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SUMMARY OF REACTOR DESIGN INFORMATION FROM THREE YEARS' OPERATION OF A SMALL PWR

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SUMMARY OF REACTOR DESIGN INFORMATION FROM THREE YEARS' OPERATION OF A SMALL PWR

By:
J. G. Gallagher, Head, Nuclear Technology Unit

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SUMMARY

Reactor design information obtained from three years' operation of a small pressurized water reactor, the SM-1 (formerly APPR-1), is presented and discussed. The SM-1 reactor, designed to produce 10 tMW power, employs fully enriched uranium fuel in the form of UO_2 dispersed in stainless steel fuel plates. The reactor is cooled by water at 1200 psia and mean temperature of 440°F. The first core for this reactor was installed in April, 1957, and operated at approximately 50% load factor until May, 1960. During this period the reactor was utilized for crew training power operation, and as a source of reactor design information. Experimental measurements were performed of nuclear, thermal, kinetics and shielding characteristics and the majority of these were compared with analytical calculations.

Core physics measurements were performed of temperature coefficient, pressure coefficient, rod calibration, stuck rod position and transient xenon as a function of core burnout. Core burnout characteristics were compared with few group calculations and reasonable agreement obtained.

Thermal heat balance data was obtained on the reactor core. The temperature pattern in the nominal and hot channels under operating conditions was calculated. These calculations indicated certain of the fuel channels operated in the nucleate boiling regime. Examination of one of the fuel channels suspected of nucleate boiling indicated no adverse effects.

The system response to load perturbations and during pump coastdown was measured utilizing plant instrumentation. This response was compared with analytical predictions using a lumped kinetic model and reasonable agreement found.

Both neutron and gamma traverses were made through the primary shield during reactor operation. Gamma traverses were also made through the primary shield as a function of time after reactor shutdown. Conventional shielding calculational methods are found to give agreement with experiment sufficient for design purposes. An absolute ionization chamber was employed to measure N-16 activity in the reactor coolant. These measurements were compared with N-16 calculated from the n, p reaction on O-16.

A large portion of the information presented in this paper is applicable to small low enriched PWR's.

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SUMMARY OF REACTOR DESIGN INFORMATION FROM THREE YEARS' OPERATION OF A SMALL PWR

J. G. Gallagher

1. Introduction

In the course of three years' operation of a small PWR, the SM-1,(1) design information has been developed which should be of value to nuclear engineers designing small pressurized water power reactors. This paper presents a summary of this design information in the areas of core physics, core heat transfer, core and system kinetics, and shielding.

2. Reactor Characteristics

In order to facilitate presentation of design information on this small PWR a brief description of the reactor will be given.

The SM-1 is a pressurized water reactor which generates about 1900 eKW (net) of electricity from 10 tMW.(2) The reactor employs fully enriched uranium as fuel in the form of UO_2 dispersed in stainless steel fuel plates.(3) The reactor is cooled by approximately 4000 gpm of water at 1200 psia and a mean temperature of 440°F. The pressurized water is used to generate steam at 200 psia and 20°F superheat in a steam generator. The primary coolant is circulated at about 4000 gpm by a canned motor pump. Figure 1 shows the general arrangement of the primary and secondary system. Figure 2 shows the reactor and primary shield configuration.

The reactor core is composed of 38 stationary elements and 7 control rod assemblies. Figure 3 shows a fixed element and Figure 4 shows a control rod assembly which is composed of control rod tube, fuel element, absorber and cap. The stationary and control rod assemblies are arranged as shown in Figure 5 to form the reactor core.

3. Core Physics Data

The dimensions and material content of SM-1 Core I are presented in Table 1. As shown in Figure 2 the core is surrounded by an essentially infinite water reflector. The control rods are adjusted so that rods 1, 2, 3, 4 and C (see Figure 5) are positioned as a bank while control rods A and B are essentially fully withdrawn. The control rods are actuated by motion of individual switches. Neutrons are provided for initial startup by a 15 curie PoBe source. After initial operation, startup neutrons are provided by (γ,n) reactions on a beryllium block attached to the core skirt. This block is 3" x 3" x 0.5".

Initial Criticality, 68°F

An assembly of ten fixed elements and seven control rod elements constitute the initial critical loading.(4) This loading corresponds to a loading of U^{235} and B^{10} of 8.08 kg and 5.65 gm respectively.

Initial Full Core Criticality, 68°F

The full 45 element core is critical with the five rods 1, 2, 3, 4 and C withdrawn 3.7".(2) Rods A and B are fully withdrawn.

The five rod bank has been roughly calibrated over the position 3.7" to 22" withdrawn by the addition of boron steel strips to the water channels in each element. The integral of the rod worth curve from 3.7" to fully withdrawn is 22.8$.(4)

Using an effective delayed neutron fraction of 0.0073 and the equation (5):

$$\rho = 1-e^{-0.0073\ K_{ex}}$$

where K_{ex} is the integral of the rod worth in $

The calculated reactivity is 15.4% and core eigenvalue or K_{eff} is 1.18. It should be noted that reactivity measurements of this magnitude are difficult to make and are subject to different interpretations.

Initial Full Core Criticality, 440°F

The five rod bank was withdrawn to 6.7" when the mean coolant temperature was slowly increased to 440°F.(2) Using the room temperature calibration curves for the bank worth the core reactivity is 15.4$. This corresponds to a reactivity of 10.6% or a K_{eff} of 1.12. The temperature coefficient at startup and 440°F is $-2.2 \times 10^{-4}\ {}^{o}F^{-1}$.(2)

Core Criticality as a Function of Core Condition and Energy Extracted

The five rod bank position as a function of core temperature, xenon concentration and energy extracted is shown in Figure 6.(6,7) The core bank insertion is a maximum at startup. The variation with core life reflects the rather small B^{10} content of the core. The measured energy release of the core is 16.4 MWYR. Using 2 MWYR equivalent to 1 Kg U-235 destroyed, the fuel content at end of life is 14.3 Kg.

Comparison of Calculation and Measurements

As an indication of the usefulness of two-group theory, a comparison of calculated (8) and measured eigenvalues for initial criticality and initial full core criticality at 68° and 440°F is presented in Table 2. The calculations were based on

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two-dimensional diffusion theory for a plane thru the core as shown in Figure 5. In calculating the thermal group parameters P_3 theory was used to obtain intercell distributions of thermal neutrons.(9) The fast group parameters were based on Goertzel-Selengut theory.(10) The calculated eigenvalues are low for all three configurations. The agreement is entirely adequate for engineering purposes.

A comparison of the calculated (11) and measured bank position as a function of core energy release is shown in Figure 7. The calculation is based on the following: (1) a one-shot radial burnout of the core to obtain an estimate of the effect of non-uniform radial burnout on K_{eff}, and (2) a one-shot axial burnout of the core in which the control rod bank insertion is represented as a uniform poison. The calculation is corrected to agree with initial measured bank position.

Conclusions

Physics measurements have established all the major characteristics of SM-1 Core I, including bank position under all conditions. A comparison of modified two-group diffusion theory calculations with initial criticality and core burnout indicates that it can be used with confidence to predict core burnout characteristics for cores of this type.

4. Core Thermal Data

The core thermal characteristics are governed by the following:

a. Fuel element geometry

b. Power distribution

c. Hot channel factors

d. Coolant flow rate and distribution

The fuel element and core geometry is shown in Figures 3, 4 and 5. Figures 8 and 9 illustrate the calculated radial (8) and axial (12) power generation at startup, 440°F. Table 3 summarizes all the hot channel factors used in the thermal analysis.(13) Table 4 gives the flow distribution through the core. The element positions within the core are given in Figure 5. The distribution was established by the use of a full size air flow model.(14) Due to a lack of definition of the power peak at the core reflector interface, at the time the flow orifices were sized, the flow is overtailored in positions adjacent to the core reflection interface.

In-core instrumentation is not now available in SM-1. The detail thermal characteristics can only be calculated from overall heat balance data.(2) Measurements indicate that the core thermal output can be 10.8 MW at a coolant flow of 3960 gpm and inlet temperature of 428°F. The calculations made for conditions close to the heat balance point indicate that elements in positions 47, 57 and 67 (and symmetrical elements) next to the core reflector interface operated with some nucleate boiling. Figures 10 and 11 present the calculated surface temperature (15) for the hot and nominal channel, respectively, in positions 47 and 67. The nominal channel is that channel in which none of the deviations numbers 1-6 in Table 3 exist. The flat portion of the calculated surface temperature indicates regions of local boiling.

Post-irradiation examination of the element from position 57 after 2/3 core life indicates no adverse effects from this operation.(16) Periscopic examination of elements from position 70 and 71 at end of core life indicated no adverse effects.

Conclusions

The effect of calculated limited nucleate boiling in certain hot channels of the SM-1 was negligible on overall core performance and on element metallurgy.

5. Kinetic Data

The kinetics of a pressurized water reactor can be divided in several areas as follows:

a. Pump coastdown.

b. Decay heat removal

c. Load perturbation

d. Xenon transients

Measurements to date at the SM-1 have been handicapped by the lack of in-core instrumentation and use of conventional power plant instrumentation.

Pump Coastdown

The pump coastdown period begins at the failure of the primary pump and continues until the natural circulation flow begins. This period is of particular interest in a reactor, such as the SM-1, which is operating on temperature coefficient control only. Following a loss of primary pump the reactor is normally scrammed when the flow has dropped to 90% of nominal. On a series of special tests at SM-1 the reactor was not scrammed for 15 sec following cut off of the primary pump power.(17) Figure 12 shows the result of a test at an initial power of 10 MW. Also shown is the calculated reactor power using a lumped thermal model of the reactor core and the usual reactor kinetic equations.(18) A measured fluid coastdown curve was used in the calculation.

Decay Heat Removal

Following the loss of the primary pump power and scramming of the reactor on a low flow signal the primary fluid is circulated by natural convection. Due to the primary system arrangement shown in Figure 1, the primary fluid continues to circulate through the primary system. After the reactor scram and decay of fission power, the heat release in the core is determined by fission product decay. Figure 13 shows the measured (2) and calculated reactor outlet temperature during the decay heat removal period. The calculated curve is based on a lumped thermal model of the reactor core and steam generator.

Load Perturbation

The response of the SM-1 to load perturbation has been graphically demonstrated on many occasions. The SM-1 is controlled by temperature coefficient only during these tests. In one test the generator load was increased from 200 to 2000 KW gross in 60 sec. The reactor thermal power responded to this load demand on temperature coefficient only. Figure 14 shows the measured (2) and the calculated (19) reactor response. The calculated reactor response is based on a lumped

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reactor and steam generator model.

Figure 15 shows the measured (2) variation in reactor power following a sudden reduction in generator load caused by opening a circuit breaker. Also shown in Figure 15 is the calculated (19) reactor response.

Xenon Transient

Following a change in reactor power there is a change in Xe^{135} concentration which causes a change in core reactivity. In the SM-1 the control rods are adjusted manually to keep the mean temperature at 440°F ± 1°F. A comparison of calculated and measured control rod bank positions indicated a considerable discrepancy when a uniform xenon distribution was assumed. It was found (19) that in order to get agreement between experiment and calculated bank positions, a distribution factor, α, must be employed. A constant value of α equal to 1.3 was used in the buildup of equilibrium xenon and the following α after a reduction in power

$$\alpha = 1.3 + 0.3 \left(\frac{t}{12}\right) \qquad 0 \leq t \leq 12$$

where t is in hours

$$\alpha = 1.6 \qquad t > 12$$

Figure 16 shows the measured bank movement following an increase in power from 0 to 7.7 MW and following a reduction in power from 7.7 to 2.2 MW. Also shown on Figure 16 is the calculated bank movement with the distribution factor given above.

Conclusions

A comparison of measurements and analysis of the reactor and system kinetics indicated that a lumped model can be used to satisfactorily predict the kinetic behavior during pump coastdown, decay heat removal and load perturbations of this type of small pressurized water reactors. Analysis of transient xenon bank measurements indicate that a distribution factor must be included to account for Xe^{135} distribution in order to obtain close agreement between measurements and calculation.

6. Shielding Data

Primary Shielding

The primary shield for SM-1 is shown in Figure 2 and the dimensions are given in Table 5. The neutron and gamma distributions in the primary shield have been measured both during and after operation.

Figure 17 shows the measured and calculated neutron distribution in the primary shield.(20) The calculated neutron distribution is based on multiregion two-group diffusion theory.

Figure 18 shows the gamma flux distribution through the primary shield as measured in a shield mockup at a critical facility (21) and at the SM-1 together with a calculated curve. Figure 19 shows the gamma flux as a function of time after shutdown between various rings in the primary shield tank.

Primary Coolant Activation

An absolute ionization chamber (22) was constructed and used to measure the N^{16} activity in the outlet pipe of the reactor. The chamber was a graphite-walled ionization chamber. The measured ionization current was used to calculate the volume source in the primary system pipe. This value was corrected back to reactor outlet. The resulting specific activity in disintegrations per cc sec is 1.6×10^6. Using an $O^{16}(n,p)N^{16}$

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cross section of 16.3 mb averaged over a fission spectrum above 10 Mev, the calculated specific activity is 1.3×10^6 disintegrations per cc sec.

Conclusions

Neutron diffusion theory and standard shielding calculation methods (20)(23) give adequate predictions of neutron and gamma distribution thru a primary shield. A value of 16 mb for the $O^{16}(n,p)N^{16}$ cross-section gives reasonable agreement between calculated and measured N^{16} disintegrations per cc sec at the reactor vessel outlet.

TABLE 1

DIMENSIONS AND MATERIAL CONTENT OF CORE

Dimensions of Core		
Active length	in	21.75
Equivalent diameter	in	22.2
Cell size	in	2.9375 x 2.9375
Number of Cells		45
Configuration		7 x 7 with corners missing
Fixed Element		
Number		38
Number of plates		18
Plate dimension	in	0.030 x 2.778 x 23
Weight of U^{235}	gm	515.16
Weight of UO_2	gm	630.36
Weight of B^{10}	gm	0.3605
Weight of B_4C	gm	2.626
Control Rod Element		
Number		7
Number of plates		16
Plate dimensions		0.030 x 2.56 x 23
Weight of U^{235}	gm	417.76
Weight of UO_2	gm	512.16
Weight of B^{10}	gm	0.2926
Weight of B_4C	gm	2.132
Control Rod Absorber		
Number of plates (box)		4
Weight of B^{10}	gm	56.4
Material Content of Core		
Weight of U^{235}	Kg	22.50
Weight of B^{10}	gm	15.75
Weight of SS	Kg	208.9
Weight of Water (68°F)	Kg	111.08

TABLE 2

CALCULATED AND MEASURED EIGENVALUES

Condition	Measured	Calculation
Initial criticality	1.00	0.99
Initial full core criticality, 68°F	1.18	1.15
Initial full core criticality, 440°F	1.12	1.10

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TABLE 3

HOT CHANNEL FACTORS

Item	Deviation	$F_{Average}$	F_{Local}
1. Plate Spacing	±.003" average ±.007" local	1.023	1.056
2. Uranium Content	±.5% average ±1%	1.005	1.010
3. Clad Thickness	±.0002" average ±.0004" local	1.003	1.005
4. Inlet Box			
Stationary	±3% average local	1.031	1.035
Control Rod	±6% average local	1.064	1.051
5. Orifice Size	±4% average local	1.042	1.033
6. Meat Thickness	±.0008" average ±.0013" local	1.040	1.065
7. Overall Factor			
Stationary Element		1.151	1.209
Control Element		1.188	1.239
8. System Pressure	±20 psi		
9. Inlet Temperature	±3°F		
10. Reactor Power	+3.5%		

TABLE 4

FLOW DISTRIBUTION THROUGH CORE

Position	No. of Elements	Available Flow (gpm)	Total Flow (gpm)
44	7	57	401
34	4	106	424
35	2	101	202
25	8	91	728
26	4	69	276
16	8	45	360
15	8	56	448
14	4	59	236
Lattice Flow			883
Total Flow			3960

TABLE 5

DESCRIPTION OF RADIAL REACTOR SHIELD

Description	Material	Outer Radius In.	Thickness In.
Core	---------	11.1	---
Reflector	Primary Water	17.5	6.4
Thermal Shield	Stainless Steel(1)	19.5	2.0
Inlet Passage	Primary Water	23.75	4.25
Pressure Vessel	Steel	26.5	2.75
Insulation	Glass Wool(2)	30.125	3.625
Insulation Cladding	Steel	30.375	0.25
Clearance Space	Void	32.25	1.875
Vessel Support and Shield Tank Wall	Steel	34.25	2.0
1st Cooling Passage	Shield Water	35.25	1.0
1st Shield Ring	Steel	37.25	2.0
2nd Cooling Passage	Shield Water	38.5	1.25
2nd Shield Ring	Steel	40.5	2.0
3rd Cooling Passage	Shield Water	41.5	1.0
3rd Shield Ring	Steel	43.5	2.0
4th Cooling Passage	Shield Water	44.5	1.0
4th Shield Ring	Steel	46.5	2.0
5th Cooling Passage	Shield Water	47.5	1.0
5th Shield Ring	Steel	49.5	2.0
6th Cooling Passage	Shield Water	50.5	1.0
6th Shield Ring	Steel	52.5	2.0
7th Cooling Passage	Shield Water	53.5	1.0
7th Shield Ring	Steel	55.5	2.0
Neutron Shield	Shield Water	80.0	24.5
Shield Tank Outer Wall	Steel	80.5	0.5

(1) Considered as steel for shielding purposes.

(2) Considered as void for shielding purposes.

(13) LOVE, J. D., Thermal Design Criteria for APPR Pressurized Water Reactor, Alco Products, Inc., APAE Memo 157, November 1958.

(14) INGENERI, S. M., Coolant Flow Tailoring Program of the APPR-1 Core Employing Full Scale Model of the Reactor Vessel, Alco Products, Inc., APAE Memo 108, November 1957.

(15) BERETSKY, I., and PACINO, S., ABWAC-MNC, A Boiling Water Analysis Code on the IBM-650, Alco Products, Inc., APAE Memo No. 181, March 1959.

(16) Army Package Power Reactor Project Annual Progress Report, Oak Ridge National Laboratory, ORNL 2907, April 1960.

(17) MORRISON, J. H., and LEE, D. H., Evaluation of Loss of Flow Accident in the SM-1, Final Report Test 600, Alco Products, Inc., April 1960.

(18) SCHULTZ, M. A., Control of Nuclear Reactors and Power Plants, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, McGraw-Hill Book Co., Inc., (1955).

(19) BRONDEL, J. O., and TOMONTO, J. R., Plant Transient Analysis of the APPR-1 by Analog Computer Methods, Alco Products, Inc., APAE No. 38, October 1958.

(20) ROSEN, S. S., APPR-1 Research and Development Program Shielding Experiments and Analyses, Task No. VI, Alco Products, Inc., APAE No. 35, October 1958.

(21) NOAKS, J. W., et al., Extended Zero Power Experiments on the Army Package Power Reactor, ZPE-2, Alco Products, Inc., APAE No. 21, November 1957.

(22) DARDEN and SHEPPARD, A Thimble Type Gamma Ray Dosimeter, Oak Ridge National Laboratory, ORNL 1028.

(23) ROCKWELL III, Reactor Shielding Design Manual, McGraw-Hill Book Co., Inc., Cambridge, Mass. 1953.

BIBLIOGRAPHY

(1) Phase III, Design Analysis for the Army Package Power Reactor, Alco Products, Inc., APAE Report No. 10, August 1957.

(2) MEEM, J. L., Initial Operation and Testing of the Army Package Power Reactor, APPR-1, Alco Products, Inc., APAE No. 18, August 1957.

(3) BEAVER, R. J., et al., Specifications for Army Package Power Reactor (APPR-1), Fuel and Control Rod Components, Oak Ridge National Laboratory, ORNL 2225, August 1957.

(4) NOAKS, J. W., and JOHNSON, W. R., Army Package Power Reactor Zero Power Experiments, ZPE No. 1, Alco Products, Inc., APAE No. 8, February 1957.

(5) GALLAGHER, J. G., FAIRBANKS, F. B., APPR-1 Control Rod Experiments and Calculations, Alco Products, Inc., APAE Memo No. 92, March 1957.

(6) MACKAY, S. D., SM-1 Formerly (APPR-1) Research and Development Program (Interim Report On) Core Measurements, Task No. VII, Alco Products, Inc., APAE Memo 178, March 1959.

(7) MACKAY, S. D., SM-1 Research & Development Program Interim Report No. 2 on Core Measurements Task No. VII, Alco Products, Inc., APAE Memo No. 206, June 1959.

(8) PALUSZKIEWICZ, S. To be published.

(9) BYRNE, B. J., and OBY, P. V., Analysis of Extended Zero Power Experiments on the Army Package Power Reactor, ZPE No. 2., Alco Products, Inc., APAE No. 27, May 1958.

(10) HELLENS, R. L., Multi-group Fourier Transform Calculation Description of the MUFT III Code, Westinghouse Atomic Products Division, WAPD-TM-4, July 1956.

(11) WILLIAMSON, T. G., Reactor Analysis APPR-1 Core II, Alco Products, Inc., APAE No. 32, July 1958.

(12) BYRNE, B. J., Power Distribution for APPR-Type Cores, Alco Products, Inc., APAE Memo No. 195, July 1959.

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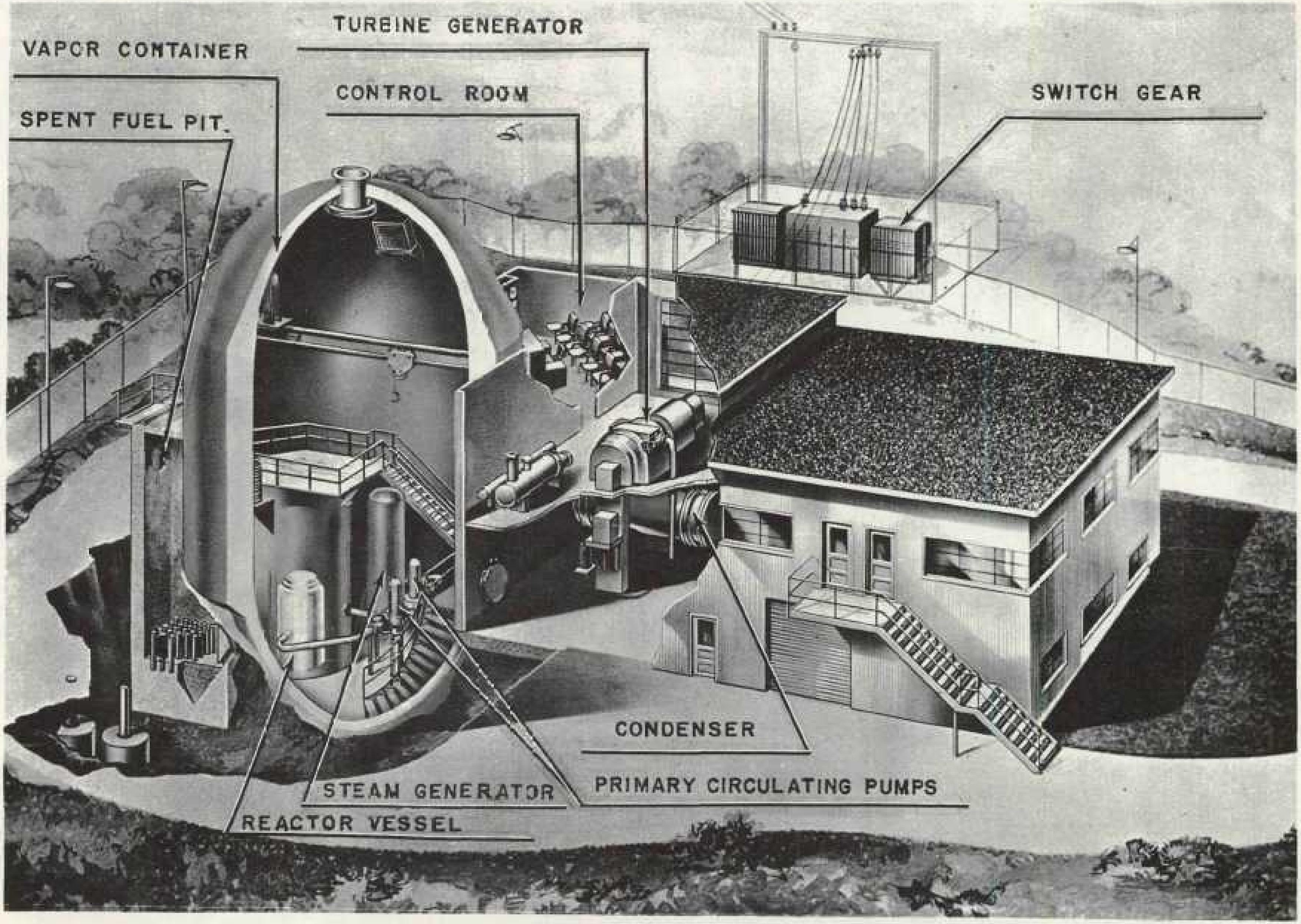


FIG. I GENERAL ARRANGEMENT OF THE PRIMARY AND SECONDARY SYSTEM

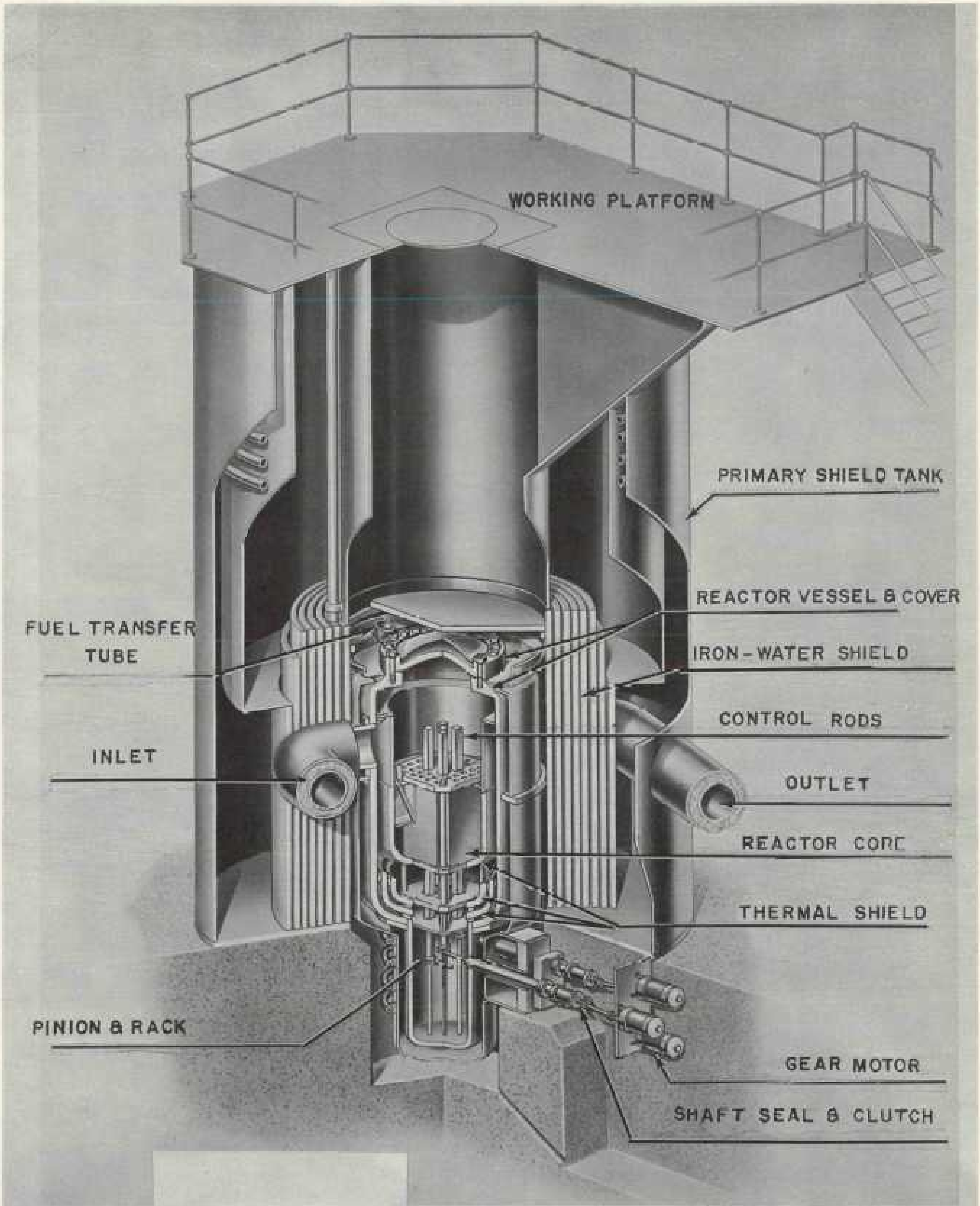


FIG.2 REACTOR AND PRIMARY SHIELD ARRANGEMENT

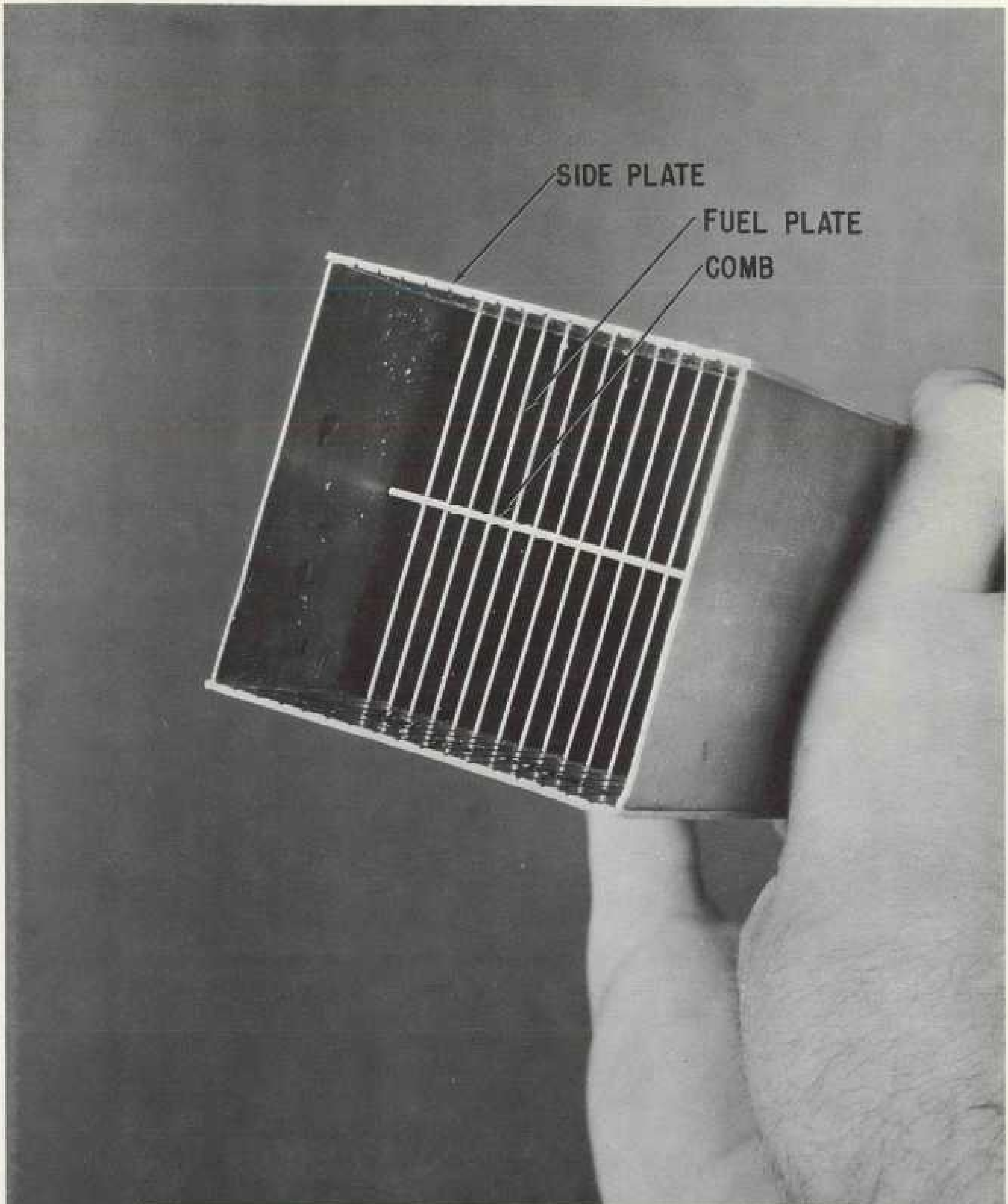


FIG. 3 FIXED ELEMENT WITH END BOX REMOVED

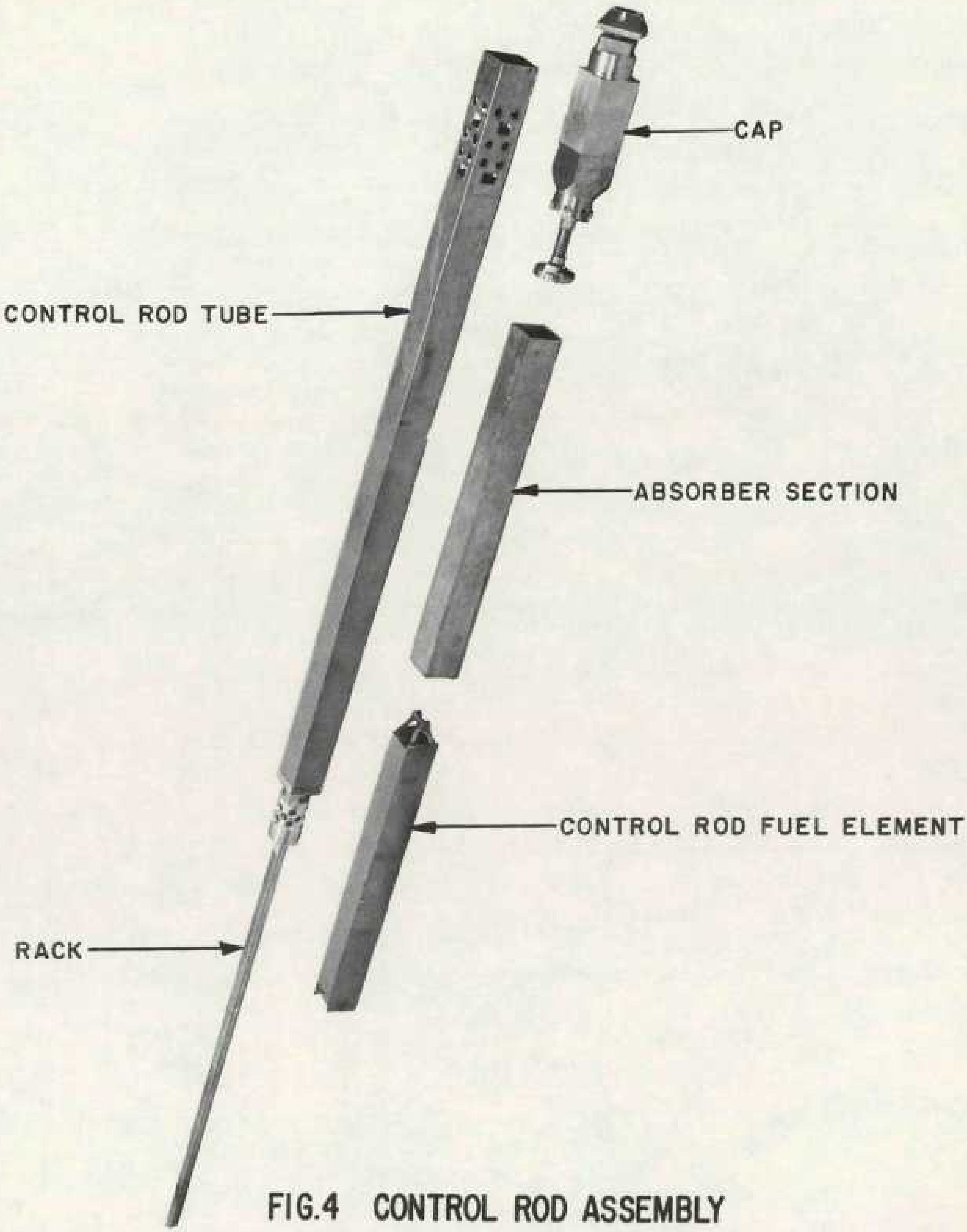


FIG.4 CONTROL ROD ASSEMBLY

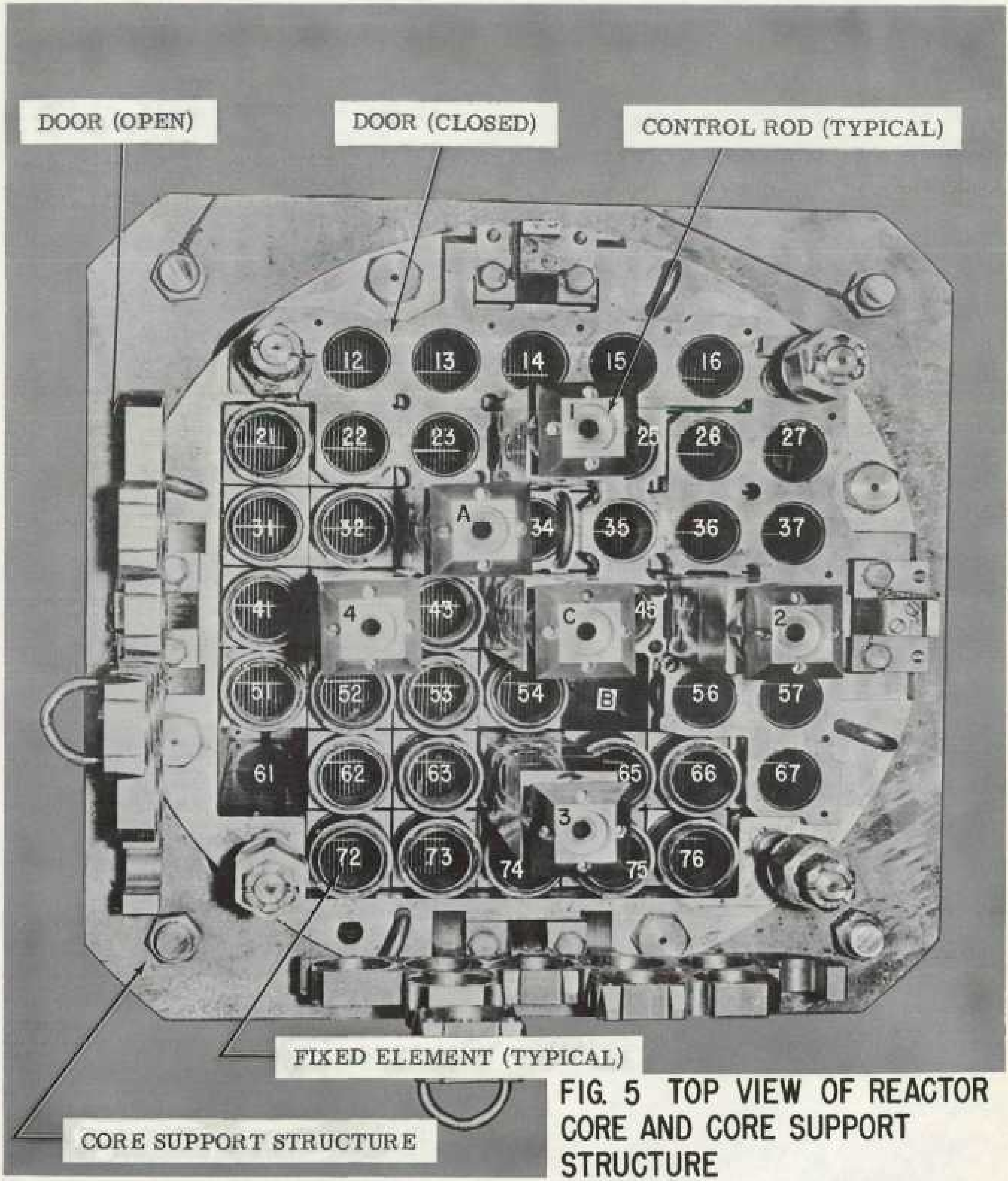
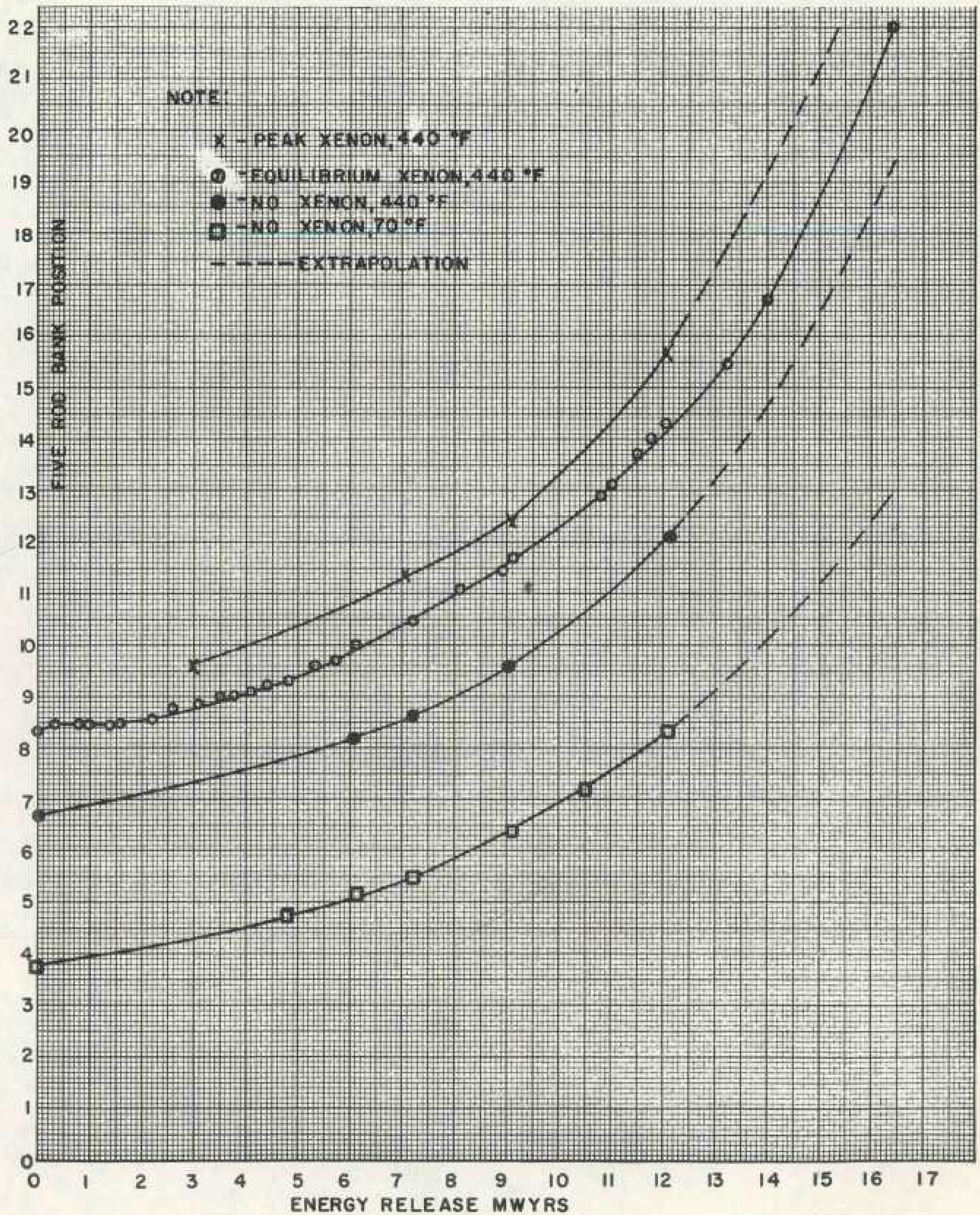


FIG. 5 TOP VIEW OF REACTOR CORE AND CORE SUPPORT STRUCTURE

FIG.6 FIVE ROD BANK POSITION VS. ENERGY RELEASE



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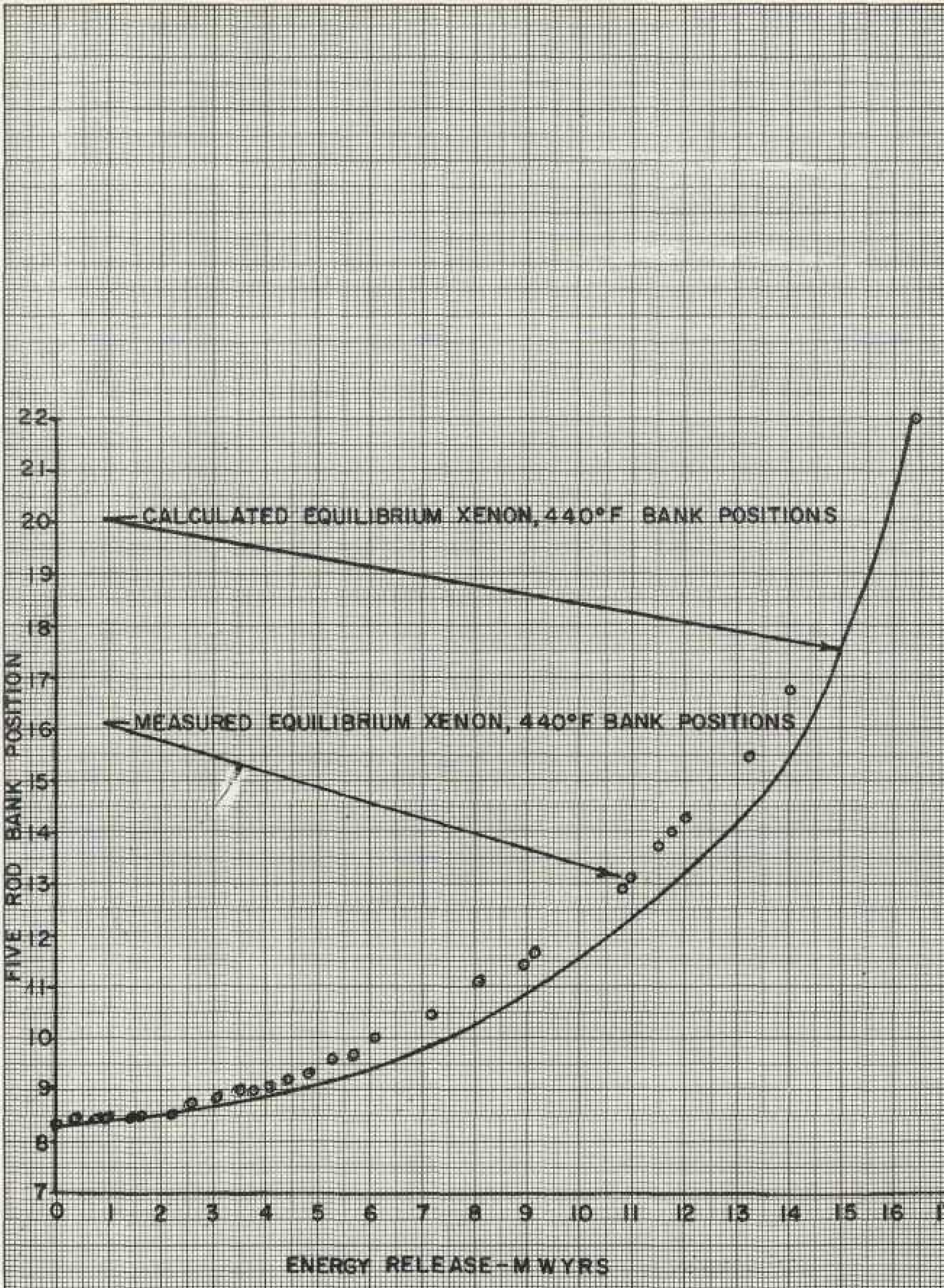

FIG.7 FIVE ROD BANK POSITION CALCULATED AND MEASURED

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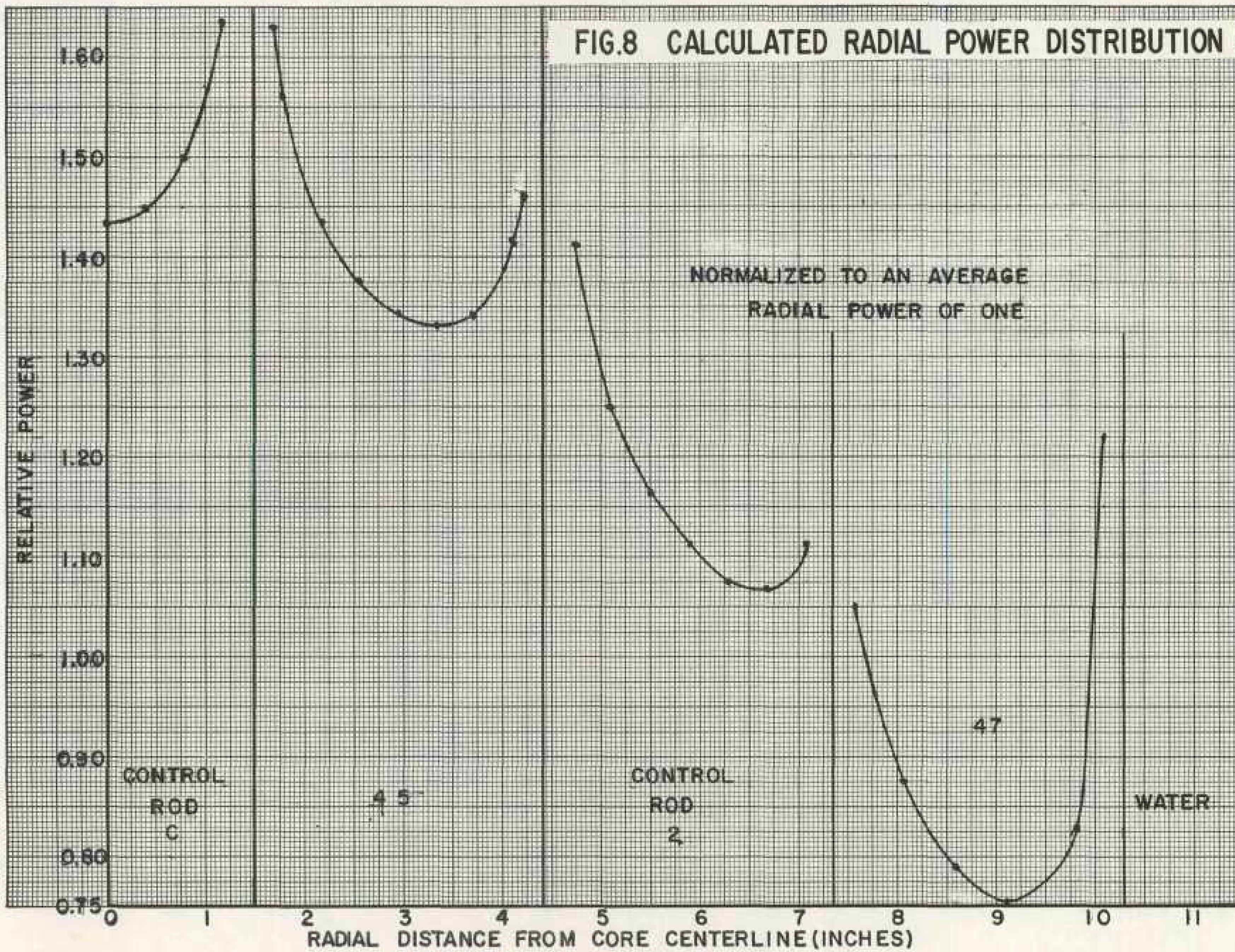
FIG.8 CALCULATED RADIAL POWER DISTRIBUTION
NORMALIZED TO AN AVERAGE
RADIAL POWER OF ONE
RELATIVE POWER
1.60
1.50
1.40
1.30
1.20
1.10
1.00
0.90
0.80
0.75
CONTROL
ROD
C
45
CONTROL
ROD
2
47
WATER
0
1
2
3
4
5
6
7
8
9
10
11
RADIAL DISTANCE FROM CORE CENTERLINE (INCHES)

223 032

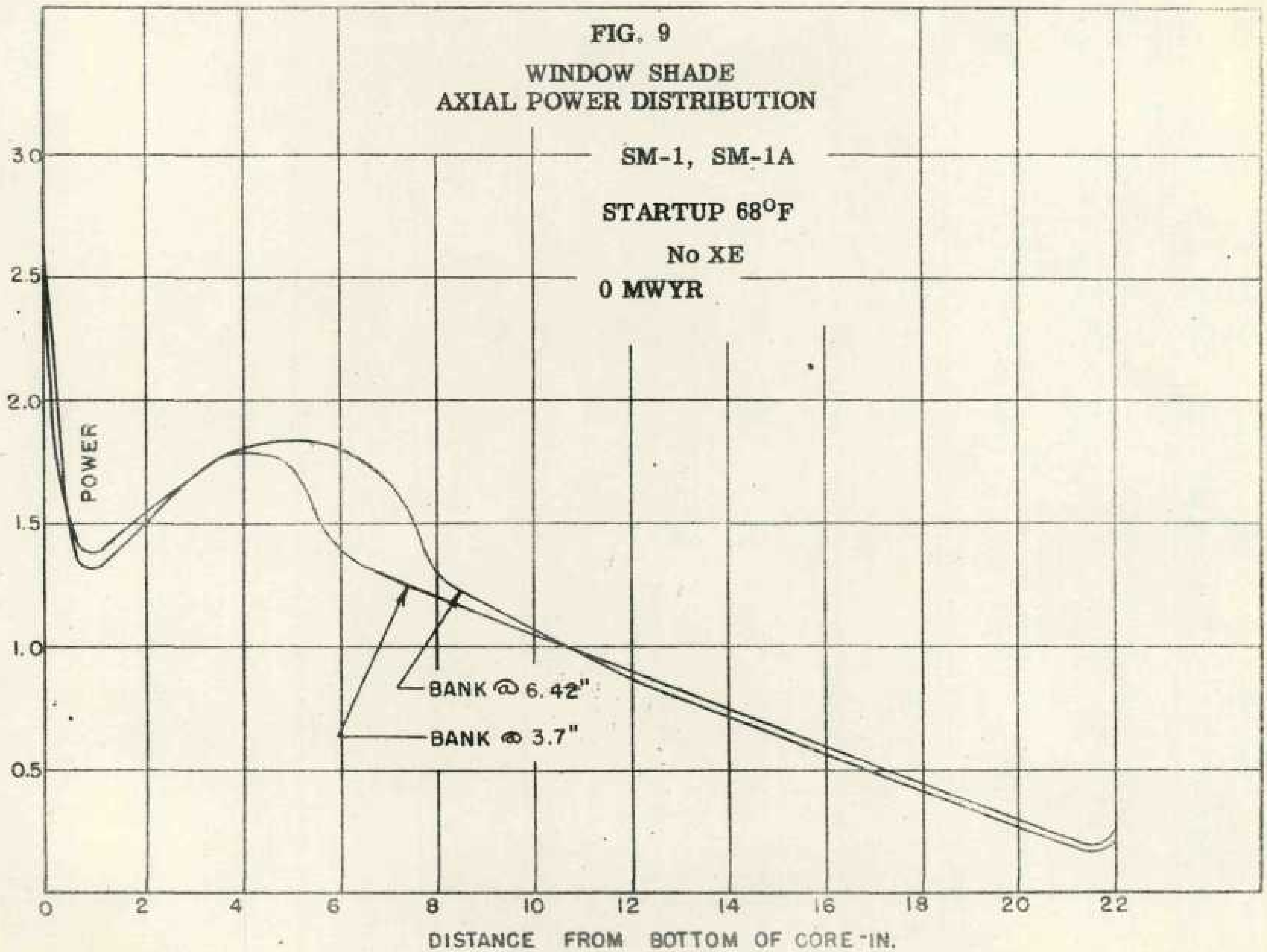

FIG. 9

WINDOW SHADE
AXIAL POWER DISTRIBUTION

SM-1, SM-1A

STARTUP 68°F

No XE

0 MWYR

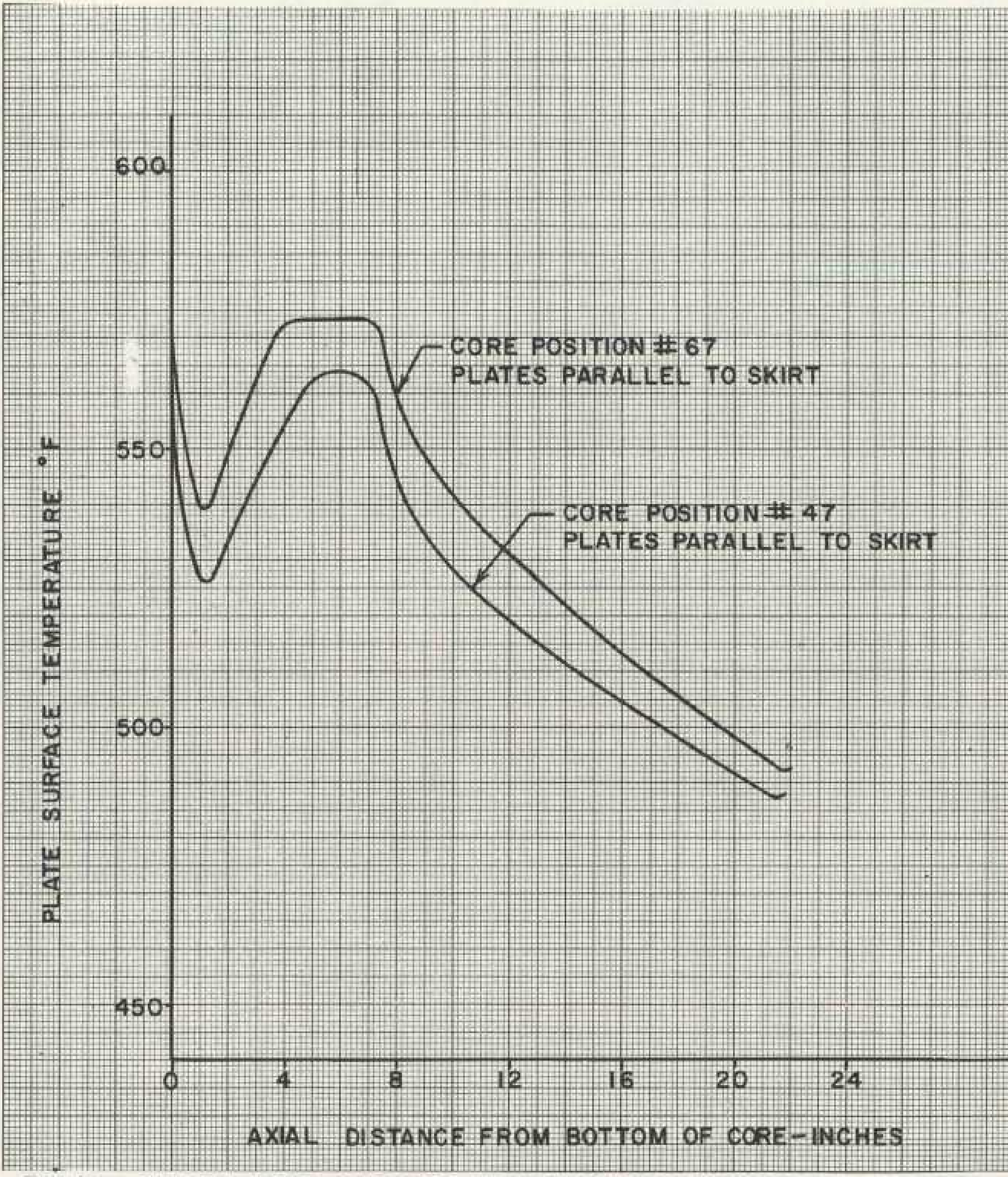

FIG.10 CALCULATED SURFACE TEMPERATURE IN NOMINAL CHANNEL AT BEGINNING OF CORE LIFE

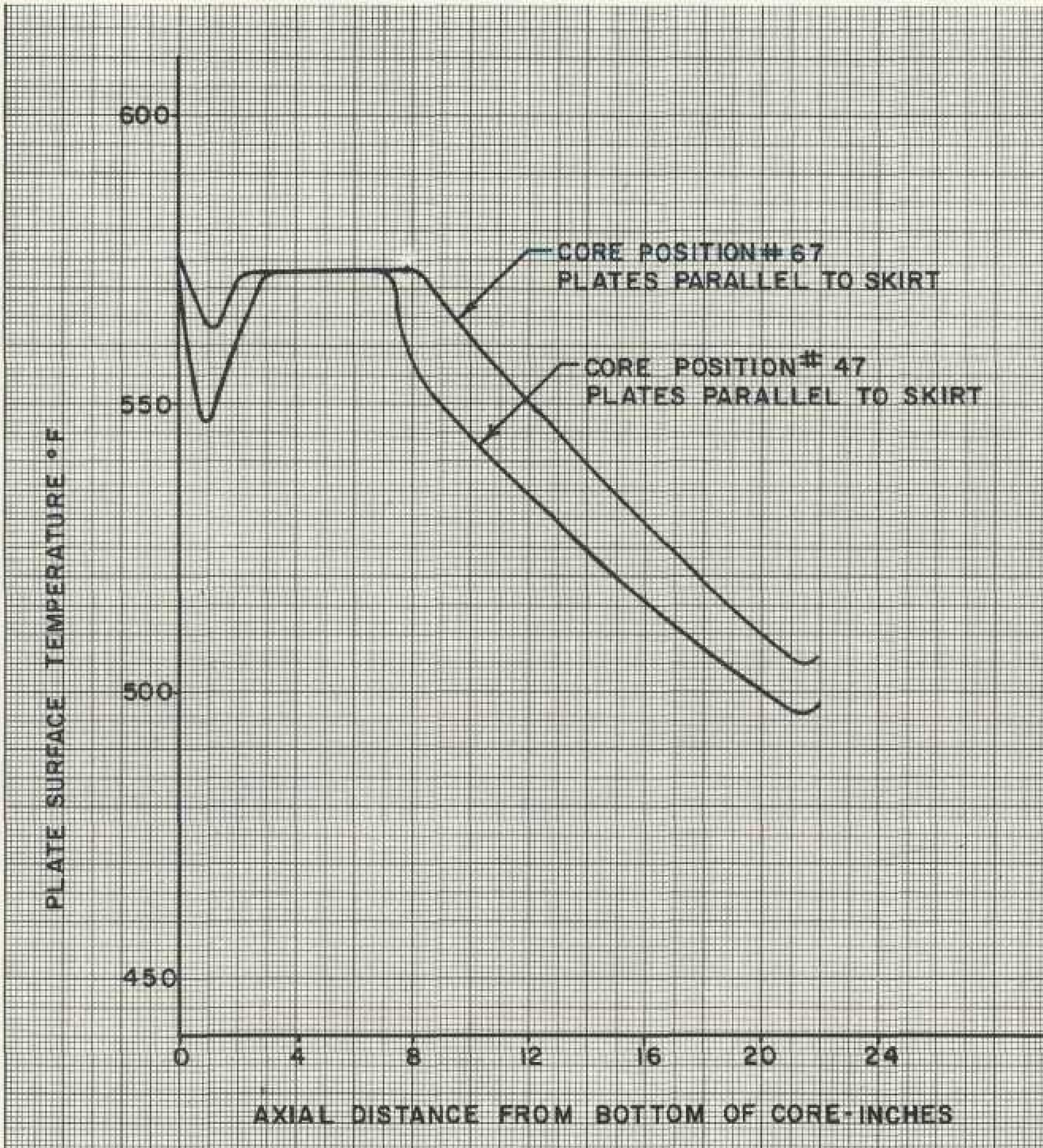

FIG.11 CALCULATED SURFACE TEMPERATURE IN HOT CHANNEL AT BEGINNING OF CORE LIFE

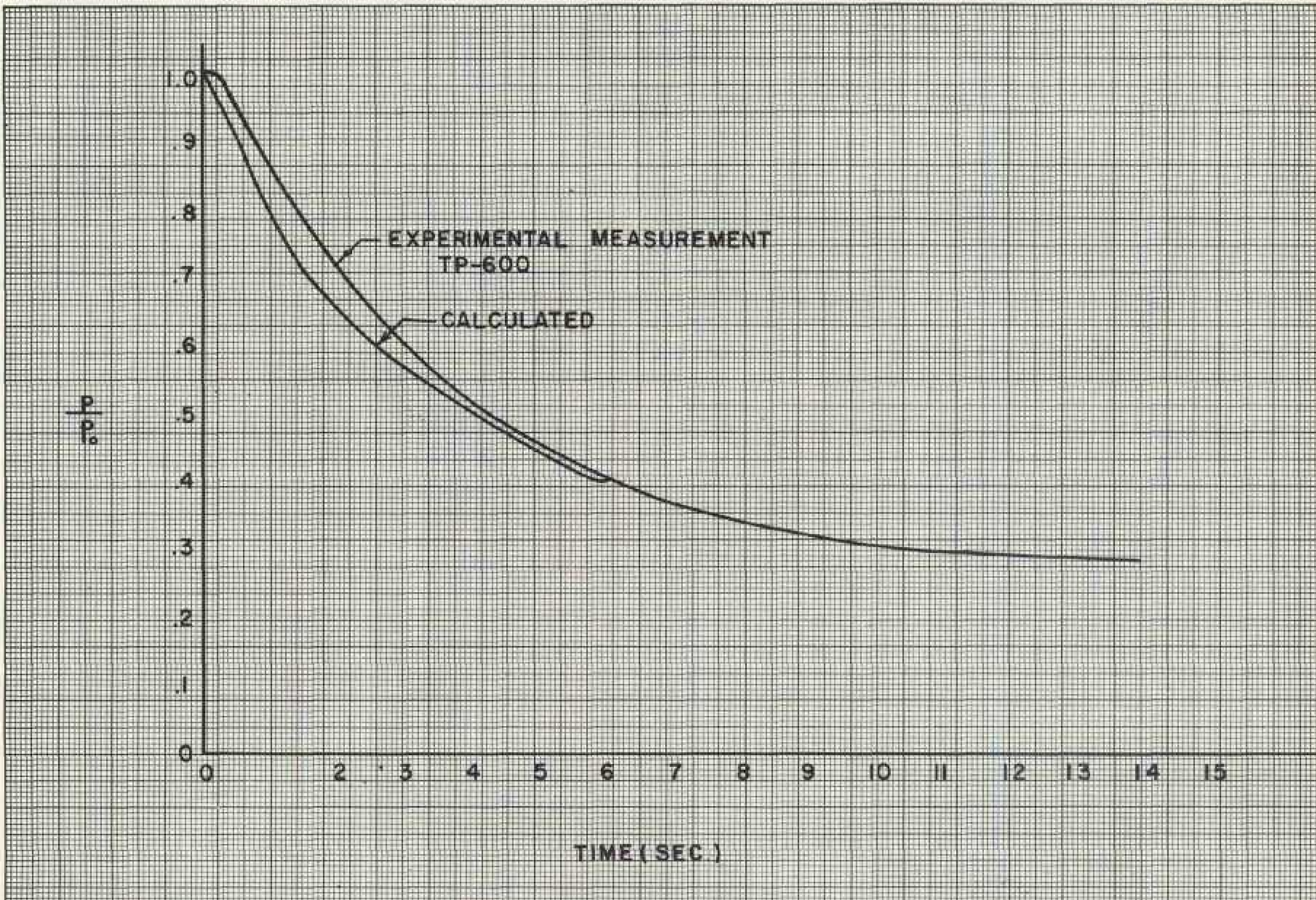

FIG. 12 REACTOR POWER VARIATION FOLLOWING DURING PUMP COASTDOWN

223 036

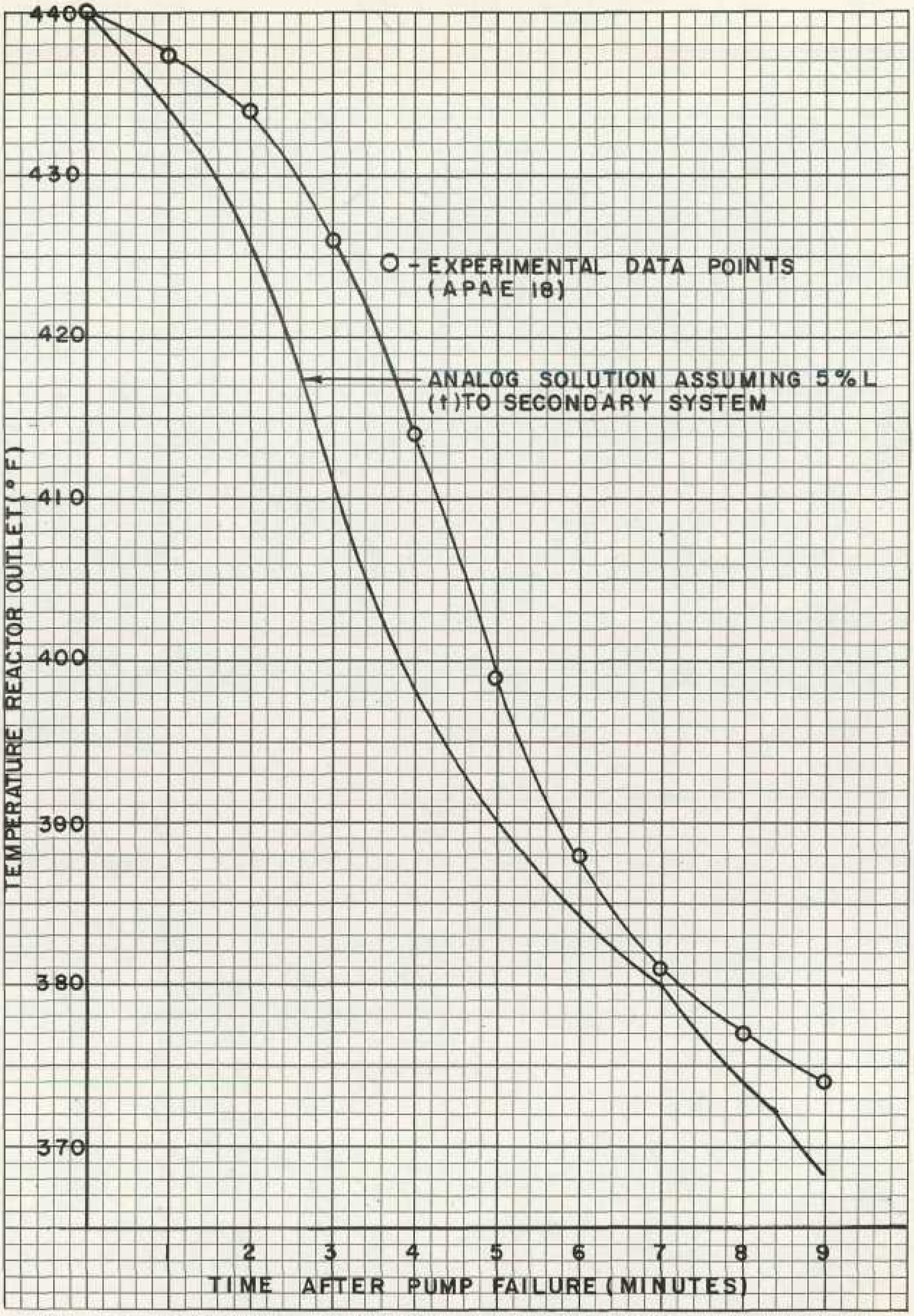

FIG. 13 REACTOR OUTLET TEMPERATURE VARIATION DURING DECAY HEAT REMOVAL

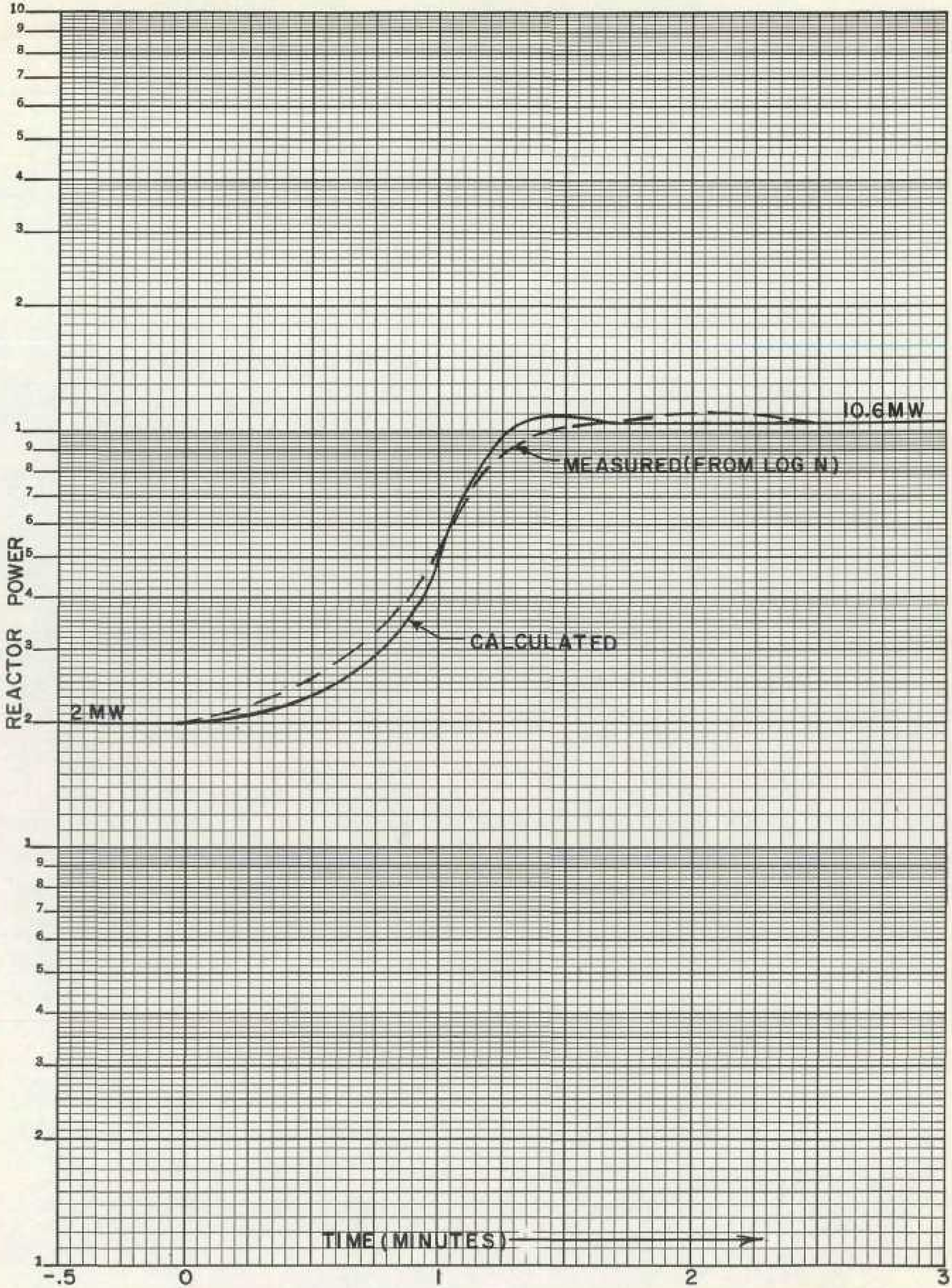

FIG. 14 REACTOR RESPONSE TO RAPID LOAD INCREASE

223 038

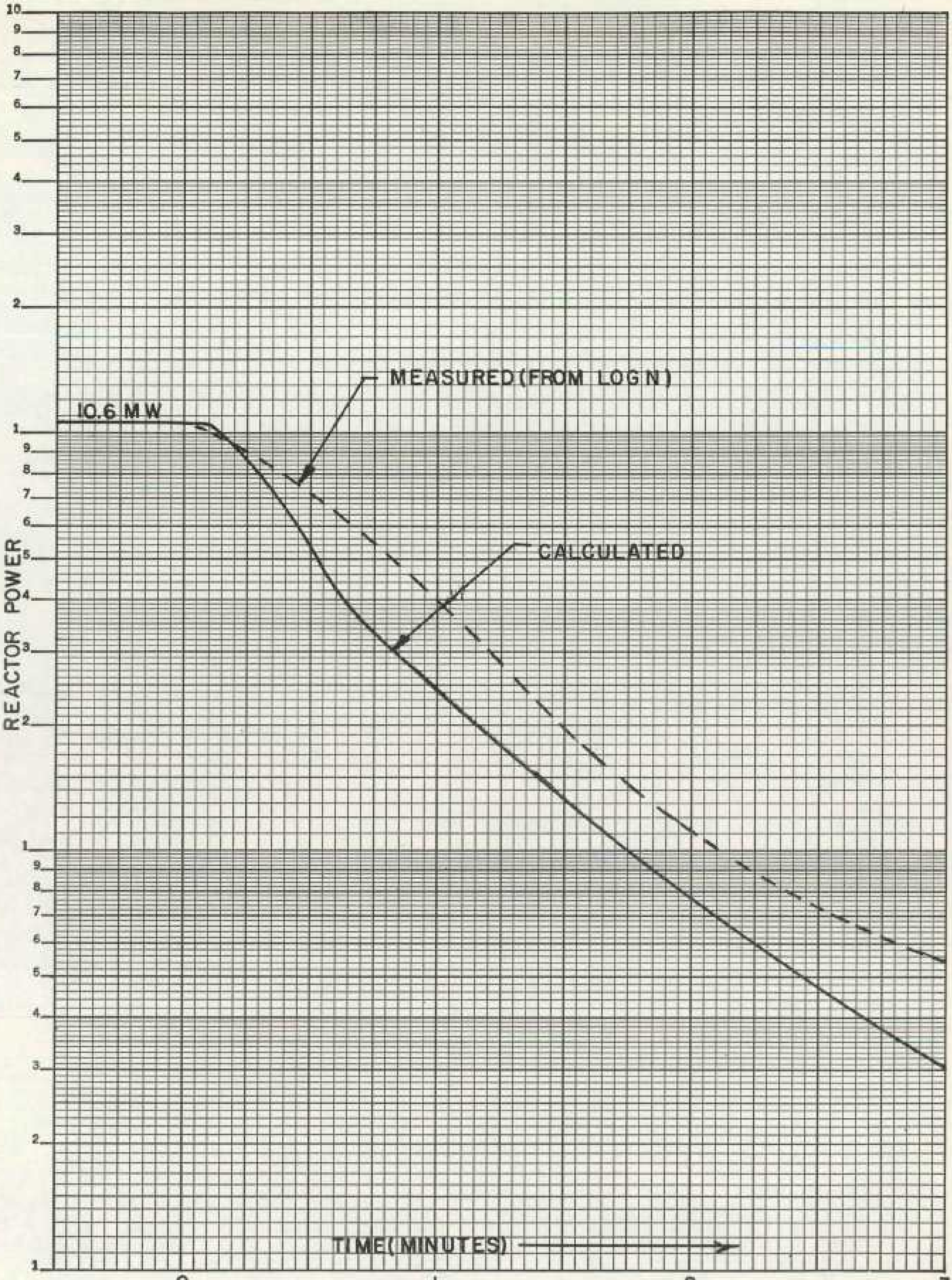

FIG. 15 REACTOR RESPONSE TO STEP REDUCTION IN LOAD

223 039

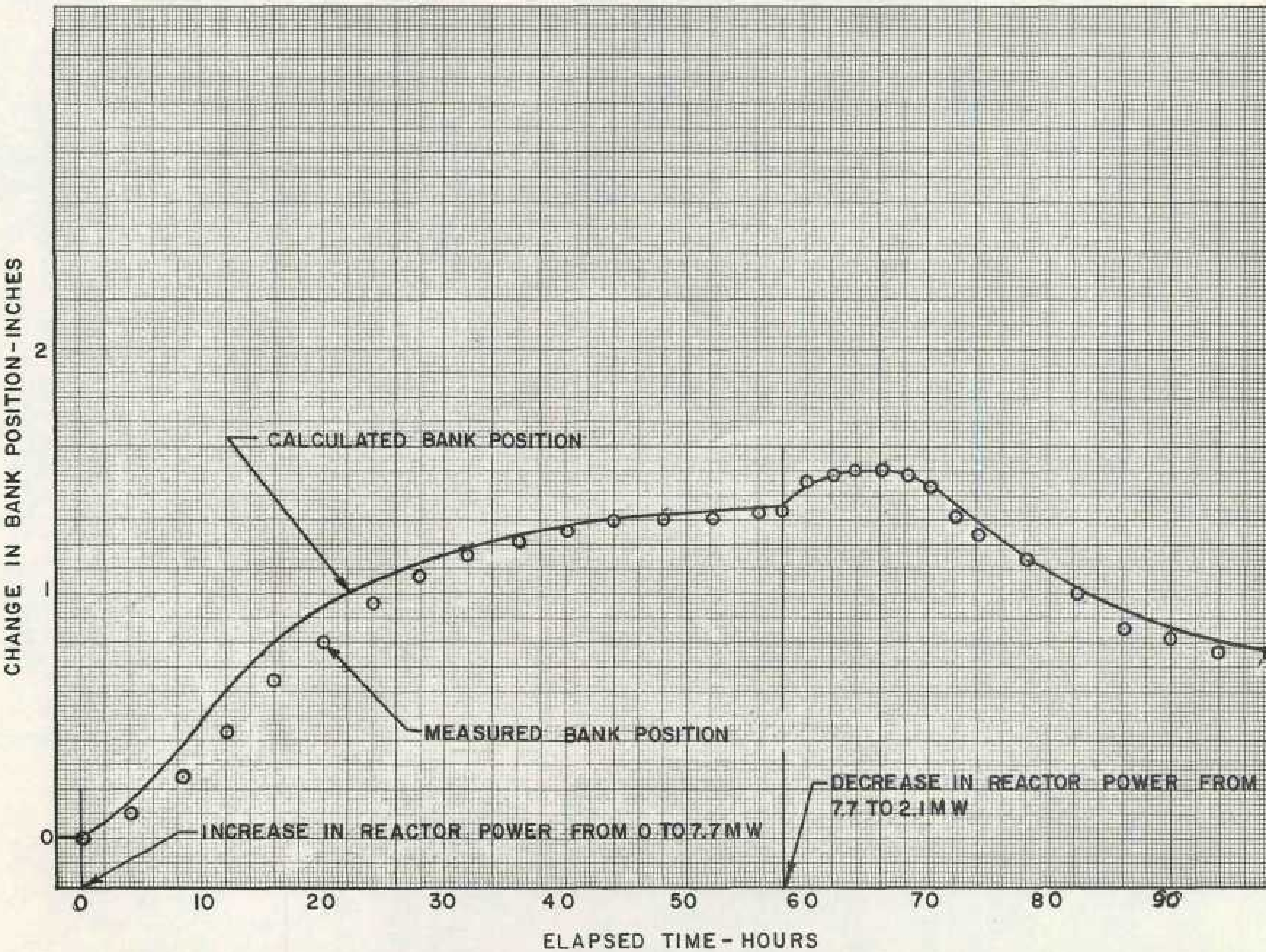

FIG. 16 BANK POSITION FOLLOWING A REDUCTION IN REACTOR POWER

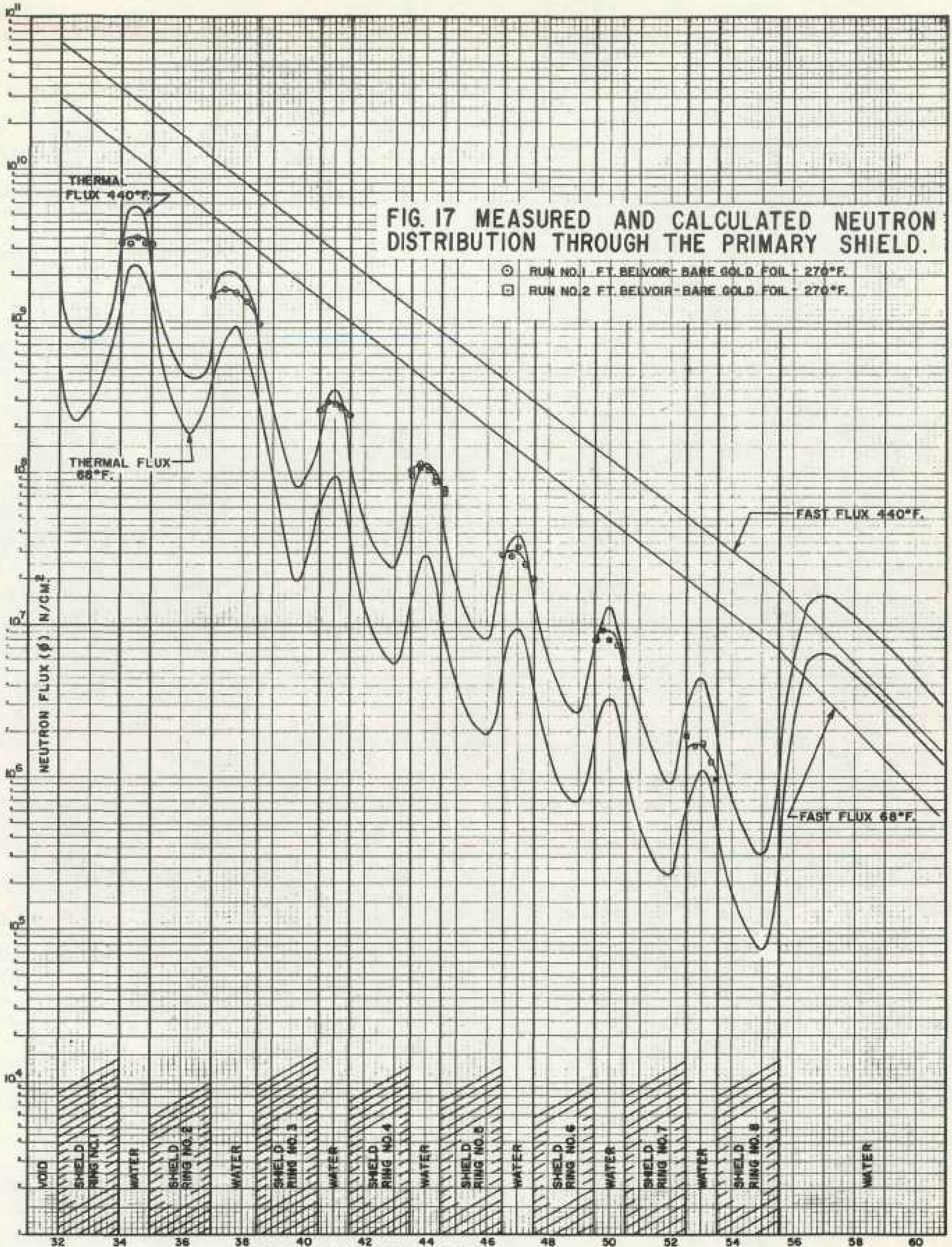
FIG. 17 MEASURED AND CALCULATED NEUTRON DISTRIBUTION THROUGH THE PRIMARY SHIELD.
RUN NO.1 FT. BELVOIR - BARE GOLD FOIL - 270°F.
RUN NO.2 FT. BELVOIR - BARE GOLD FOIL - 270°F.
THERMAL FLUX 440°F.
THERMAL FLUX 68°F.
FAST FLUX 440°F.
FAST FLUX 68°F.
NEUTRON FLUX (φ) N/CM²
DISTANCE FROM CORE CENTER (INCHES)
VOID
SHIELD RING NO.1
WATER
SHIELD RING NO.2
WATER
SHIELD RING NO.3
WATER
SHIELD RING NO.4
WATER
SHIELD RING NO.5
WATER
SHIELD RING NO.6
WATER
SHIELD RING NO.7
WATER
SHIELD RING NO.8
WATER

223 041

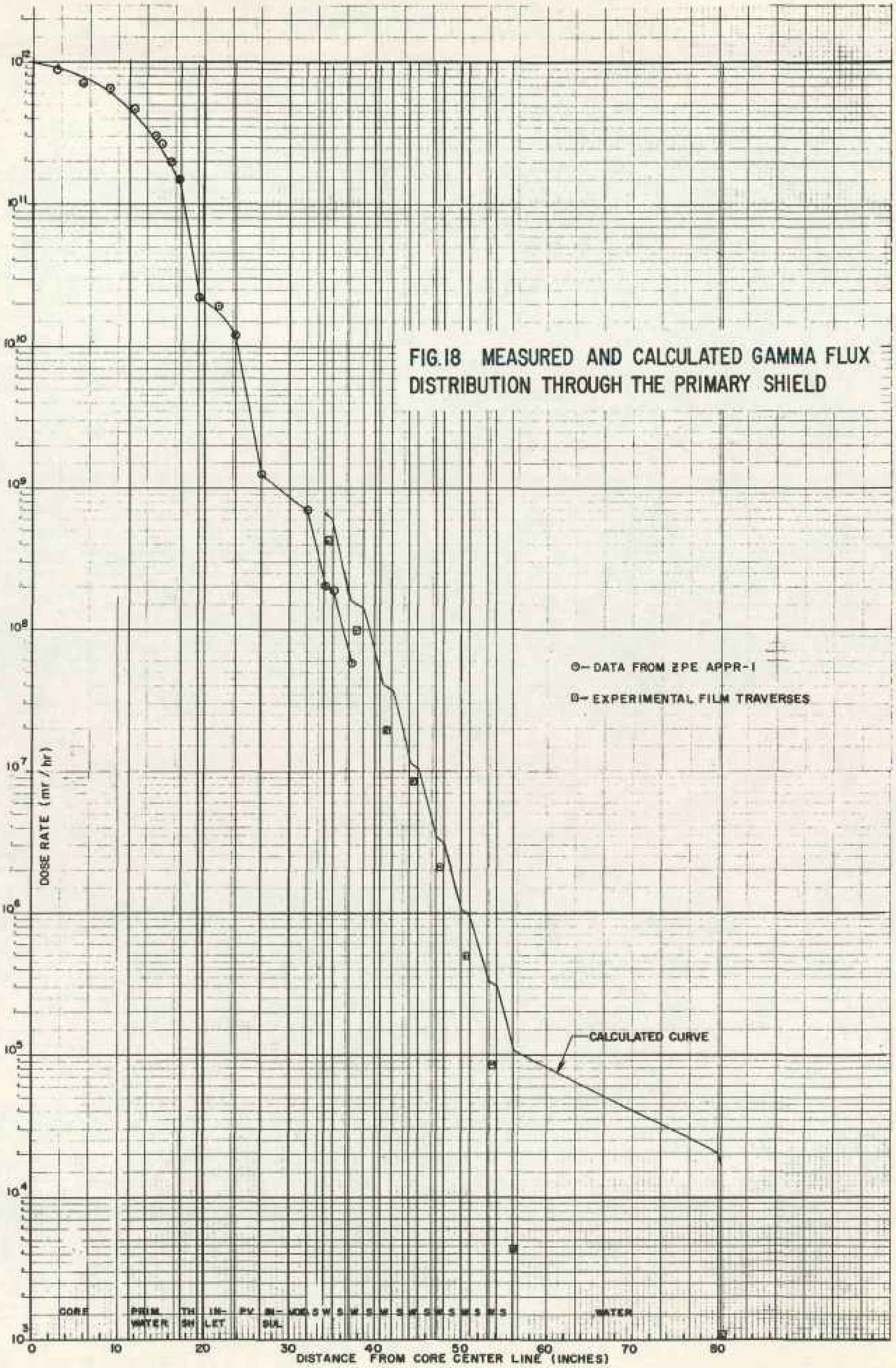

FIG.18 MEASURED AND CALCULATED GAMMA FLUX DISTRIBUTION THROUGH THE PRIMARY SHIELD

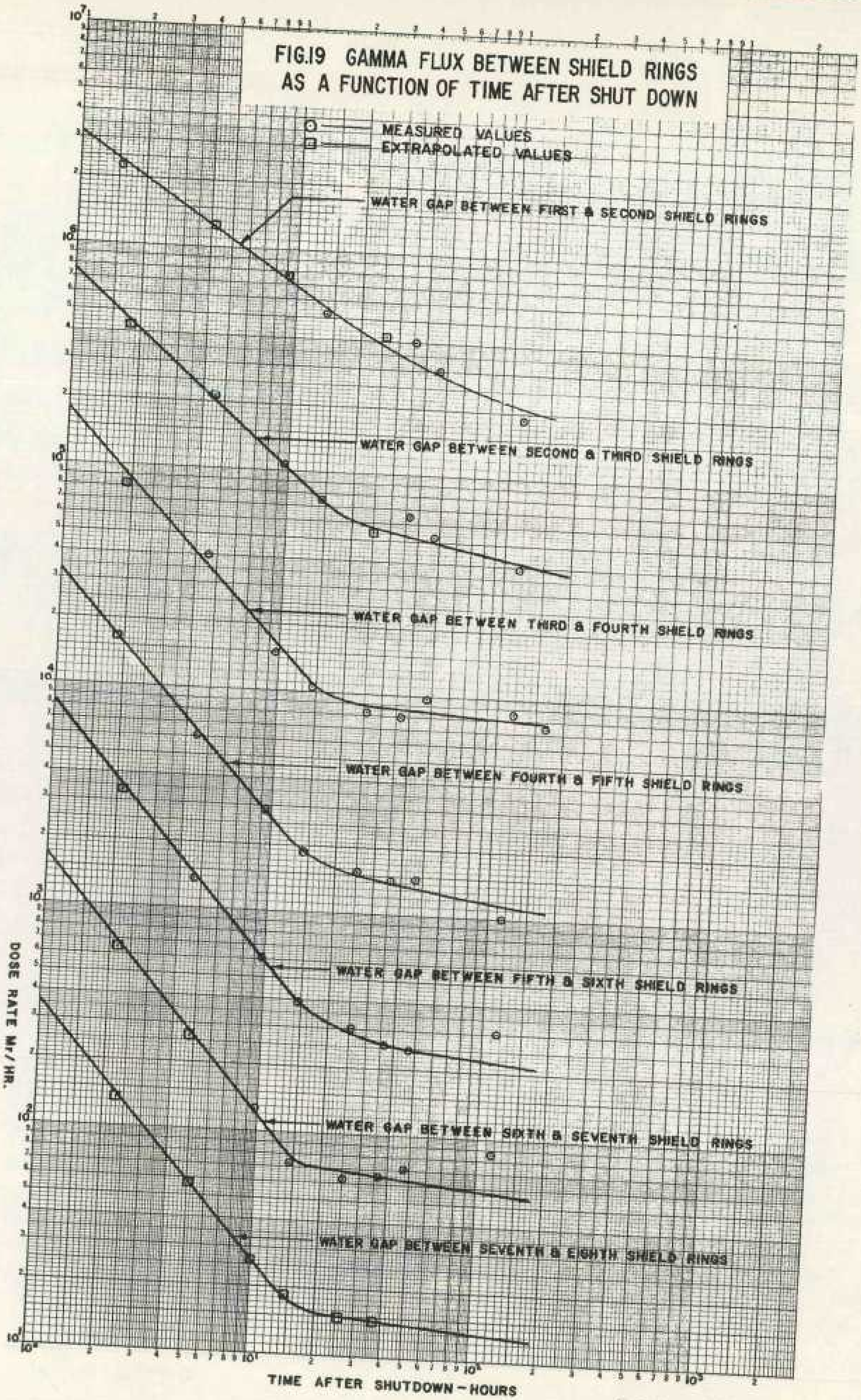
FIG.19 GAMMA FLUX BETWEEN SHIELD RINGS AS A FUNCTION OF TIME AFTER SHUT DOWN
MEASURED VALUES
EXTRAPOLATED VALUES
WATER GAP BETWEEN FIRST & SECOND SHIELD RINGS
WATER GAP BETWEEN SECOND & THIRD SHIELD RINGS
WATER GAP BETWEEN THIRD & FOURTH SHIELD RINGS
WATER GAP BETWEEN FOURTH & FIFTH SHIELD RINGS
WATER GAP BETWEEN FIFTH & SIXTH SHIELD RINGS
WATER GAP BETWEEN SIXTH & SEVENTH SHIELD RINGS
WATER GAP BETWEEN SEVENTH & EIGHTH SHIELD RINGS
DOSE RATE Mr/HR.
TIME AFTER SHUTDOWN - HOURS

223 043