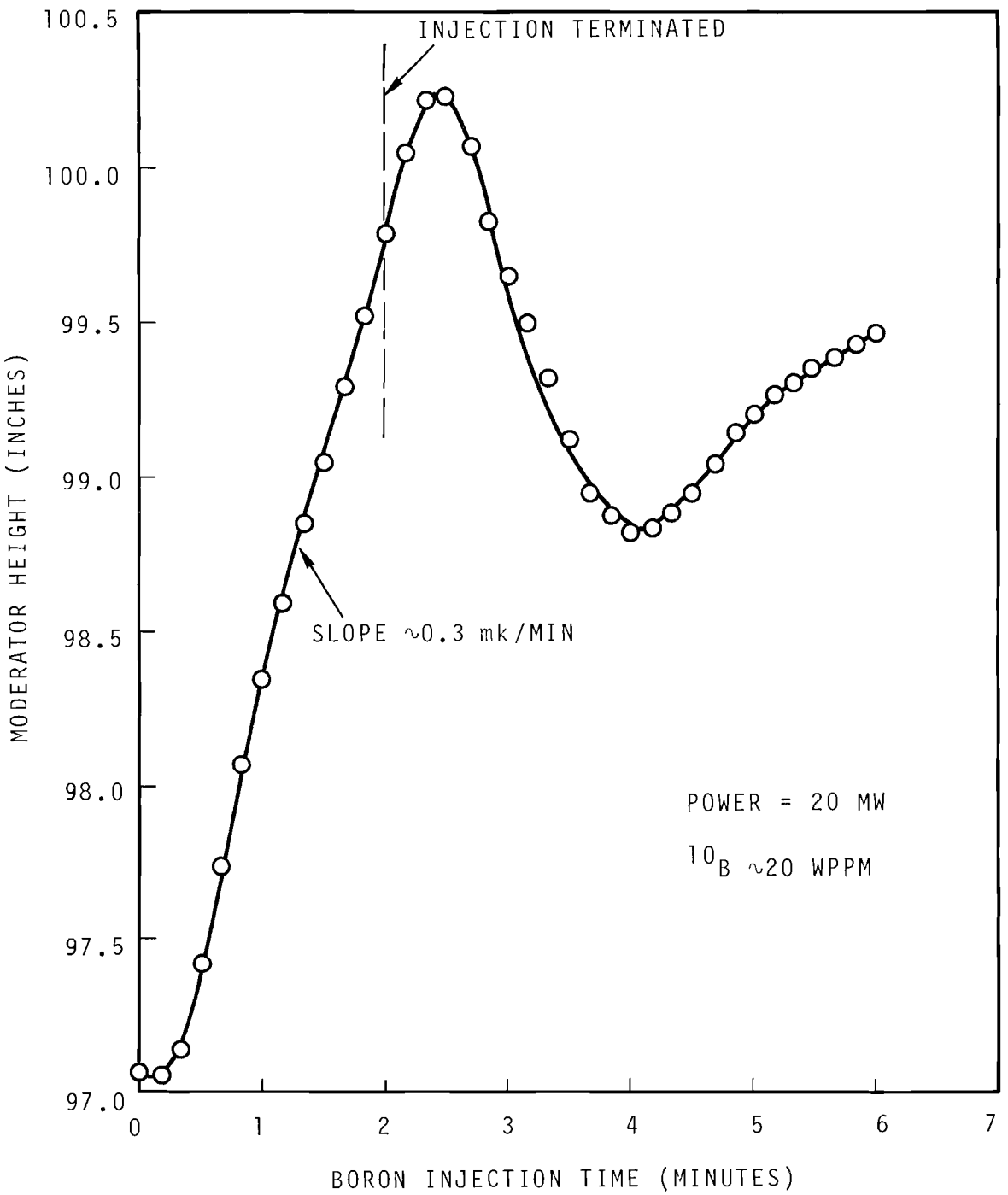

FIGURE 11. MODERATOR LEVEL VERSUS TIME - BORON INJECTION AT 10#/HR

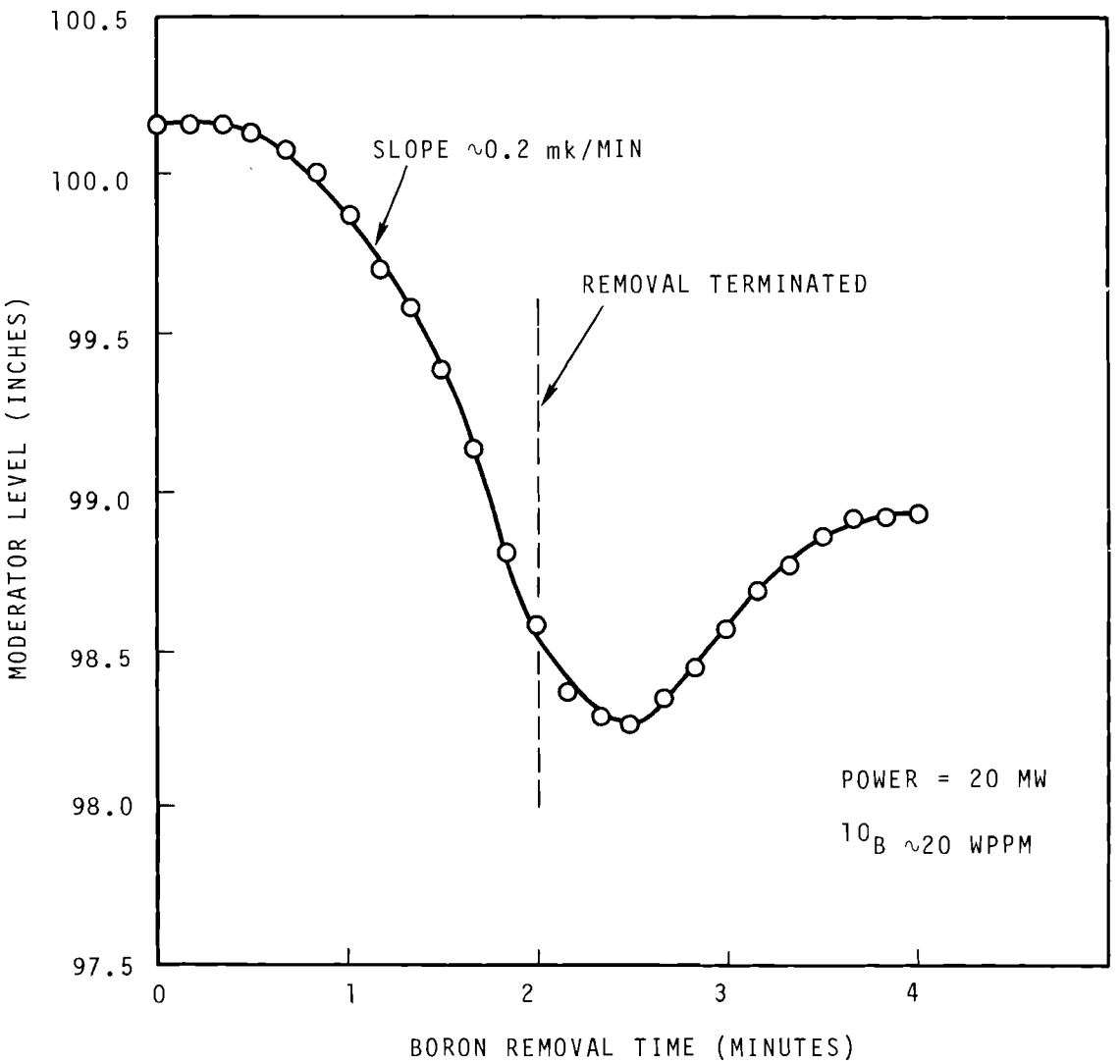

FIGURE 12. MODERATOR LEVEL VERSUS TIME - BORON REMOVAL AT 20#/MIN FLOW THROUGH BIX

E. Determination of Reactor Power

The heat generated in the PRTR is ultimately carried off by the reactor coolant systems: primary, moderator, reflector, shields, and FERTF. A summation of the heat removed by each system should yield an accurate value for total reactor power. This measurement was performed at indicated reactor powers of 9.2, 16.1, 21.0, 26.2, 36.0, 48.0 and 51.5 MW. Data from these measurements are listed in Tables II-VIII.

Power is calculated for each system from the measured flow and change in temperature, i.e.:

$$P = k_6 \cdot F \cdot \Delta T$$

where P = power in MW

F = flow in gal/min

ΔT = (outlet temperature - inlet temperature) in ^{o}F

and

$$k_6 = \frac{(\rho \text{ in lb/ft}^3)(C_p \text{ in BTU/lb } ^{o}F)(60 \text{ min/hr})}{(7.48 \text{ gal/ft}^3)(3412 \text{ BTU/kw hr})(1000 \text{ kw/MW})}$$

$$= (2.351 \times 10^{-6})(\rho)(C_p) \quad \text{MW min/gal } ^{o}F$$

where density ρ and specific heat C_p are both functions of temperature and D_2O/H_2O isotopic purity.[8] The indicated flow is corrected to take into account the fact that the flow measuring equipment is not temperature compensated.

The system labeled "shield loop" is the sum of powers in five loops: top shield, bottom shield, side shield, thermal shield, and rotating shield. The "pump friction" term is a constant correction for the heating of the primary coolant between the inlet and outlet temperature probes due to its friction against the walls of the loop at high flow rates.

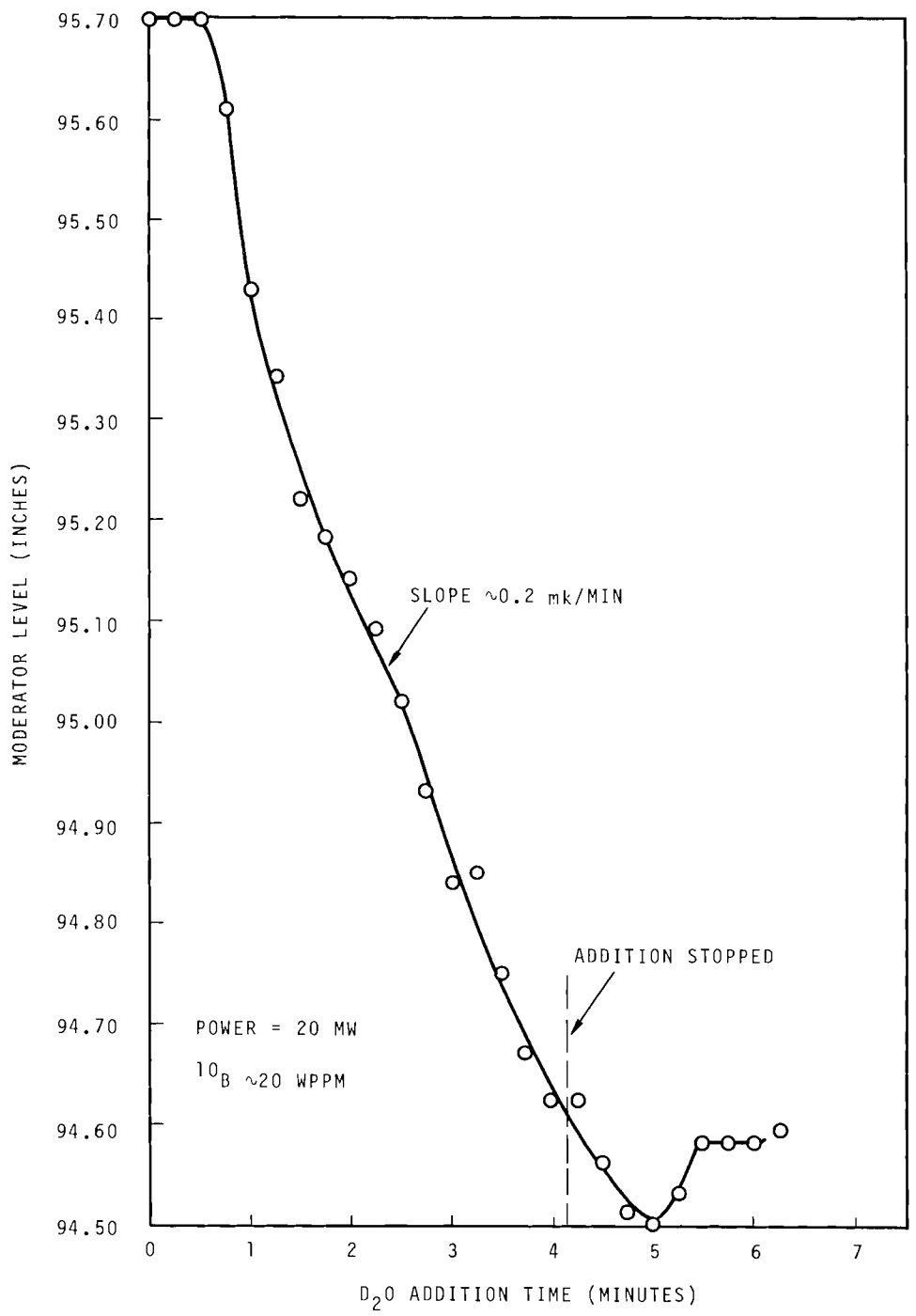

FIGURE 13. MODERATOR LEVEL VERSUS TIME - CLEAN D=O ADDITION AT 30#/MIN

The indicated reactor powers come from the Power Calculator. This system monitors the power level of the reactor as determined by measurement of the heat removed by the primary and moderator reactor coolants. A preset compensation potentiometer is used to account for the other heat removal systems.

Some relatively large discrepancies may be noted, particularly at low powers, between power indicated by the Power Calculator and power determined by summing the heat removed by the various systems. These discrepancies are apparently due to uncertainties in the read-out of temperatures, which become most significant for small values of ΔT, i.e. at low power.

TABLE II

Reactor Power Calculation - 9.2 MW

System	Temperature (^{o}F) Inlet	Outlet	Flow (gpm)	Power (MW)
Primary coolant	355	361	10,200	8.51 ± 1.50
Moderator loop	119	124	1070	0.77 ± 0.20
Reflector loop	78	89	172	0.27 ± 0.02
Shield loop	—	—	—	0.34 ± 0.02
Pump friction	—	—	—	-0.25
FERTF	321	326	164	0.11 ± 0.02
	Total reactor power (from heat removal)			9.75 ± 1.52
	Indicated reactor power (from power calculator)			9.2

TABLE III

Reactor Power Calculation - 16.1 MW

System	Temperature (°F) Inlet	Outlet	Flow (gpm)	Power (MW)
Primary Coolant	443	455	10,500	18.35 ± 1.50
Moderator loop	120	124	1100	0.64 ± 0.20
Reflector loop	84	92	172	0.20 ± 0.02
Shield loop	—	—	—	0.54 ± 0.02
Pump friction	—	—	—	-0.25
FERTF	456	471	167	0.33 ± 0.02
Total reactor power (from heat removal)				19.81 ± 1.52
Indicated reactor power (from power calculator)				16.1

TABLE IV

Reactor Power Calculation - 21.0 MW

System	Temperature (°F) Inlet	Outlet	Flow (gpm)	Power (MW)
Primary Coolant	445	458	10,050	19.19 ± 1.50
Moderator loop	118	125	1100	1.05 ± 0.20
Reflector loop	94	104	173	0.25 ± 0.02
Shield loop	—	—	—	0.68 ± 0.02
Pump friction	—	—	—	-0.25
FERTF	433	452	165	0.44 ± 0.02
Total reactor power (from heat removal)				21.36 ± 1.52
Indicated reactor power (from power calculator)				21.0

TABLE V

Reactor Power Calculation - 26.2 MW

System	Temperature (°F) Inlet	Outlet	Flow (gpm)	Power (MW)
Primary Coolant	449	465	10,300	24.30 ± 1.50
Moderator loop	117	124	1100	1.06 ± 0.20
Reflector loop	87	96	171	0.22 ± 0.02
Shield loop	—	—	—	0.46 ± 0.02
Pump friction	—	—	—	-0.25
FERTF	430	454	166	0.52 ± 0.02
Total reactor power (from heat removal)				26.31 ± 1.52
Indicated reactor power (from power calculator)				26.2

TABLE VI

Reactor Power Calculation - 36.0 MW

System	Temperature (°F) Inlet	Outlet	Flow (gpm)	Power (MW)
Primary Coolant	450	471	10,225	31.48 ± 1.50
Moderator loop	100	110	1100	1.61 ± 0.20
Reflector loop	84	91	171	0.17 ± 0.02
Shield loop	—	—	—	0.44 ± 0.02
Pump friction	—	—	—	-0.25
FERTF	434	465	168	0.69 ± 0.02
Total reactor power (from heat removal)				34.14 ± 1.52
Indicated reactor power (from power calculator)				36.0

TABLE VII

Reactor Power Calculation - 48.0 MW

System	Temperature (°F) Inlet	Outlet	Flow (gpm)	Power (MW)
Primary Coolant	456	486	10,200	44.90 ± 1.50
Moderator loop	123	137	1095	2.27 ± 0.20
Reflector loop	85	100	172	0.38 ± 0.02
Shield loop	—	—	—	0.58 ± 0.02
Pump friction	—	—	—	-0.25
FERTF	441	482	169	0.91 ± 0.02
Total reactor power (from heat removal)				48.79 ± 1.52
Indicated reactor power (from power calculator)				48.0

TABLE VIII

Reactor Power Calculation - 51.5 MW

System	Temperature (°F) Inlet	Outlet	Flow (gpm)	Power (MW)
Primary Coolant	455	490	9980	51.74 ± 1.50
Moderator loop	106	121	1090	2.41 ± 0.20
Reflector loop	115	123	170	0.20 ± 0.02
Shield loop	—	—	—	0.48 ± 0.02
Pump friction	—	—	—	-0.25
FERTF	453	495	140	0.80 ± 0.02
Total reactor power (from heat removal)				55.38 ± 1.52
Indicated reactor power (from power calculator)				51.5

A significant result of these measurements is the determination of the percentage of total reactor power carried off by each of the coolant systems. The results indicate that 93.5 ± 1.2% of the total reactor power is captured in the primary coolant system, while 4.8 ± 1.0% is captured in the moderator system, and 1.7 ± 0.5% in the reflector, shields, and FERTF. These percentages will vary slightly depending upon the fuel loading in the FERTF at any given time.

F. Temperature Coefficients of Reactivity

Temperature coefficients of reactivity were measured for the fuel-plus-primary-coolant system and for the moderator system. As much as possible the temperature of one system was held constant while the temperature of the other system was varied. The measurements were made with 22.5 wppm ^{10}B in the moderator.

The temperature coefficient of reactivity of the primary coolant system was measured over a period of approximately one hour, during which time reactor power was increased from 1.5 MW to 6.0 MW. The average temperature of the primary coolant was increased from 178°F to 356°F. During this time the moderator temperature was held constant by controlling flow through the heat exchanger and the reflector temperature increased by 2°F. The temperature coefficient of reactivity for the primary coolant system, over the range from 178°F to 356°F, was determined to be -0.025 ± 0.001 milli-k/^{o}F. The temperature given for the primary system is the bulk temperature and there are, of course, variations from tube to tube, depending on the power profile. These variations reflect temperature variations in the fuel. For the low powers involved in this test, however, the average ΔT is about 12°F, so that tube-to-tube variations in average temperature are of the order of

2° or 3°. This is not significant for the 178° range used in this measurement.

The temperature coefficient of the moderator system was measured over a period of approximately two hours, with reactor power held at 6.0 MW. During this time the moderator temperature was increased from $92.5^{o}F$ to $141^{o}F$. The reflector temperature increased by $17^{o}F$, while the primary coolant temperature remained constant due to the constant power. Reactivity changes were determined by measurement of critical moderator height. When the moderator temperature was varied it was necessary to correct the manometer reading of height for density changes in the moderator. This correction was 0.8 in. in the most extreme case measured. The temperature coefficient of reactivity for the moderator, containing 22.5 wppm ^{10}B, over the range from $92.5^{o}F$ to $141^{o}F$, was determined to be $+0.101 \pm 0.003$ milli-k/^{o}F.

G. Kinetics Parameters

The value of β/ℓ, the ratio of the effective delayed neutron fraction to the neutron lifetime, was measured 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 could be adjusted from 0.1 to 15 cycles per second.

The measurement was performed with the reactor at a power of 1 MW. The core contained 77 fuel elements and the moderator contained 22.5 wppm ^{10}B. The transfer function amplitude response is shown in Figure 14. The points

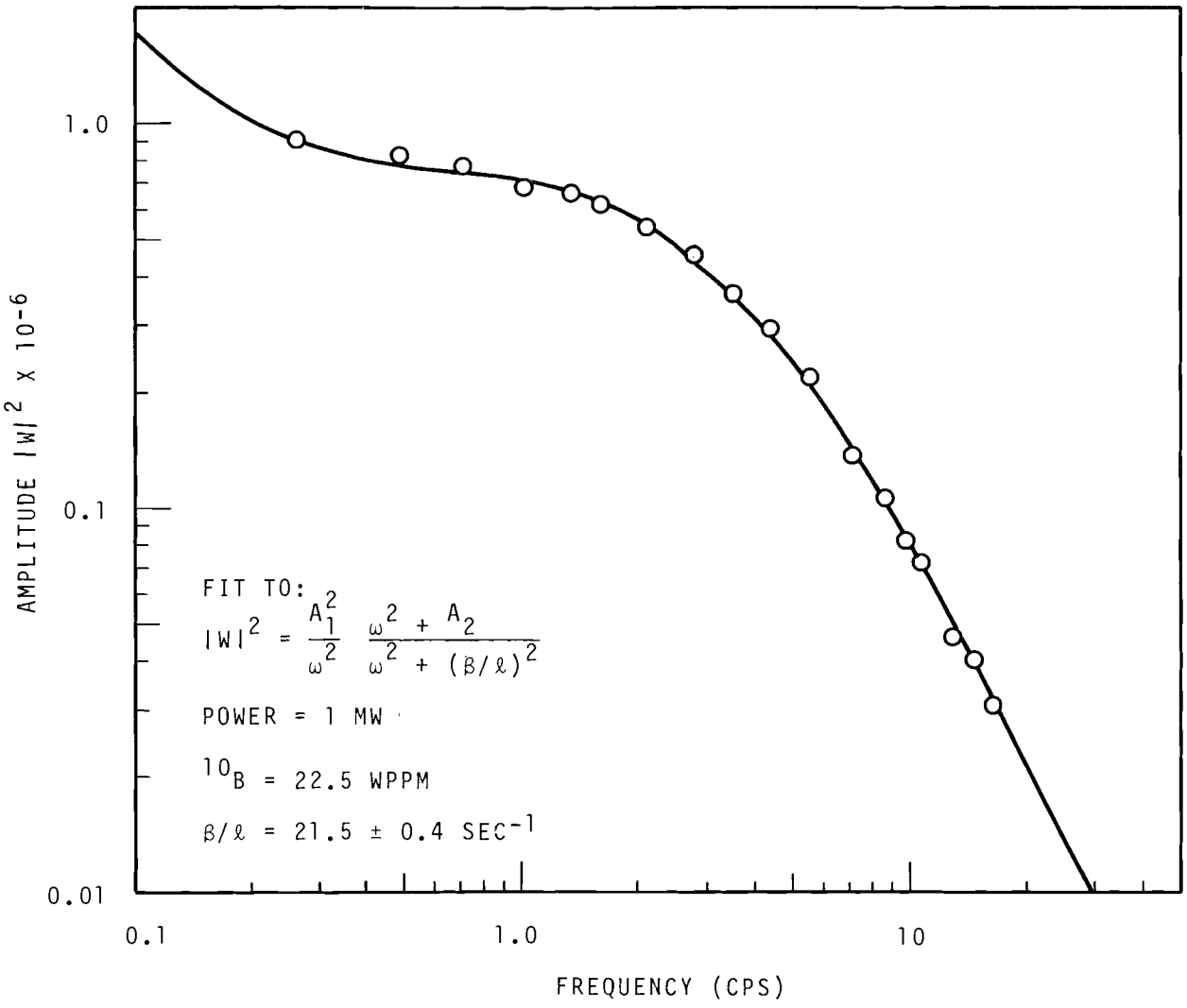

FIGURE 14. PRTR TRANSFER FUNCTION AMPLITUDE RESPONSE

represent the experimental data, while the curve is a fit to the formula

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

where

A_1, A_2 = constants

ω = frequency in radians/second

β = delayed neutron fraction

ℓ = prompt neutron lifetime in seconds

using the computer code LEARN[6,7]. From the fitting, the best value of β/ℓ was determined to be $(21.5 \pm 0.4)\ \mathrm{sec}^{-1}$. Assuming the value 0.00319 for β[9], the prompt neutron lifetime, ℓ, is (0.15 ± 0.01) milliseconds.

H. Growth of Saturable Fission Product Poisons

The growth of saturable fission product poisons, primarily xenon, was measured during the cold reactor startup. The core had a total burnup of 1830 MWd (972 MWd/MTM average) at the time this test was performed, but the reactor had been shut down for one week so that the short-lived fission product poisons from previous operation had decayed to an insignificant level.

Operating considerations precluded raising the power directly from zero to its equilibrium value of 47 MW. Therefore, the power level was increased in a series of steps, as shown in Figure 15.

The points in Figure 15 represent the experimentally measured reactivity values. These were determined by measuring the critical moderator height and boron concentration during equilibrium reactor operation. The base

point was taken from the initial cold approach-to-critical. The reactivity effects of temperature changes in coolant, moderator, and reflector-discussed earlier-were taken into account in these determinations. The curve of fission product poison shown in Figure 15 was calculated for the indicated power history using the computer code XRST[10,11]. The measured and calculated values are seen to agree satisfactorily.

I. Growth and Decay of Fission Product Poisons Following a Shutdown

The reactivity transient due to the growth and decay of xenon was measured after shutdown from a 125 hour period of operation at an equilibrium power level of 51 MW. The reactor was shutdown in the normal manner, over a half-hour period. During the four hours required for the primary system to be cooled and depressurized, no reactivity measurements could be made. Hence the first measurements were made five hours after shutdown.

The type of approach-to-critical used in making these measurements involved holding the moderator level and boron concentration constant and plotting inverse multiplication versus time. This enabled the determination of the time at which a given configuration would be critical. From this, the reactivity worth of the poison could be determined as a function of time.

The results of these measurements appear in Figure 16. The curve shown for comparison was calculated using XRST. The base point of 37 milli-k represents the equilibrium fission product poisoning at a power of 51 MW for the core with a total burnup of 2400 MWd (1275 MWd/MTM average) at the time of the test. The measured and calculated values are seen to agree quite well, except when the poison is increasing. The reason for the discrepancy in this case is not known.

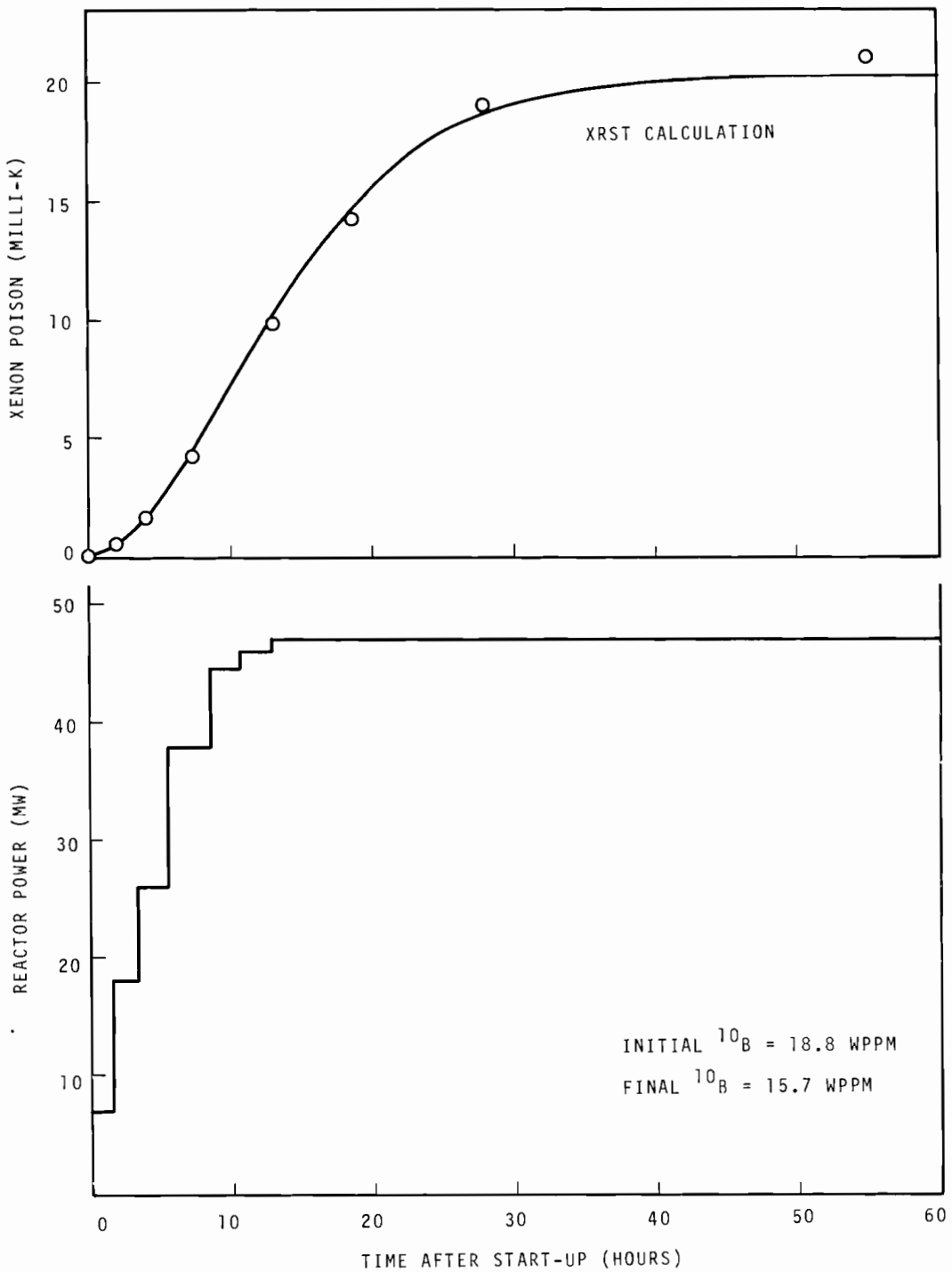

FIGURE 15. XENON BUILD-UP IN PRTR

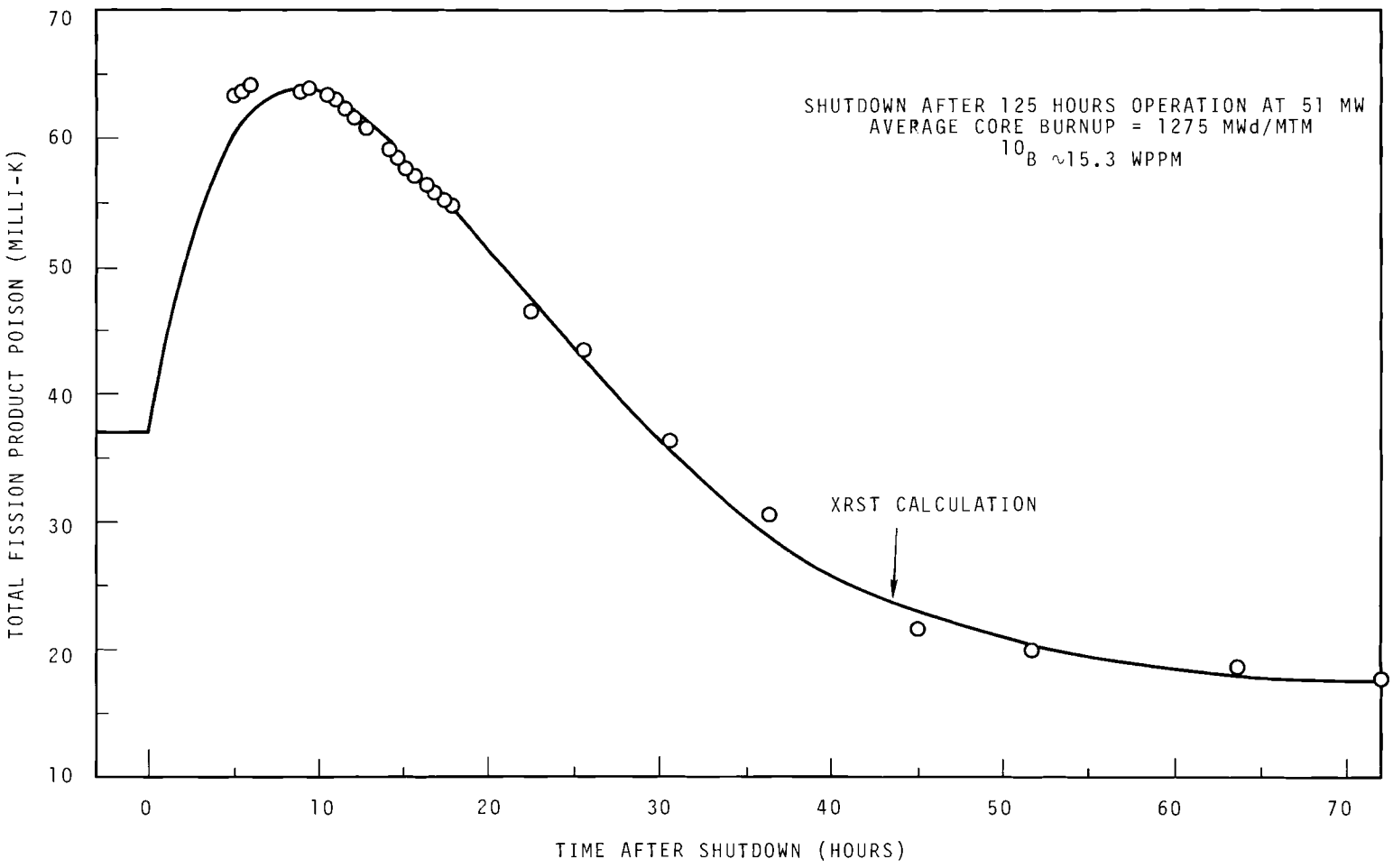

FIGURE 16. SHUTDOWN XENON TRANSIENT IN PRTR

IV. CONCLUSIONS

The results of the Batch Core Power Tests have, in general, confirmed the predicted operational behavior of the PRTR. The ability of the reactor control systems to control reactivity changes induced by the chemical shim system and the traveling flux wire has been demonstrated.

Useful data which permit more efficient operation of the reactor have been obtained on the operation of flux chambers, the traveling flux wire system, the chemical shim system, and the power calculator.

Physics data, complementing those obtained in the Batch Core Critical Tests[2], have been obtained. These data include temperature coefficients of reactivity, kinetics parameters, and growth and decay of fission product poisoning. Some of these quantities have also been measured at selected intervals of burnup during the Batch Core Experiment and will be presented in later reports.

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