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SHIELDING MEASUREMENTS AT THE SM-1 REACTOR
(JUNE 1961)

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ABSTRACT

Neutron flux and gamma radiation measurements through the SM-1 primary shield were made at the startup of Core II in June 1961. They extend previous measurements (APAE-35) both vertically and horizontally in the primary shield and in the rod drive pit. Dose rate measurements on spent fuel elements under water are also reported.

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1.0 SUMMARY

Neutron flux and gamma radiation measurements in the SM-1 reactor primary shield have been made to show leakage and attenuation patterns in vertical as well as horizontal directions. These measurements are shown as contour maps of neutron flux and gamma dose rates.

Thermal neutron flux mapping was done by irradiating bare and cadmium covered gold foils in the water channels of the shield tanks and in the rod drive pit. Fast neutron flux in the outer shield tank was measured by irradiation of sulfur pellets.

Gamma radiation dose rates were measured by irradiation of film badge packs in the shield tanks. Dose rate surveys were made in the rod drive pit. High gamma radiation levels observed in the rod drive pit are attributed to high neutron flux in this area. The radiation from spent fuel elements during handling operations was measured above the water surface in the rod drive pit and agreed well with caclulated values.

These measurements generally confirmed previous results reported in APAE-35 in respect to the shape of the flux distributions. In magnitude, however, the recent measurements were generally higher by a factor of three then were those shown by the previous results.

Recommendations are made for further studies of neutron flux and gamma dose rates in the rod drive pit and for additional fast flux measurements. Also, analysis of the shielding data in this report should be made to insure continuing adequacy of the shielding.

2.0 INTRODUCTION

A series of shielding measurements made at the SM-1 Army Package Power Reactor has been performed under the Program Plan for Engineering Support and Development of Army Pressurized Water Reactors.* The purpose of these measurements was to supplement and extend previous measurements reported in APAE-35 to assure safety of operation with optimum shielding design and to verify design methods.

The previous measurements were limited to radial traverses through the primary shield at a vertical level of the core midplane. However, additional data were required in the vertical as well as the radial direction in order that specifications could be accurately made for the heights above the core to which the shield has to be built at the various increasing radial distances from the core.

Since high levels of shutdown gamma dose rate in the rod drive pit interfere with maintenance on the drive components, it is important to know the distribution of gamma radiation. It was desirable to determine whether the source of activity in the SM-1 rod drive area was neutron induced or resulted from core shutdown gammas.

2.1 TEST DESCRIPTION

The measurements for this program were carried out as far as possible in accordance with the following test procedures:

TP-A401	Primary Shielding, Neutron Flux
TP-A402	Primary Shielding, Gamma Flux
TP-A403	Instrument Wells, Neutron Flux
TP-A404	Instrument Wells, Gamma Flux
TP-A406	Rod Drive Pit, Neutron Flux
TP-A407	Rod Drive Pit, Gamma Flux
TP-A408	Spent Fuel Element Transfer Dose Rate

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* AP Note-286, Item 4.4 (FY-61 Program), October 10, 1960, and AP Note-378, Subtask 5.3 (FY-62 Program), September 6, 1961.

The neutron flux measurements were made by activating gold foils mounted on aluminum strips. These strips were inserted in the water channels of the primary shield, in the inner shield tank above the vessel cover, on the outside of the shield tank, in instrument well A, and in the rod drive pit. At each location, some of the gold foils were cadmium covered to make corrections for the epithermal neutron flux. Sulfur pellets also were inserted in the primary shield channels to measure the fast flux by the S (n, p) reaction. (E 2.9 Mev)

Gamma radiation measurements were made by irradiating film badges in polyethylene bags mounted on aluminum strips. These strips were inserted in the water channels of the primary shield, in the inner shield tank, and in instrument well A. A portable survey meter was used for radiation measurements in the rod drive pit. Dose rate measurements were made just above the water surface in the spent fuel pit during handling and transfer of fuel elements under water, with various depths of water shielding.

Four reactor power runs were used for these tests, plus a gamma background run before Core II startup. They were planned to obtain continuity and overlap for film-badge and foil activation and are identified as follows:

Run 0 - Before startup for 8.0 hr on June 7, 1961. Film badges were placed in instrument well A to obtain background measurement.

Run 1 - Average power 11.1 kw for 20 min June 10, 1961. Film badges were placed in shield slots 1, 2, 3 and 4, and in instrument well A. (2.75 hr total exposure)

Run 2 - Average power 26.6 kw for 8.0 hr June 10, 1961. (12.5 hr in reactor shield) Film badges were placed in shield slots 1, 2, 3 and 4, and in instrument well A. (12.5 hr total exposure)

Run 3 - Average power 237 kw for 60 hr June 10-14, 1961. Film badges were placed in the inner shield tank. Foils were placed in shield slots 4, 5, 6 and 7, and in the inner shield tank. (93 hr total exposure)

Run 4 - Average power 10.8 Mw for 64 hr June 14-24, 1961. Foils were left in place from Run 3, above in shield slots 4, 5, 6 and 7, and in inner shield tank. In addition, foils were placed outside the shield tank and in the rod drive pit. Sulfur pellets and foils were placed in shield slots 1, 3 and 6. (10 days total exposure)

There was difficulty in determining the power level of runs 1, 2 and 3 because the nuclear instruments, particularly the Log N channel, had not been properly repositioned and recalibrated at startup of SM-1 Core II. During the early part of Run 4, the Log N ion chamber was raised on six different days by a total of 17 in.; and after the full power run, it was again raised 1 in. for its final adjustment. The Log N power-level correction was determined by three methods. At the end of Run 3, the generator was operated briefly at a load of 400 kw, and the Log N reading was about 200 percent, which was too high by about a factor of 5. The ΔT integrator was recalibrated during Run 3 and was started up at the beginning of Run 4 for Core II operation.

The integrator reading was used as the correct energy output for Run 4. The thermal energy output for each day was compared with the value obtained by integrating the Log N chart readings to obtain a correction factor for the initial position of the ion chamber (Runs 1, 2 and 3) and for each of the six positions during Run 4. Finally, the flux distribution curve in instrument well A, Fig. 3.1, was used to determine an improved Log N correction factor for each of the seven positions. The final correction factor to the Log N readings for Runs 1, 2 and 3 was 0.222. This gives a power level believed to be within 50 percent of the correct value.

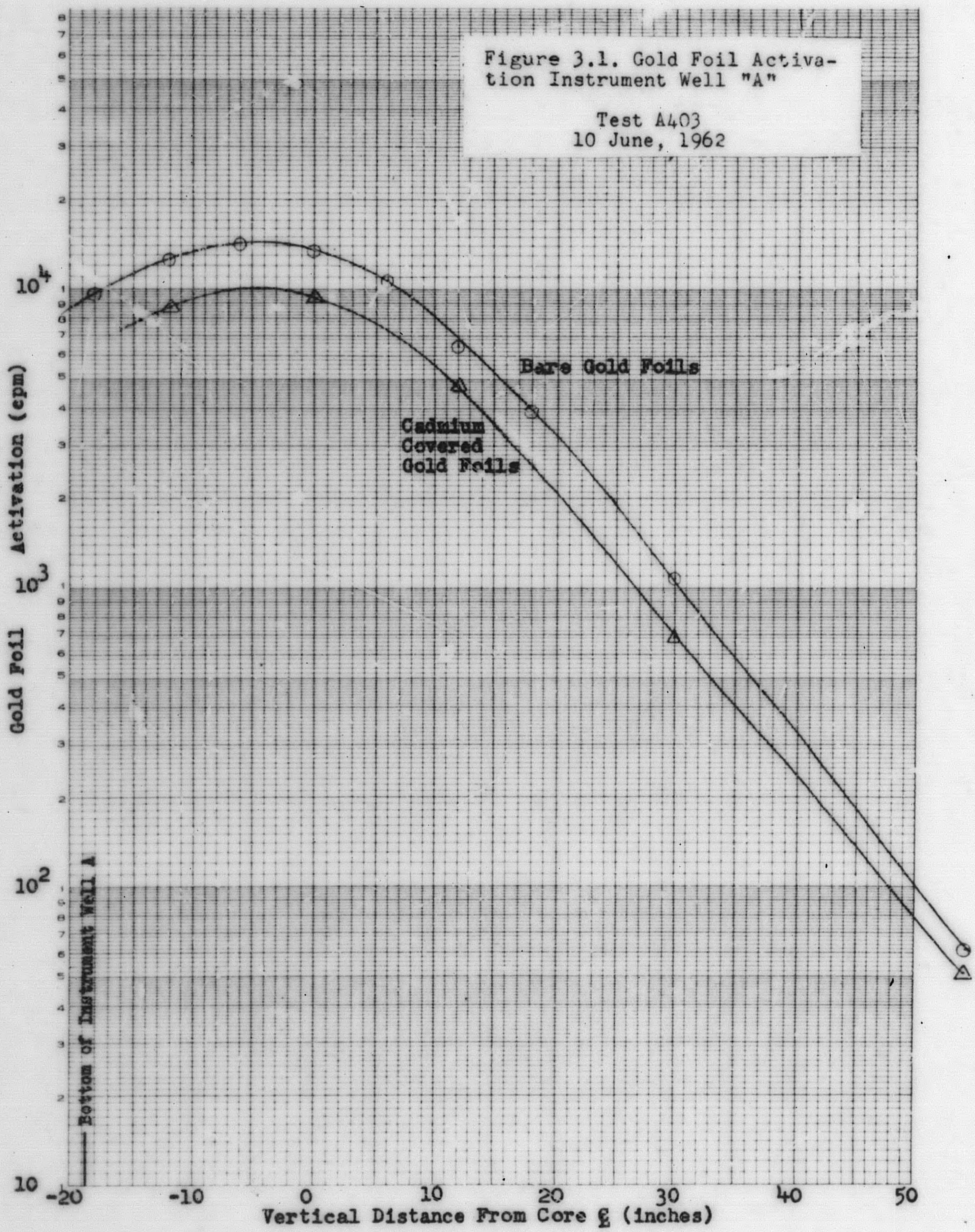

Figure 3.1. Gold Foil Activation Instrument Well "A"
Test A403
10 June, 1962
Bare Gold Foils
Cadmium Covered Gold Foils
Bottom of Instrument Well A
Gold Foil Activation (cpm)
Vertical Distance From Core ℄ (inches)
10
10^2
10^3
10^4
-20
-10
0
10
20
30
40
50

3.0 NEUTRON FLUX MEASUREMENTS

The earlier gold foil activations reported in APAE-35[1] were done at only one elevation, the core centerline. They were not reduced to absolute flux measurements, but the relative activities were normalized to agree with flux calculations using the VALPROD computer code.

In the present measurements, bare and cadmium covered gold foils were inserted at several elevations in the shield sections, as well as along the inner shield tank axis, outside the shield tank, and in the rod drive pit, in order to explore the flux pattern more completely. An attempt was made to reduce the activation measurements to absolute flux measurements, although many uncertain factors were involved.

The activated gold foils were removed and counted on the single channel analyzer at the SM-1 reactor; then they were shipped to the Alco Criticality Facility at Schenectady where those from Run 2 were counted again. The latter counter efficiency was known to be 2.39 percent from previous measurements with a standard foil. The SM-1 counting data for Run 4 was normalized to agree with the critical facility counting data for Run 2. The SM-1 counter efficiency and background were widely different for each run, so that no standard values could be used.

The usual relationship for thermal flux and foil activity was used:

$$\frac{\text{Activity (cps)}}{\text{mass (gm) x efficiency}} = \frac{\hat{\sigma} N_o \phi}{A} (1-e^{-\lambda T})\, e^{-\lambda t}$$

where:

$\hat{\sigma}$ = effective activation cross section for gold foil
N_o = Avogadro's number = .6023 x 10^{24} gm-mole
A = atomic weight of Au^{197} = 197 gm-mole
ϕ = thermal neutron flux at 2200 m/sec
λ = decay constant for Au^{198} = .0107 hr^{-1}
T = time of irradiation for run
t = time of decay from shutdown to counting

The use of an effective cross section is described in ANL-5800.[2] Corrections are made for a Maxwellian flux distribution, neutron temperature as affected by flux hardening, an epithermal flux index, and an effective resonance integral. Additional corrections were made for self-shielding in the gold foil: for both thermal and resonance energies and for epithermal shielding by the cadmium covers, as described in Section 8 of ANL-5800.[2]

The effective cross section can be written as:

$$\hat{\sigma} = \sigma_0 (g + rs)$$

or with self-shielding factors

$$\hat{\sigma} = \sigma_0 (g F_0 + rs F_{res})$$

where:

σ_0 = cross section for 2200 m/sec neutrons
g = correction for Maxwellian flux at temp. T
F_0 = thermal self-shielding factor = 0.93
r = epithermal neutron density fraction
s = a form of effective resonance integral, at temp. T
F_{res} = resonance self-shielding factor = 0.56

The epithermal fraction r is determined from the cadmium ratio (bare-foil activity/cadmium covered foil activity). Table 3.1 shows the cadmium ratio, epithermal fraction, and effective cross section at various positions in the shielding. The effective cross section was first calculated for instrument well A, where it was found to be 98.3 x 2.56 barns. This value was used to calculate all the thermal flux values tabulated and plotted in this report. Correction factors at various positions in the shield can be determined from Table 3.1.

3.1 NEUTRON FLUX IN INSTRUMENT WELL A - TEST A403

Gold foils, both bare and cadmium covered, were mounted on a 3 in. wide aluminum strip, which fitted loosely across the diameter of an instrument well, and were lowered into well A for Run 2. The strip was oriented perpendicular to the radius from the core centerline, so that all foils were close to the same radial distance of 36.37 in. from the core center. Bare and cadmium covered foils at the same elevation were spaced 3 in. apart (the width of the strip) to avoid interference by flux depression.

The positions and relative activities in counts per minute are shown in Table 3.2 and Fig. 3.1. The cadmium ratio (bare foil activity/cadmium covered foil activity) averaged 1.46. The activities have been converted to thermal flux at 10 Mw, and are included with the flux measurements of the following section.

3.2 NEUTRON FLUX IN PRIMARY SHIELD - TEST A401

Gold foils were mounted on aluminum strips and inserted into the 1 in. water channels between the 2 in. steel shield rings in the outer shield tank. A strip was inserted in each of channels 1, 2, 3 and 4 for Run 2, and in channels 4, 5, 6 and 7 as well as the inner shield tank axis for Runs 3 and 4. For Run 4 another strip of foils was mounted outside the shield tank at a position behind

TABLE 3.1
CADMIUM RATIOS, EPITHERMAL FRACTION AND EFFECTIVE GOLD CROSS SECTION

Position	Rod	r	σ
Instrument Well A	1.46 to 1.5	0.154	98.3 x 2.56
Shield Channel 1	1.8 to 2.0	0.103	2.03
Shield Channel 2	1.85 to 2.5		
Shield Channel 3	2.0 to 2.3	0.080	1.79
Shield Channel 4	1.85 to 2.5	0.054	1.51
Shield Channel 4	2.3 to 2.8		
Shield Channel 5	3.0 to 3.2	0.041	1.37
Shield Channel 6	3.5 to 3.0		
Shield Channel 7	1.8 to 1.8		

the steam generator and near the corner of the rod drive pit. Several other foils were mounted in the rod drive pit in positions previously used for gamma radiation measurements on the 4 in. by 4 in. wood shield and on the water box face.

The positions of the foils, their activity, and the calculated thermal neutron flux at 10 Mw are shown in Tables 3.2 to 3.5 and Fig. 3.2 to 3.6. Figure 3.2 shows that the peak flux is about 4 in. below the core centerline and that the slope of the vertical attenuation curves is steeper above the core than below it. There was evidently a positioning error for the strip of foils in channel 2, so that all points in this channel were adjusted by 8 in. to fit the other curves. Figure 3.3 shows the radial attenuation through the shield at each elevation. These flux values differ in magnitude from the results in APAE-35,[1] Fig. 1.3 and are generally higher by a factor of 3 to 5 than the previous measurements, as normalized on the inner shield tank axis are shown in Table 3.3 and Fig. 3.4. Only the three lower foils had activities significantly above background; hence, the flux distribution is not very well defined.

The flux measurements on the outside of the shield tank and in the rod drive pit are shown in Tables 3.4 and 3.5 and in Fig. 3.5. The neutron flux in the rod drive pit was much higher than expected and could account for some activation of components and the observed gamma radiation levels. Previous measurements reported in APAE Memo-237[3] are approximately a factor of 10 lower and those in APAE-18[4] are a factor of 100 lower. The foils outside the shield tank were located close to one corner of the rod drive pit, and the shape of the vertical flux distribution curve indicates a neutron source from below, i.e. in the rod drive pit rather than from the reactor core.

All the flux measurements are shown in Fig. 3.6 as flux contours in the primary shield structure.

3.3 FAST FLUX MEASUREMENTS

Sulfur pellets were mounted on aluminum strips and inserted in shield channels 1, 3 and 6 for Run 4. The pellets were sent to the Alco Criticality Facility laboratory at Schenectady for measurement of activity and calculation of the fast flux ($E_n > 2.9$ Mev) from the S^{32} (n, p) P^{32} reaction. The results are shown in Table 3.6 and in Fig. 3.7 and 3.8.

Some gold foils were also mounted on the strips in channels 3 and 6, as shown in Table 3.2, but the activities were inconsistent with each other and with the other runs.

TABLE 3.2
THERMAL NEUTRON FLUX THROUGH SHIELD RINGS

Run 2 - 8.0 Hr at 26.6 Kw

Vertical Position From Core Centerline, In.	corr	Activity, Cpm	Background, Cpm	Act. Net, Cpm	Decay Time, Hr	$e^{\lambda t}$	Foil Mass, Gm	Neutron Flux at 10 Mw, N/cm^2 sec
INSTRUMENT WELL A								
-18		9882	163	9719	111.2	3.28	.0314	4.22×10^{9}
-12		12790	163	12627	111.1	3.28	.0314	5.48×10^{9}
-6		14244	163	14081	111.0	3.28	.0314	6.12×10^{9}
0		13660	163	13497	111.1	3.28	.0314	5.87×10^{9}
+6		10967	163	10804	111.0	3.28	.0314	4.69×10^{9}
+12		6717	163	6554	111.0	3.28	.0314	2.85×10^{9}
+18		4127	163	3964	111.2	3.28	.0314	1.73×10^{9}
+30		1249	163	1086	111.3	3.28	.0314	4.72×10^{8}
+54		225	163	62	111.0	3.28	.0314	2.69×10^{7}
+72		184	163	21	111.0	3.28	.0314	9.12×10^{6}
SHIELD CHANNEL #1								
-29		6134	163	5971	110.033	3.25	.0311	2.58×10^{9}
-12		26716	163	26553	110.083	3.25	.0311	1.15×10^{10}
0		39731	163	39568	110.05	3.25	.0311	1.72×10^{10}
+12		21299	163	21136	110.017	3.25	.0311	9.18×10^{9}
+30		3738	163	3575	110.117	3.25	.0311	1.55×10^{9}
+54		512	163	349	110.15	3.25	.0311	1.51×10^{8}
+72		184	163	21	109.982	3.25	.0311	9.12×10^{6}
SHIELD CHANNEL #2								
-29	-21	5560	163	5397	109.8	3.24	.0317	2.29×10^{9}
-12	-12	14868	163	14705	109.6	3.24	.0317	6.25×10^{9}
0	+8	10711	163	10548	109.7	3.24	.0317	4.48×10^{9}
+12	+20	3482	163	3319	109.7	3.24	.0317	1.41×10^{9}
+30	+38	502	163	339	109.7	3.24	.0317	1.44×10^{8}
SHIELD CHANNEL #3								
-29		645	163	482	109.4	3.22	.0310	2.08×10^{8}
-12		5468	163	5305	109.1	3.22	.0310	2.29×10^{9}
0		6492	163	6329	109.1	3.22	.0310	2.72×10^{9}
+12		2847	163	2684	109.1	3.22	.0310	1.16×10^{9}
+30		369	163	206	109.3	3.22	.0310	8.87×10^{7}
+54		184	163	21	109.2	3.22	.0310	9.10×10^{6}
SHIELD CHANNEL #4								
-29		287	163	124	110.7	3.26	.0316	5.33×10^{7}
-12		922	163	759	110.5	3.26	.0316	3.26×10^{8}
0		1413	163	1250	110.6	3.26	.0316	5.37×10^{8}
+12		983	163	820	110.4	3.26	.0316	3.52×10^{8}
+30		276	163	113	110.4	3.26	.0316	4.84×10^{7}
+54		Damaged						
+72		195	163	32	110.5	3.26	.0316	1.37×10^{7}

TABLE 3.2 (CONT'D)

Run 3 - 60 Hr at 237 Kw
Run 4 - 64 Hr at 10.8 Mw (Normalized to Run 2, Channel 4)

Vertical Position From Core Centerline, In.	Activity, Cpm	Background, Cpm	Act. Net, Cpm	Decay Time, Hr	$e^{\lambda t}$	Foil Mass, Gm	Neutron Flux at 10 Mw, N/cm^2 sec
			SHIELD CHANNEL #4				
-29	59479	504	58975	89.5	2.55	.0316	3.65×10^7
-12	837515	504	837011	89.5	2.55	.0316	5.12×10^8
0	870293	504	869789	89.5	2.55	.0316	5.37×10^8 *Norm.
+12	584621	504	584117	89.5	2.55	.0316	3.61×10^8
+30	29287	504	28783	89.5	2.55	.0316	1.78×10^7
+54	953	504	449	89.5	2.55	.0316	2.78×10^5
+72	699	504	195	89.5	2.55	.0316	1.20×10^5
			SHIELD CHANNEL #5				
-29	16173	541	15632	86.0	2.505	.0311	9.5×10^6
-12	187035	541	186494	86.0	2.505	.0311	1.13×10^8
0	256846	541	256305	86.0	2.505	.0311	1.56×10^8
+12	107540	541	106999	86.0	2.505	.0311	6.51×10^7
+30	6339	541	5798	86.0	2.505	.0311	3.52×10^6
+54	611	541	70	86.0	2.505	.0311	4.24×10^4
+72	nil	541				.0311	
			SHIELD CHANNEL #6				
-29	4438	541	3897	86.75	2.513	.0317	2.37×10^6
-12	23718	541	23177	86.75	2.513	.0317	1.41×10^7
0	38754	541	38213	86.75	2.513	.0317	2.33×10^7
+12	18788	541	18237	86.75	2.513	.0317	1.11×10^7
+30	1806	541	1265	86.75	2.513	.0317	7.71×10^5
+54	0	541					
+72	0	541					
			SHIELD CHANNEL #7				
-29	1639	504	1135	90.0	2.558	.0310	7.05×10^5
-12	5345	504	4841	90.0	2.558	.0310	3.00×10^6
0	7147	504	6643	90.0	2.558	.0310	4.11×10^6
+12	4120	504	3616	90.0	2.558	.0310	2.24×10^6
+30	848	504	344	90.0	2.558	.0310	2.13×10^5
+54	0	504					
+72	0	504					
			Run 4 - 64 Hr at 10.8 Mw (Normalized to Run 2 Channel 4)*				
			SHIELD CHANNEL #3				
-12	452533	504	452029	89.0	2.546	.0311	2.79×10^8
+12	601451	504	600947	89.0	2.546	.0311	3.71×10^8
+54	3332	504	2828	89.0	2.546	.0311	1.75×10^6
+72	0						
			SHIELD CHANNEL #6				
-12	54161	504	53657	89.0	2.546	.0311	3.30×10^7
+12	36650	504	36146	89.0	2.546	.0311	2.23×10^7
+54	nil	504					
+72	nil	504					

* These gold foil measurements were intended for comparison with fast flux measurements (sulfur pellets), but data was inconsistent with each other and with other runs.

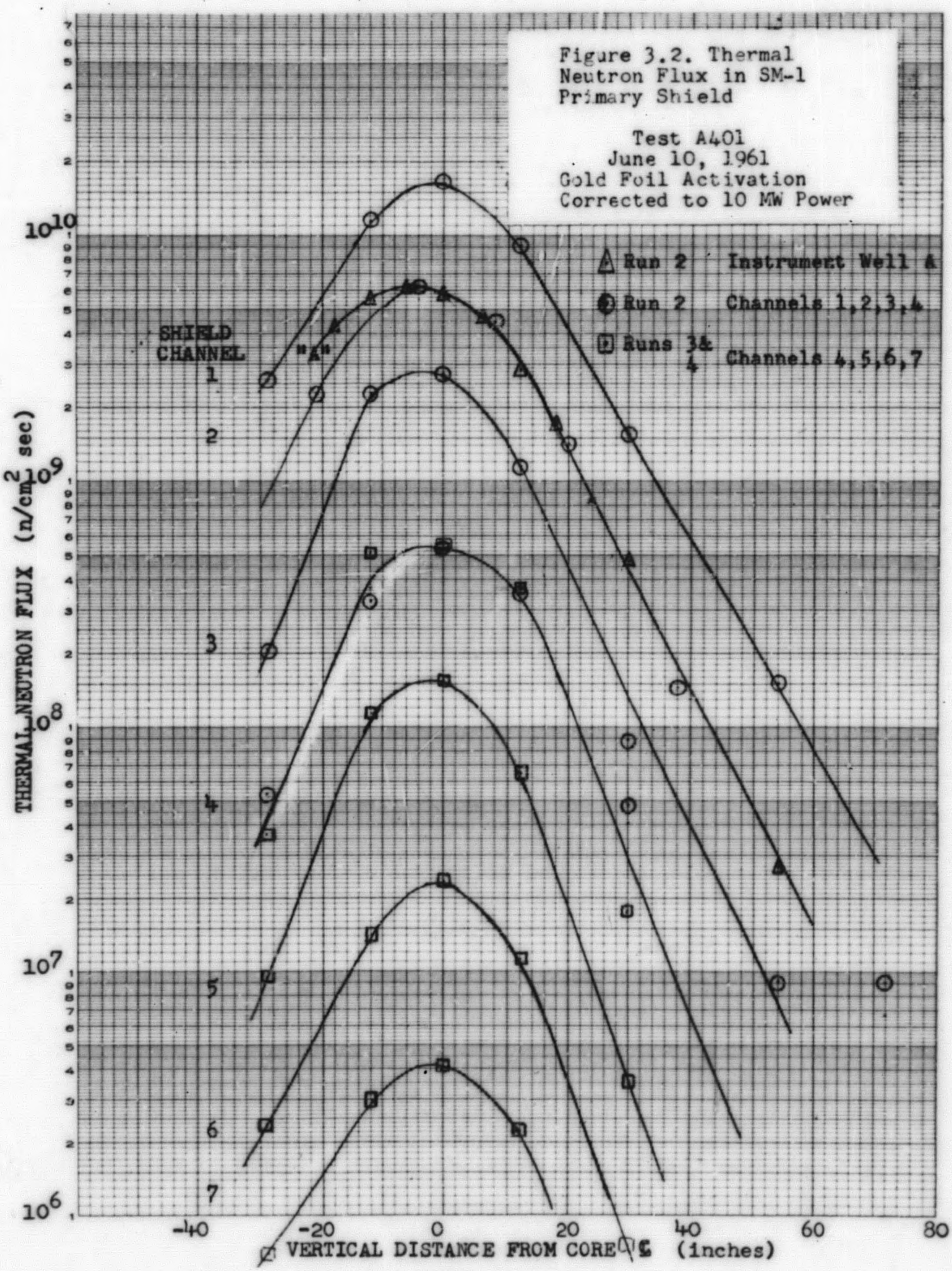

Figure 3.2. Thermal Neutron Flux in SM-1 Primary Shield

Test A401
June 10, 1961
Gold Foil Activation
Corrected to 10 MW Power

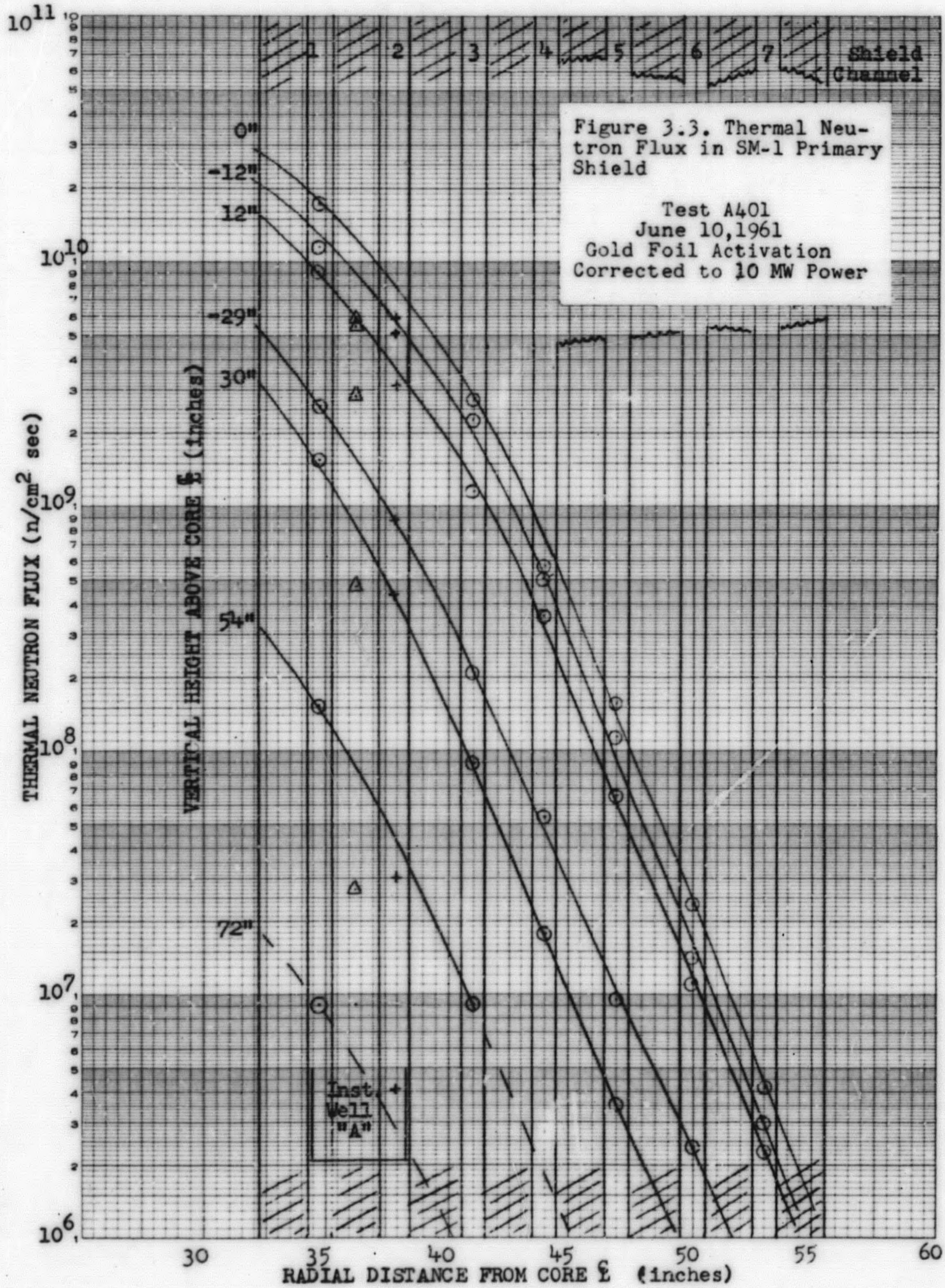

Figure 3.3. Thermal Neutron Flux in SM-1 Primary Shield
Test A401
June 10,1961
Gold Foil Activation
Corrected to 10 MW Power
Shield Channel
1
2
3
4
5
6
7
0"
-12"
12"
-29"
30"
54"
72"
Inst. Well "A"
THERMAL NEUTRON FLUX (n/cm2 sec)
VERTICAL HEIGHT ABOVE CORE ℄ (inches)
RADIAL DISTANCE FROM CORE ℄ (inches)
30
35
40
45
50
55
60

TABLE 3.3
THERMAL NEUTRON FLUX THROUGH INNER SHIELD TANK AXIS

Run 3 - 60 Hr at 237 Kw
Run 4 - 64 Hr at 10.8 Mw (Normalized to Run 2, Channel 4)

Vertical Position From Core Centerline, In.	Activity, Cpm	Background, Cpm	Act. Net, Cpm	Decay Time, Hr	$e^{\lambda t}$	Foil Mass, Gm	Neutron Flux at 10 Mw, N/cm^2 sec
72	3690	504	3186	89.5	2.550	0.0310	1.97×10^6
83	598	504	94	89.5	2.550	0.0310	5.82×10^4
95	589	504	85				5.23×10^4
107		504					
119		504					
143		504					
155		504					

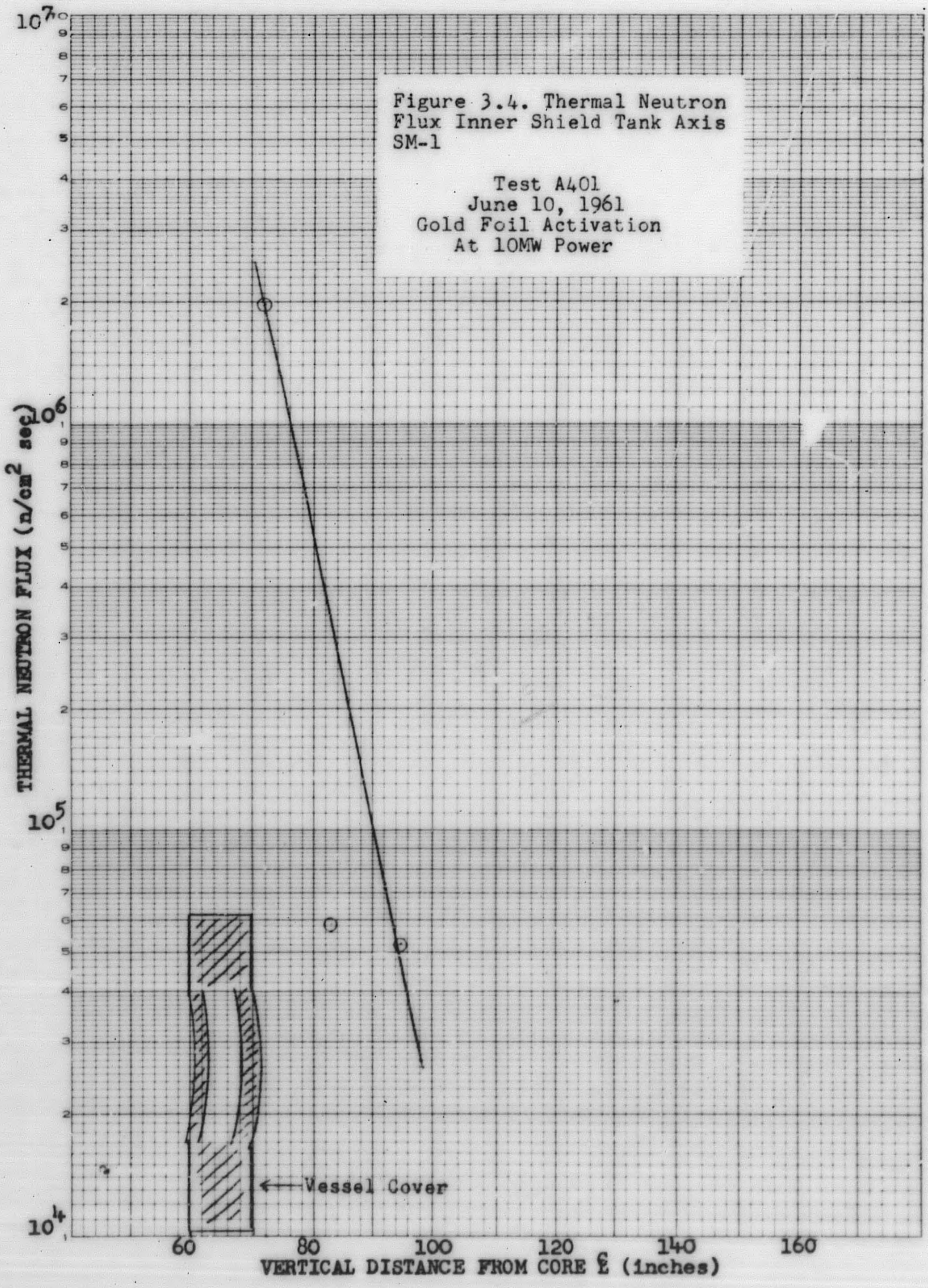

Figure 3.4. Thermal Neutron Flux Inner Shield Tank Axis SM-1
Test A401
June 10, 1961
Gold Foil Activation
At 10MW Power
THERMAL NEUTRON FLUX (n/cm2 sec)
10^7
10^6
10^5
10^4
60
80
100
120
140
160
VERTICAL DISTANCE FROM CORE ℄ (inches)
Vessel Cover

TABLE 3.4
THERMAL NEUTRON FLUX OUTSIDE SHIELD TANK

Run 4 - 64 Hr at 10.8 Mw (Normalized to Run 2, Channel 4)

Vertical Position From Core Centerline, in.	Activity, Cpm	Background, Cpm	Act. Net, Cpm	Decay Time, Hr	$e^{\lambda t}$	Foil Mass, Gm	Neutron Flux at 10 Mw, N/cm^2 sec
-29	1792	541	1251	86.75	2.513	0.0311	7.63×10^5
-12	1279	541	738	86.75	2.513	0.0311	4.50×10^5
0	1047	541	506	86.75	2.513	0.0311	3.09×10^5
+12	804	541	263	86.75	2.513	0.0311	1.60×10^5
+30	697	541	156	86.75	2.513	0.0311	9.5×10^4
+54	552	541	11	86.75	2.513	0.0311	6.78×10^3
+72	561	541	20	86.75	2.513	0.0311	1.21×10^4

TABLE 3.5
THERMAL NEUTRON FLUX IN ROD DRIVE PIT

Run 4 - 64 Hr at 10.8 Mw (Normalized to Run 2, Channel 4)

Position	Activity, Cpm	Background, Cpm	Act, Net, Cpm	Decay Time, Hr	$e^{\lambda t}$	Foil Mass, Gm	Neutron Flux at 10 Mw, N/cm^2 sec
WOOD SHIELD 4 In. x 4 In.							
Left Top (TP 19)	396954	541	396413	86.50	2.51	.0317	2.41×10^8
Left Mid	234553	541	234012	86.50	2.51	.0317	1.42×10^8
Left Lower	73534	541	72993	86.50	2.51	.0317	4.44×10^7
Bottom Mid	76925	541	76384	86.50	2.51	.0317	4.63×10^7
Bottom Right	65118	541	64577	86.50	2.51	.0317	3.92×10^7
WATER BOX SHIELD FACE							
Drive #3	390936	541	390395	86.50	2.51	.0317	2.37×10^8
Drive #5	180746	541	180205	86.50	2.51	.0317	1.09×10^8
Drive #7	115185	541	114644	86.50	2.51	.0317	6.96×10^7
Bottom Right (TP 20)	95811	541	95270	86.50	2.51	.0317	5.78×10^7

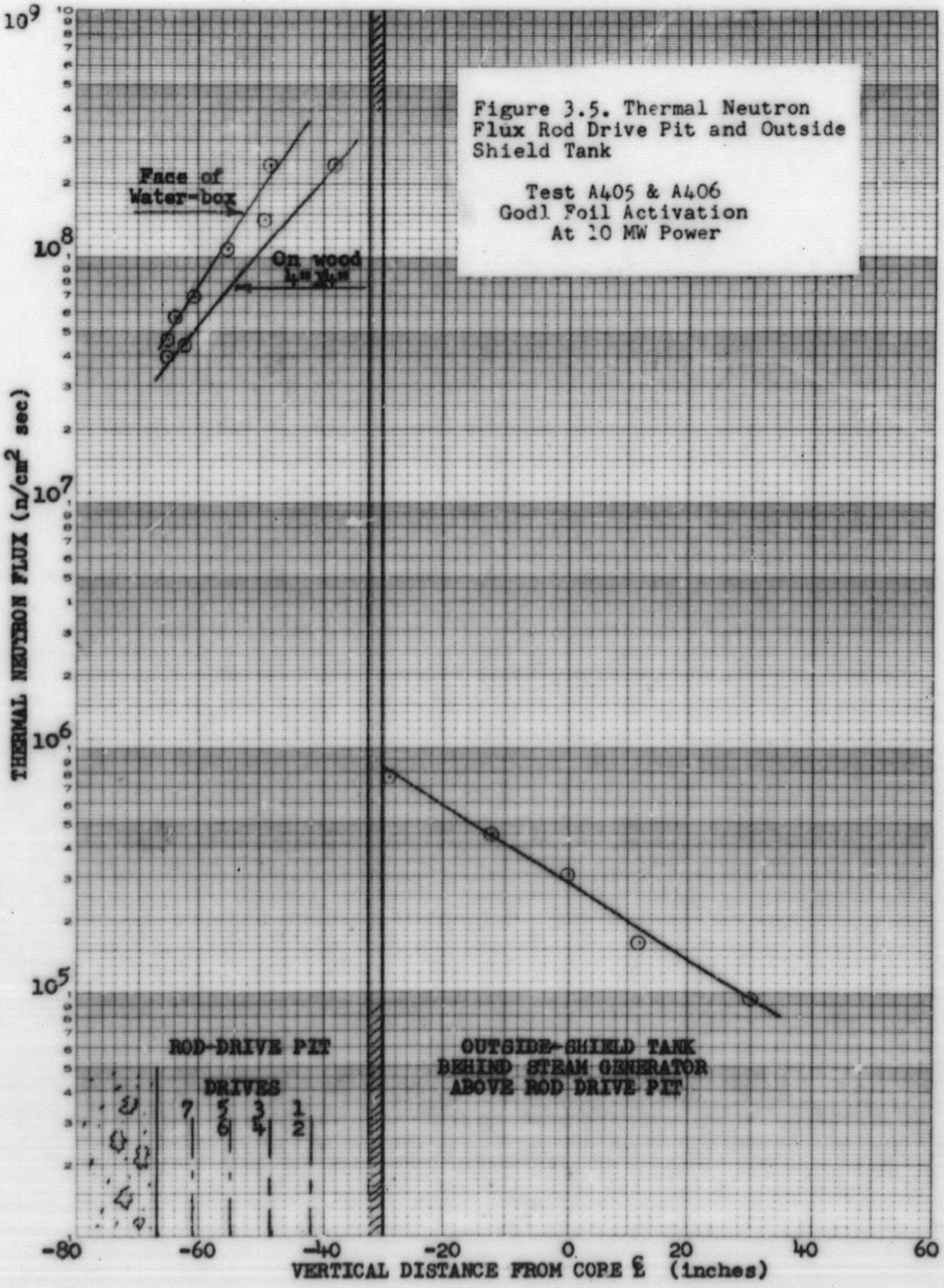

Figure 3.5. Thermal Neutron Flux Rod Drive Pit and Outside Shield Tank

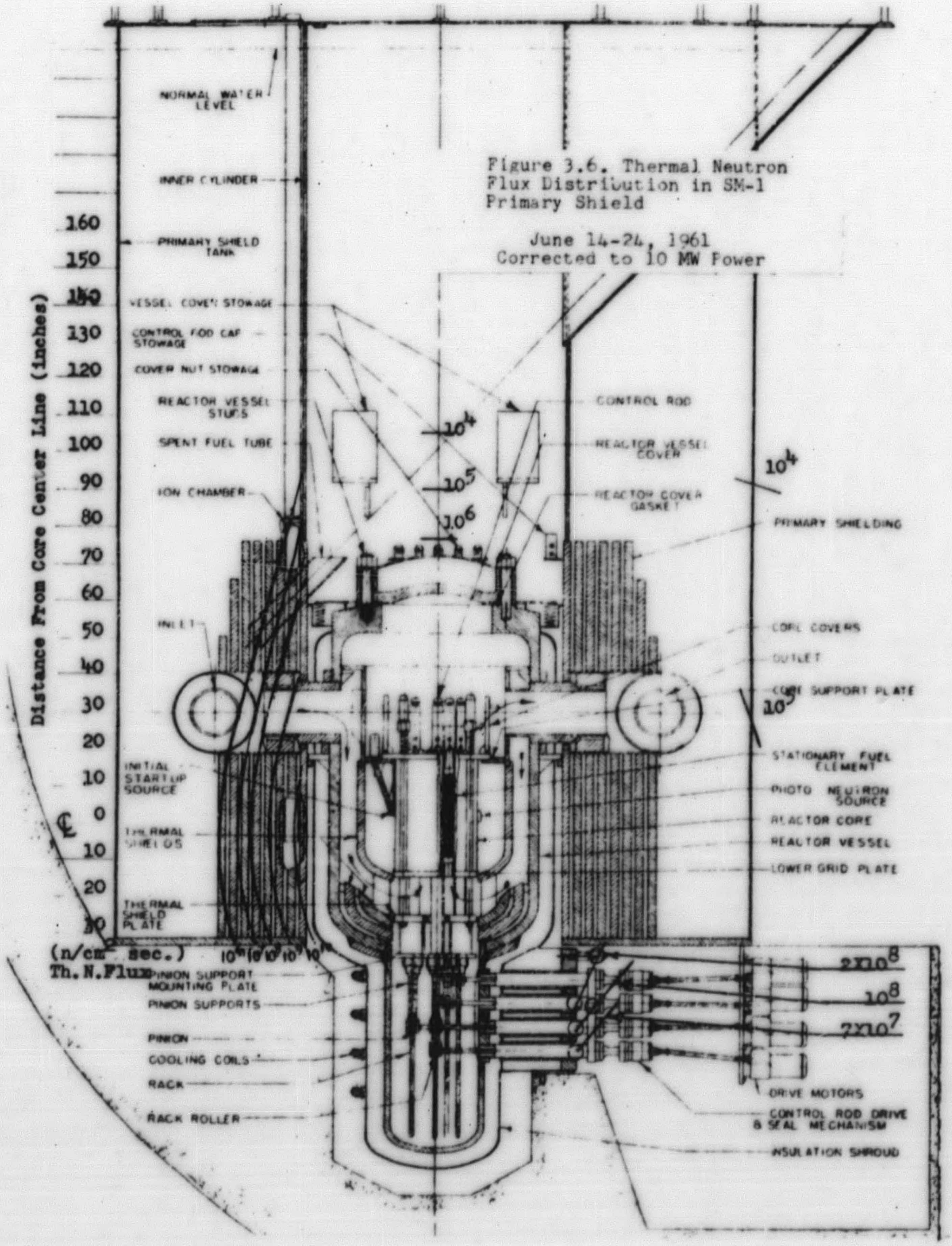

Figure 3.6. Thermal Neutron Flux Distribution in SM-1 Primary Shield

June 14-24, 1961
Corrected to 10 MW Power

TABLE 3.6

FAST NEUTRON FLUX IN SHIELD RINGS

Run 4 - 64 Hr at 10.8 Mw

Vertical Position From Core Centerline, In.	Activity, Cpm	Background, Cpm	Sulfur Mass, Gm	Neutron Flux at 10.8 Mw, N/cm^2 sec
		SHIELD CHANNEL #1		
-29	56731/2	17	2.8689	3.77×10^8
-12	395976/2	17	2.8689	2.64×10^9
0	453925/2	17	2.8689	3.46×10^9
+12	312455/2	17	2.8732	2.08×10^9
+30	24605/2	17	2.8732	1.63×10^8
		SHIELD CHANNEL #3		
-29	5049/2	17	2.8739	3.27×10^7
0	62966/2	17	2.8739	4.19×10^8
+30	6925/5	17	2.8750	1.84×10^7
		SHIELD CHANNEL #6		
-29	1303/10	17	2.8750	1.71×10^6
0	9234/10	17	2.8656	1.23×10^7
+30	917/15	17	2.8656	8.01×10^5

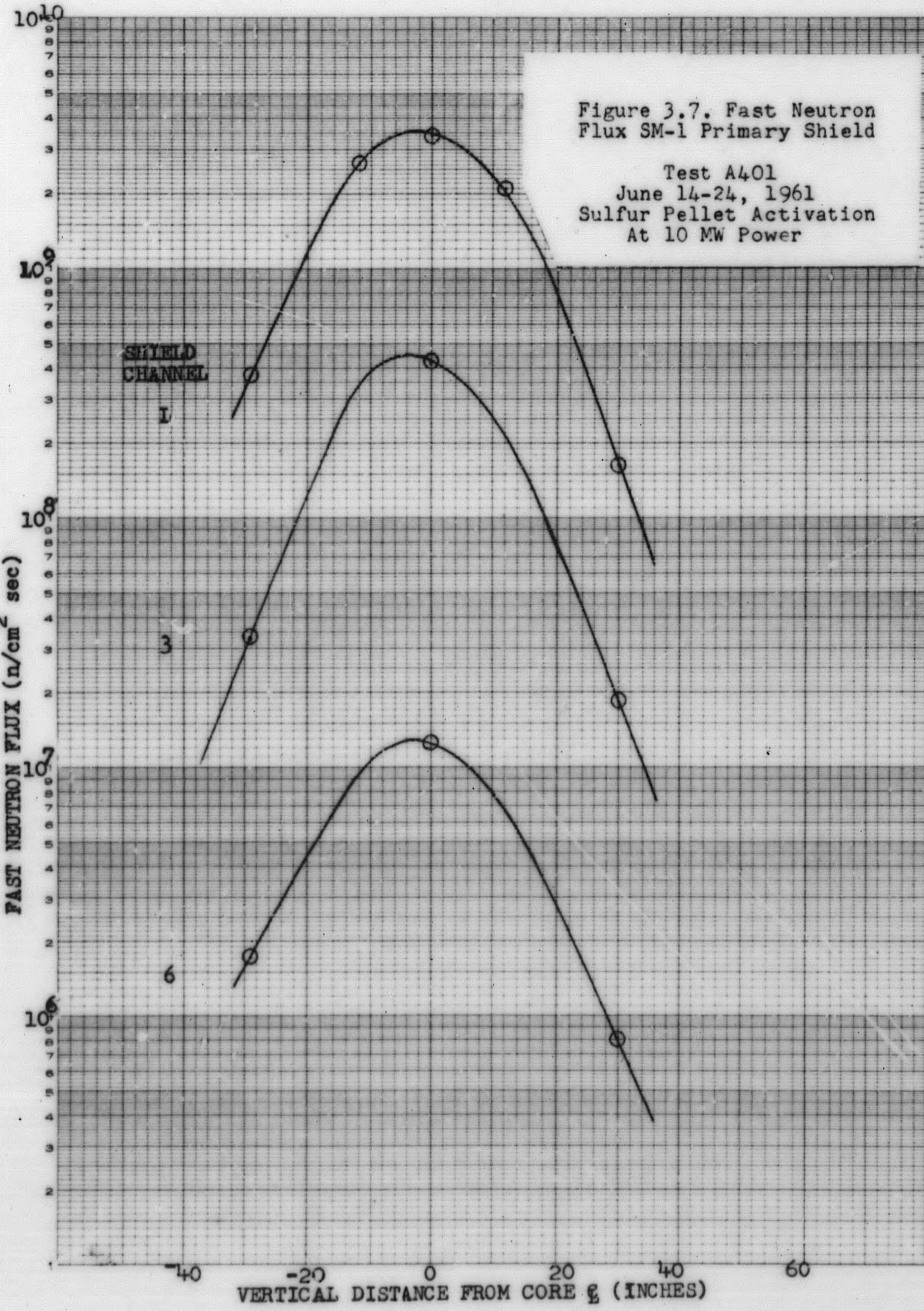

Figure 3.7. Fast Neutron Flux SM-1 Primary Shield

Test A401
June 14-24, 1961
Sulfur Pellet Activation
At 10 MW Power

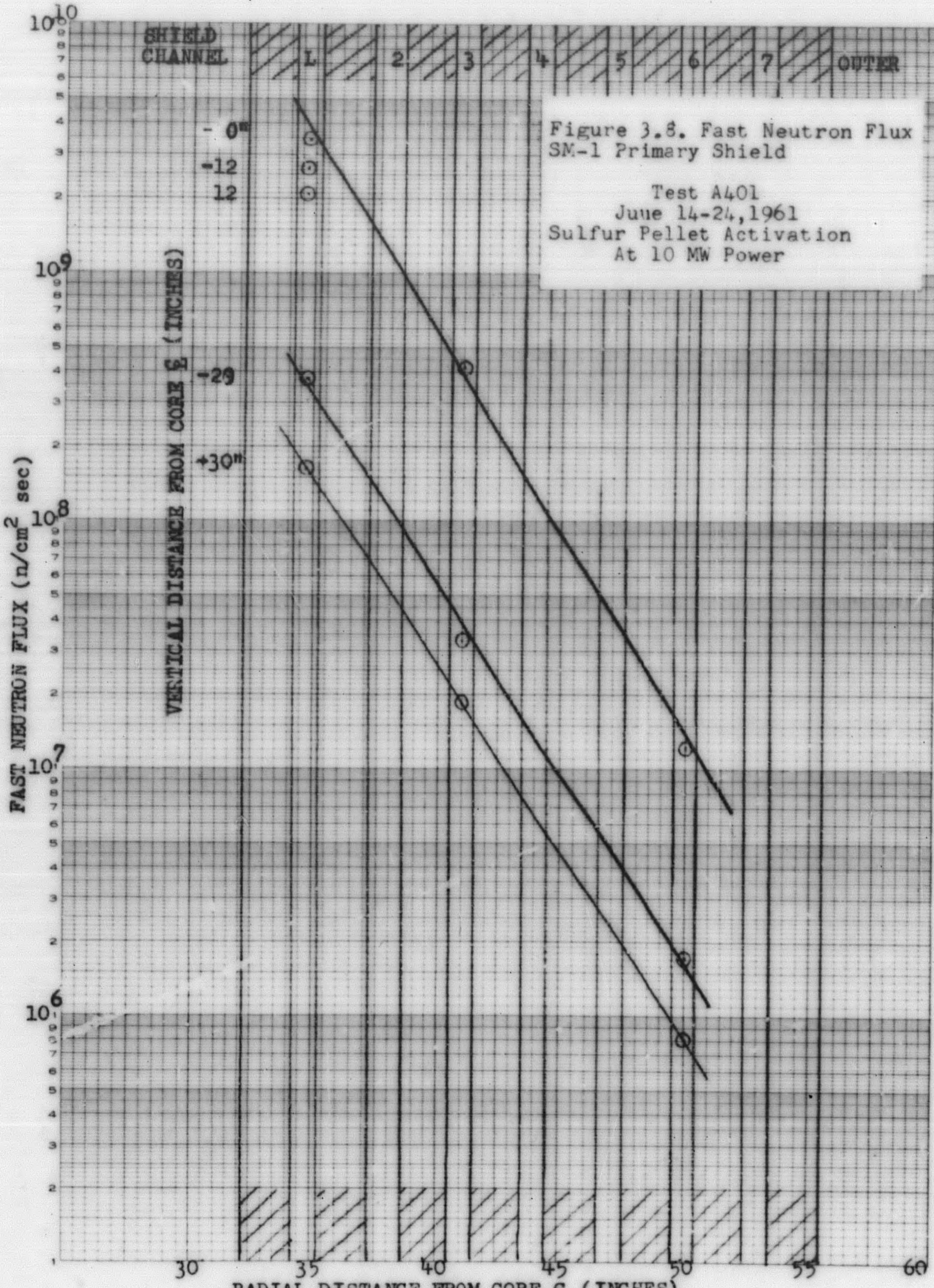

Figure 3.8. Fast Neutron Flux
SM-1 Primary Shield

Test A401
June 14-24, 1961
Sulfur Pellet Activation
At 10 MW Power

4.0 GAMMA RADIATION MEASUREMENTS

Previous measurements of the gamma radiation in the reactor shield were taken only at the elevation of the reactor core centerline, as reported in APAE-35.[1] In the present series of tests, film badges were mounted on aluminum strips at seven different elevations and inserted in the channels of the outer shield tank and axially above the reactor vessel in the inner-shield tank, in order to obtain a better indication of the dose rate distribution throughout the primary shield structure.

Radiation surveys were made in the rod drive pit on several occasions before, during and after the removal of SM-1 Core I. The dose rates showed no significant change, indicating that the core was not the source of this radiation.

Dose rate measurements were made 6 in. above the surface of the water in the spent fuel pit during handling and transfer of fuel elements.

4.1 GAMMA FLUX IN INSTRUMENT WELL A - TEST A404

Before beginning reactor operations on SM-1 Core II, a set of film badges mounted on an aluminum strip was inserted in instrument well A and irradiated for 8 hr with the reactor shut down (Run 0). The position of the films and the measured dose rate are shown in Table 4.1 and Fig. 4.1. Using the radial attenuation of dose rate from previous measurements reported in APAE-35,[1] background dose rates were determined out to shield channel 4, where they became negligible. This background correction, as shown in Table 4.1, was applied for the total exposure time in the reactor for succeeding runs; and it accounted for as much as 20% of the total dose for a few points.

For Run 1 at 11.1 kw for 20 min, another strip of films was exposed in well A. The dose rates are shown in Table 4.2 and Fig. 4.1. These were converted to a power level of 10 Mw and included with measurements in the shield channels, as reported in the following section.

4.2 GAMMA FLUX IN PRIMARY SHIELD - TEST A402

Film badge packets sealed in polyethylene bags were mounted on aluminum strips and inserted in the water channels between the shield rings in the outer shield tank. They were placed at seven elevations with respect to the core centerline: at -29 in., -12 in., 0 in., 12 in., 30 in., 54 in. and 72 in.

TABLE 4.1

GAMMA BACKGROUND IN INSTRUMENT WELL A

Run 0 - 8 Hr Exposure Before Core Startup

Vertical Position From Core Centerline, In.	Total Dose, R	Exposure Time, Hr	Dose Rate, R/Hr	Calculated Shield Ring Dose Rate, R/Hr			
				#1	#2	#3	#4
-29	-	-	-	3.4	0.85	0.23	0.062
-12	130.0	8.0	16.3	22.1	5.57	1.49	0.407
0	150.0	8.0	18.8	25.5	6.42	1.71	0.470
+12	71.0	8.0	8.9	12.0	3.04	0.81	0.222
+30	11.0	8.0	1.4	1.9	0.48	0.13	0.035
+54	1.35	8.0	0.17	0.23	0.058	0.015	0.0042
+72	0.075	8.0	0.009	0.012	0.003	0.008	0.00002

TABLE 4.2
GAMMA RADIATION IN INSTRUMENT WELL A

Run 1 - 20 Min at 11.1 Kw

Vertical Position From Core Centerline, In.	Total Dose, R	Background Exposure Time, Hr	Total Background Dose, R	Total Operating Dose, R	Exposure Time At Power, Hr	Operating Dose Rate, R/Hr	Average Power Level, Kw	Dose Rate at 10 Mw, R/Hr
-12	940.0	2.75	45.0	895.0	.333	2700	11.1	2,420,000
0	340.0	2.75	52.0	288.0	.333	865	11.1	780,000
+12	125.0	2.75	24.5	100.5	.333	300	11.1	271,000
+30	17.0	2.75	3.8	13.2	.333	40	11.1	35,700
+54	1.30	2.75	0.47	0.83	.333	2.5	11.1	2,240
+72	0.080	2.75	0.025	0.055	.333	0.165	11.1	149

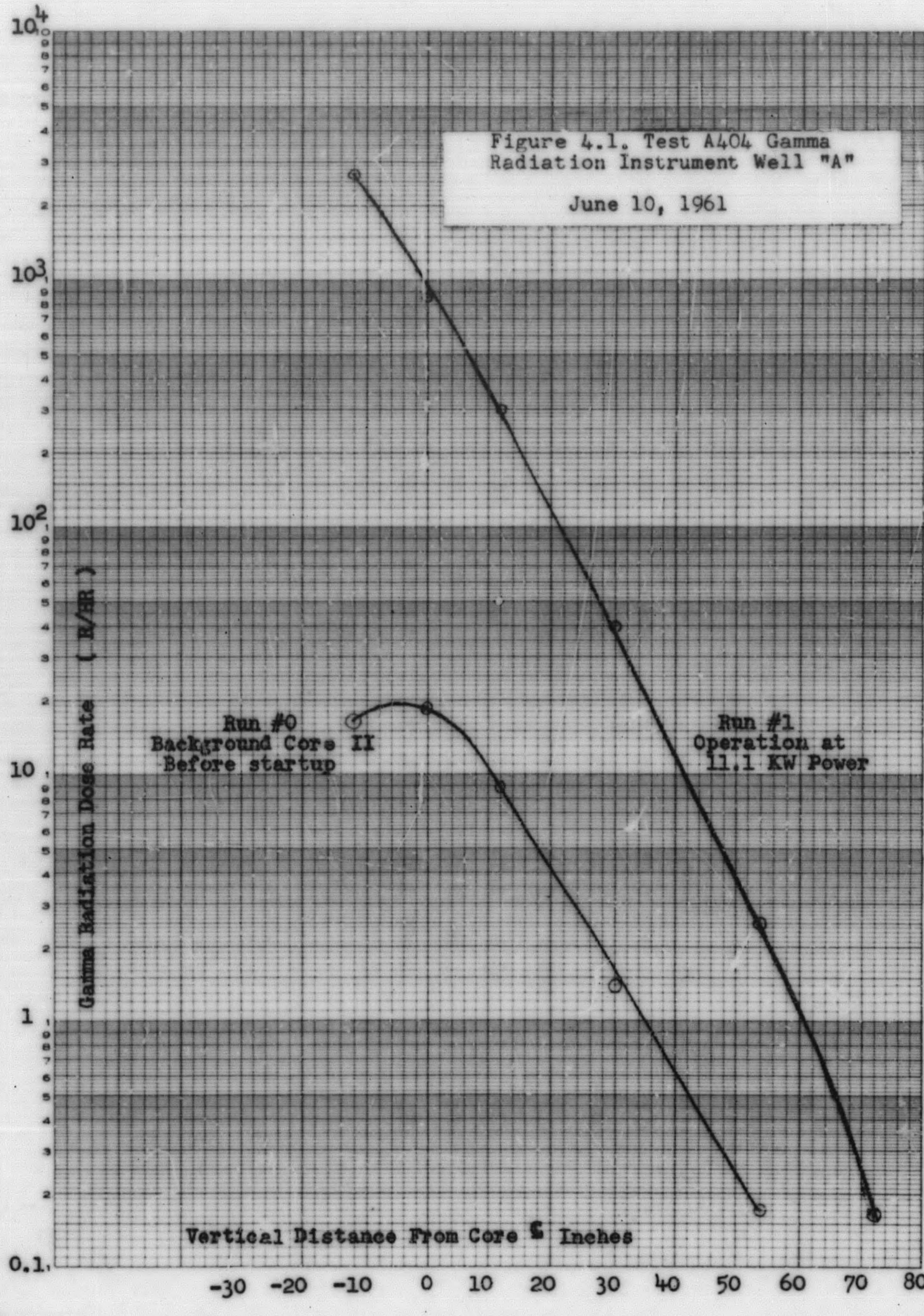
Figure 4.1. Test A404 Gamma
Radiation Instrument Well "A"
June 10, 1961
Gamma Radiation Dose Rate (R/HR)
10^4
10^3
10^2
10
1
0.1
Run #0
Background Core II
Before startup
Run #1
Operation at
11.1 KW Power
Vertical Distance From Core ℄ Inches
-30
-20
-10
0
10
20
30
40
50
60
70
80

Strips of film were inserted in shield channels 1, 2, 3 and 4 and irradiated at 11.1 kw for 20 min, Run 1. The dose rates, converted to 10 Mw power level, are shown in Table 4.3 and Figs. 4.2 and 4.3.

For Run #2 at 26.6 kw average for 8 hr, strips were inserted in shield channels 4, 5, 6 and 7. The results are shown in Table 4.3 and Figs. 4.2 and 4.3.

For Run #3 at 237 kw average for 60 hr, one strip of films was inserted close to the axis of the inner-shield tank above the reactor vessel cover. The positions and dose rates are shown in Table 4.4 and Fig. 4.4. Unfortunately, all of the films except two were damaged by water penetrating the polyethylene bags and were not readable.

These gamma radiation dose rates were all combined in Fig. 4.5 which shows contours of equal dose rate values through the primary shield structure.

The present gamma radiation values are higher than those previously shown in APAE-35, Fig. 2.4 by about a factor of three. The attenuation slope is similar to that obtained previously. The dose rates from this set of tests are higher than those shown in the calculated curve reported in APAE-35 by a factor of 1.5. This discrepancy is considered to be within the limits of error of both measurements and calculation.

4.3 DOSE RATE AT VAPOR-CONTAINER SHIELD WALL

Two film packs were mounted inside the vapor-container wall; and two others, outside at points near the electrical penetrations, about 15 ft above the reactor core centerline. They were exposed for 10 days during which time the reactor operated for Run 4 at 10.8 Mw for 64 hr.

The operating dose rate inside the vapor container wall was measured as 0.54 and 0.56 R/hr. Outside the wall the dose rate was not detectable by the film badges there. These results are in reasonable agreement with those reported in APAE-18[4] and APAE-35.[1] Figure 4.6 shows the measured dose rate and a calculated attenuation curve.

4.4 RADIATION SURVEYS IN ROD DRIVE PIT - TEST A407

The dose rate in the rod drive pit was measured on several occasions with portable radiation survey meter. Three surveys were made in April 1961 after the final shutdown of SM-1 Core I but before the core was removed. After the core was unloaded, the survey was repeated on May 4, and again on May 16, 1961 after the flushing and cleanup of crud in the bottom of the reactor vessel. Another survey was made on June 24, 1961 after completion of a two-day run at full power on Core II. This survey was repeated on October 20, 1961, four days after shutdown from four months of routine training operations on Core II.

There were no significant differences between any of these radiation surveys. Small variations could be attributed to location of the probe of the survey instrument. The measurements were also in reasonable agreement with previous surveys made from July to September 1959 and reported in APAE Memo 237.[3] These checks indicate there is no evidence of any serious increase in the radiation problem.

The highest readings were obtained at the water box shield face adjacent to the rod drive penetrations; and were made in the range of 10 to 20 R/hr. The leakage slots at the left side and bottom of the water box showed dose rates ranging from 1 to 3 R/hr. Two test points established at the upper left and lower right corners of the rod drive mounting plate usually read about 0.8 and 1.5 R/hr. The background in the accessible working space near the rod drive cables was 0.2 to 0.3 R/hr.

As mentioned in Section 3.2, the neutron flux in the rod drive pit was unexpectedly high. Activation of the steel face plate of the water box and the rod drive components may account for much of the radiation source. The reactor core is evidently not the source of this radiation. The water box is assumed to be empty, as the attempts to keep it full were discontinued a year or two before these tests.

TABLE 4.3
GAMMA RADIATION IN SHIELD RINGS

Run 1 - 20 Min at 11.1 Kw

Vertical Position From Core Centerline, In.	Total Dose, R	Background Exposure Time, Hr	Total Background Dose, R	Total Operating Dose, R	Exposure Time at Power, Hr	Operating Dose Rate, R/Hr	Average Power Level, Kw	Dose Rate 10 Mw, R/Hr
				SHIELD CHANNEL #1				
-29	25.5	2.75	9.35	16.15	0.333	48.5	11.1	43,700
-12	290.0	2.75	60.7	229.30	0.333	688.0	11.1	619,000
0	460.0	2.75	70.0	390.0	0.333	1170.0	11.1	1,054,000
+12	230.0	2.75	33.0	197.0	0.333	590.0	11.1	532,000
+30	Damaged	2.75						
+54	2.15	2.75	0.630	1.520	0.333	4.56	11.1	4,100
+72	0.11	2.75	0.033	0.077	0.333	0.23	11.1	207
				SHIELD CHANNEL #2				
-29	22.0	2.75	2.34	19.66	0.333	59.0	11.1	53,200
-12	86.5	2.75	15.30	71.2	0.333	213.5	11.1	193,000
0	115.0	2.75	17.7	97.3	0.333	292.0	11.1	263,000
+12	66.5	2.75	8.36	58.14	0.333	175.0	11.1	157,800
+30	7.2	2.75	1.32	5.88	0.333	17.7	11.1	15,950
+54	0.56	2.75	0.160	0.40	0.333	1.20	11.1	1,080
+72	0.015	2.75	0.008	0.007	0.333	0.021	11.1	19
				SHIELD CHANNEL #3				
-29	5.45	2.75	0.630	4.82	0.333	14.45	11.1	13,050
-12	22.0	2.75	4.10	17.9	0.333	53.7	11.1	48,400
0	31.0	2.75	4.70	26.3	0.333	79.0	11.1	71,200
+12	Damaged	2.75	-	-	0.333	-	11.1	-
+30	2.475	2.75	0.355	2.120	0.333	6.36	11.1	5,730
+54	0.115	2.75	0.041	0.074	0.333	0.222	11.1	200
+72	0.015	2.75	0.002	0.013	0.333	0.039	11.1	35
				SHIELD CHANNEL #4				
-29	Damaged	2.75	-	-	0.333	-	11.1	-
-12	Damaged	2.75	-	-	0.333	-	11.1	-
0	8.5	2.75	1.3	7.2	0.333	21.6	11.1	19,500
+12	5.6	2.75	0.61	5.0	0.333	15.0	11.1	13,500
+30	0.640	2.75	0.096	0.544	0.333	1.63	11.1	1,470
+54	0.075	2.75	0.011	0.064	0.333	0.190	11.1	171
+72	0.020	2.75	-	0.020	0.333	0.060	11.1	54

TABLE 4.3 (CONT'D)

Run 2 - 8 Hr at 26.6 Kw Average Power

Vertical Position From Core Centerline, In.	Total Dose, R	Background Exposure Time, Hr	Total Background Dose, R	Total Operating Dose, R	Exposure Time at Power, Hr	Operating Dose Rate, R/Hr	Average Power Level, Kw	Dose Rate at 10 Mw, R/Hr
			SHIELD CHANNEL #4					
-29	84.25	12.5	0.78	83.47	8.0	10.4	26.6	3,910
-12	300.00	12.5	5.10	294.9	8.0	36.9	26.6	13,900
0	410.00	12.5	5.90	404.1	8.0	50.6	26.6	19,100
+12	240.00	12.5	2.80	237.2	8.0	29.7	26.6	11,200
+30	28.75	12.5	0.440	28.31	8.0	3.54	26.6	1,330
+54	1.40	12.5	0.052	1.35	8.0	0.169	26.6	64
+72	0.045	12.5	0.00025	0.045	8.0	0.0056	26.6	2.1
			SHIELD CHANNEL #5					
-29	23.25	12.5	Negl.	23.25	8.0	2.91	26.6	1,090
-12	112.50	12.5	Negl.	112.50	8.0	14.10	26.6	5,300
0	112.50	12.5	Negl.	112.50	8.0	14.10	26.6	5,300
+12	50.00	12.5	Negl.	50.00	8.0	6.25	26.6	2,350
+30	6.15	12.5	Negl.	6.15	8.0	0.77	26.6	290
+54	0.23	12.5	Negl.	0.23	8.0	0.0288	26.6	11
+72	0.06	12.5	Negl.	0.06	8.0	0.0075	26.6	3
			SHIELD CHANNEL #6					
-29	19.0	125	Negl.	19.0	8.0	2.38	26.6	895
-12	30.0	125	Negl.	30.0	8.0	3.75	26.6	1,410
0	34.25	125	Negl.	34.25	8.0	4.28	26.6	1,610
+12	Damaged	125	Negl.	-	8.0	-	26.6	-
+30	2.90	125	Negl.	2.90	8.0	0.363	26.6	136
+54	0.130	125	Negl.	0.130	8.0	0.0163	26.6	6.1
+72	-	125	Negl.	-	8.0	-	26.6	-
			OUTSIDE OUTER SHIELD					
-29	2.050	125	Negl.	2.050	8.0	0.256	26.6	96
-12	2.350	125	Negl.	2.350	8.0	0.294	26.6	110
0	2.550	125	Negl.	2.550	8.0	0.319	26.6	120
+12	1.200	125	Negl.	1.200	8.0	0.150	26.6	56
+30	0.160	125	Negl.	0.160	8.0	0.020	26.6	7.5
+54	0.030	125	Negl.	0.030	8.0	0.0037	26.6	1.4
+72	0.025	125	Negl.	0.025	8.0	0.0031	26.6	1.2

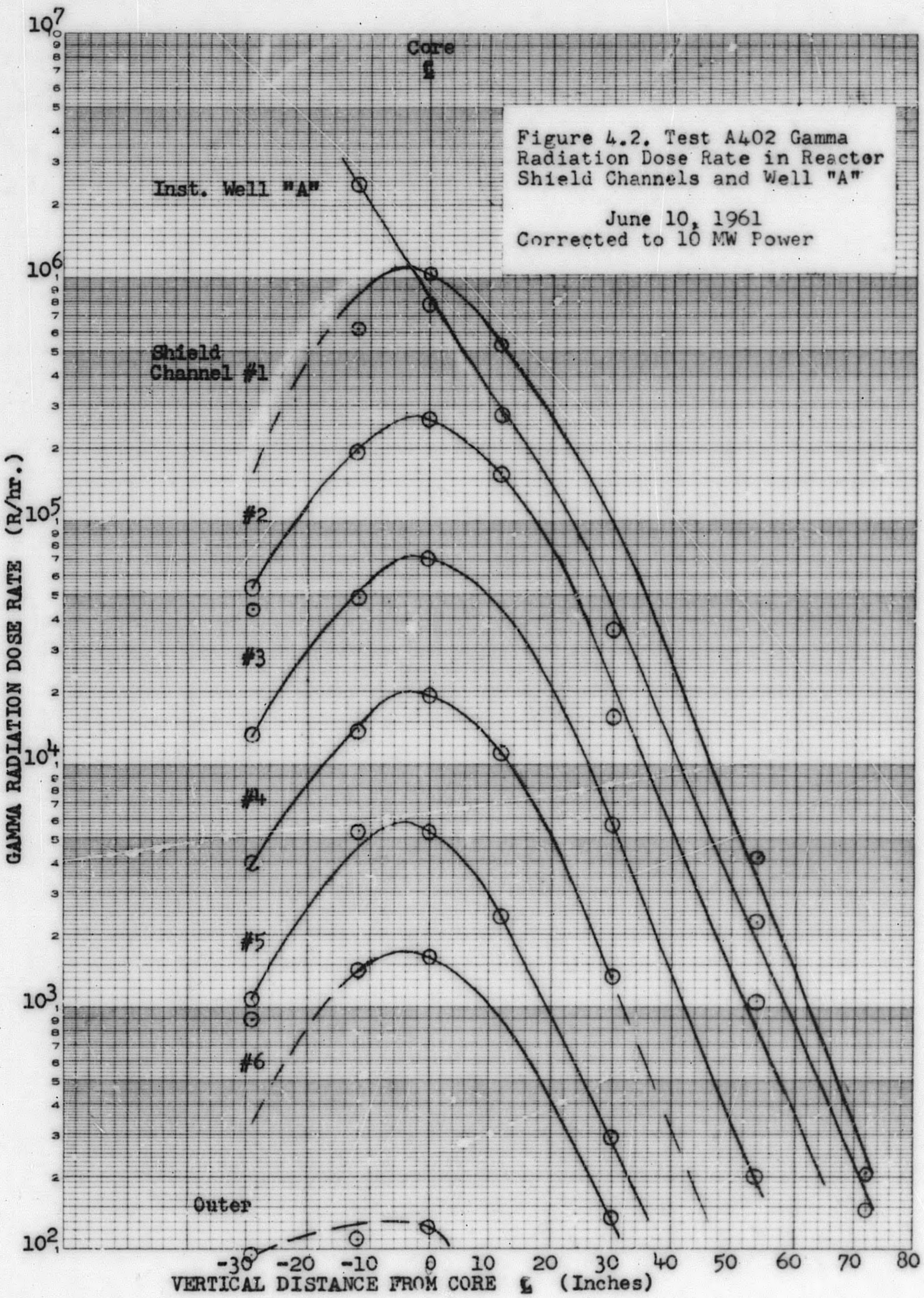

Figure 4.2. Test A402 Gamma Radiation Dose Rate in Reactor Shield Channels and Well "A"

June 10, 1961
Corrected to 10 MW Power

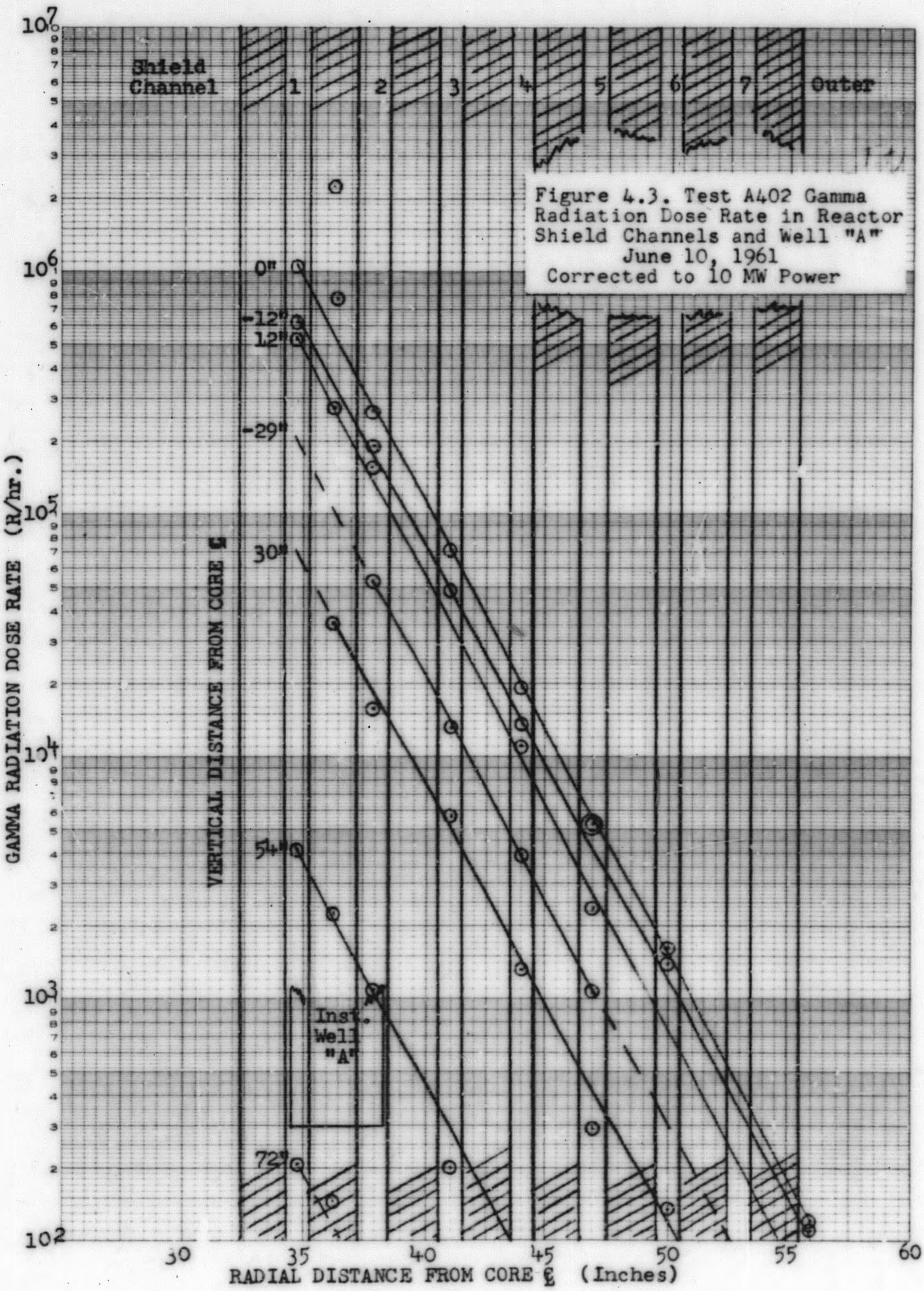

Figure 4.3. Test A402 Gamma Radiation Dose Rate in Reactor Shield Channels and Well "A" June 10, 1961 Corrected to 10 MW Power

TABLE 4.4
GAMMA RADIATION IN INNER SHIELD TANK AXIS

Run 3 - 60 Hr at 237 Kw Average Power

Vertical Position From Core Centerline, In.	Total Dose, R	Background Exposure Time, Hr	Total Background Dose, R	Total Operating Dose, R	Exposure Time at Power, Hr	Operating Dose Rate, R/Hr	Average Power Level, Kw	Dose Rate at 10 Mw, R/Hr
72	Damaged	93.0	Negl.	-	60.0	-	237.0	-
83	Damaged	93.0	Negl.	-	60.0	-	237.0	-
95	1475.0	93.0	Negl.	1475.0	60.0	24.6	237.0	1,040
107	Damaged	93.0	Negl.	-	60.0	-	237.0	-
119	Damaged	93.0	Negl.	-	60.0	-	237.0	-
131	Damaged	93.0	Negl.	-	60.0	-	237.0	-
155	32.0	93.0	Negl.	32.0	60.0	0.534	237.0	22.5

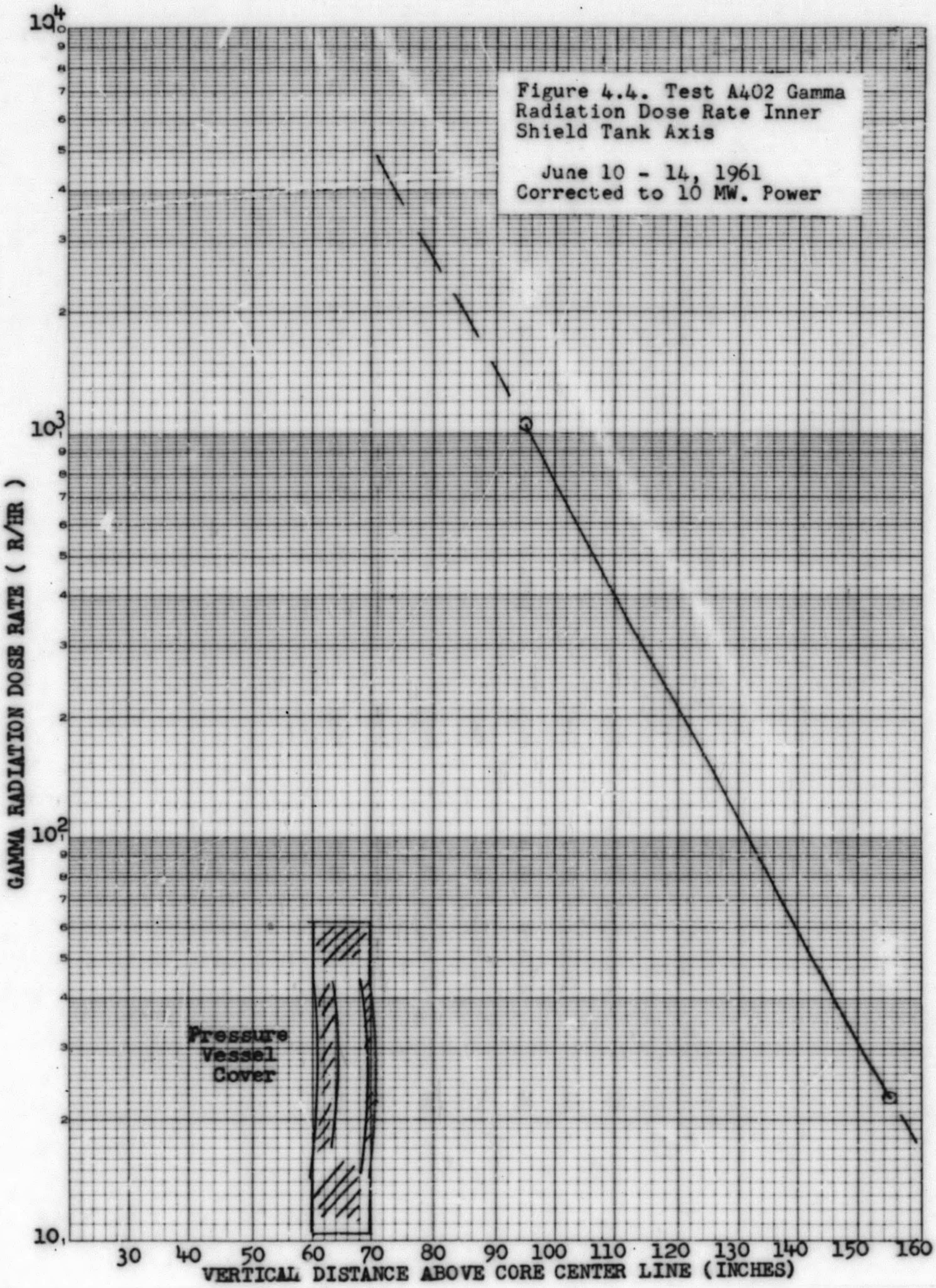

Figure 4.4. Test A402 Gamma Radiation Dose Rate Inner Shield Tank Axis
June 10 - 14, 1961
Corrected to 10 MW. Power
GAMMA RADIATION DOSE RATE (R/HR)
10^4
10^3
10^2
10
30
40
50
60
70
80
90
100
110
120
130
140
150
160
VERTICAL DISTANCE ABOVE CORE CENTER LINE (INCHES)
Pressure Vessel Cover

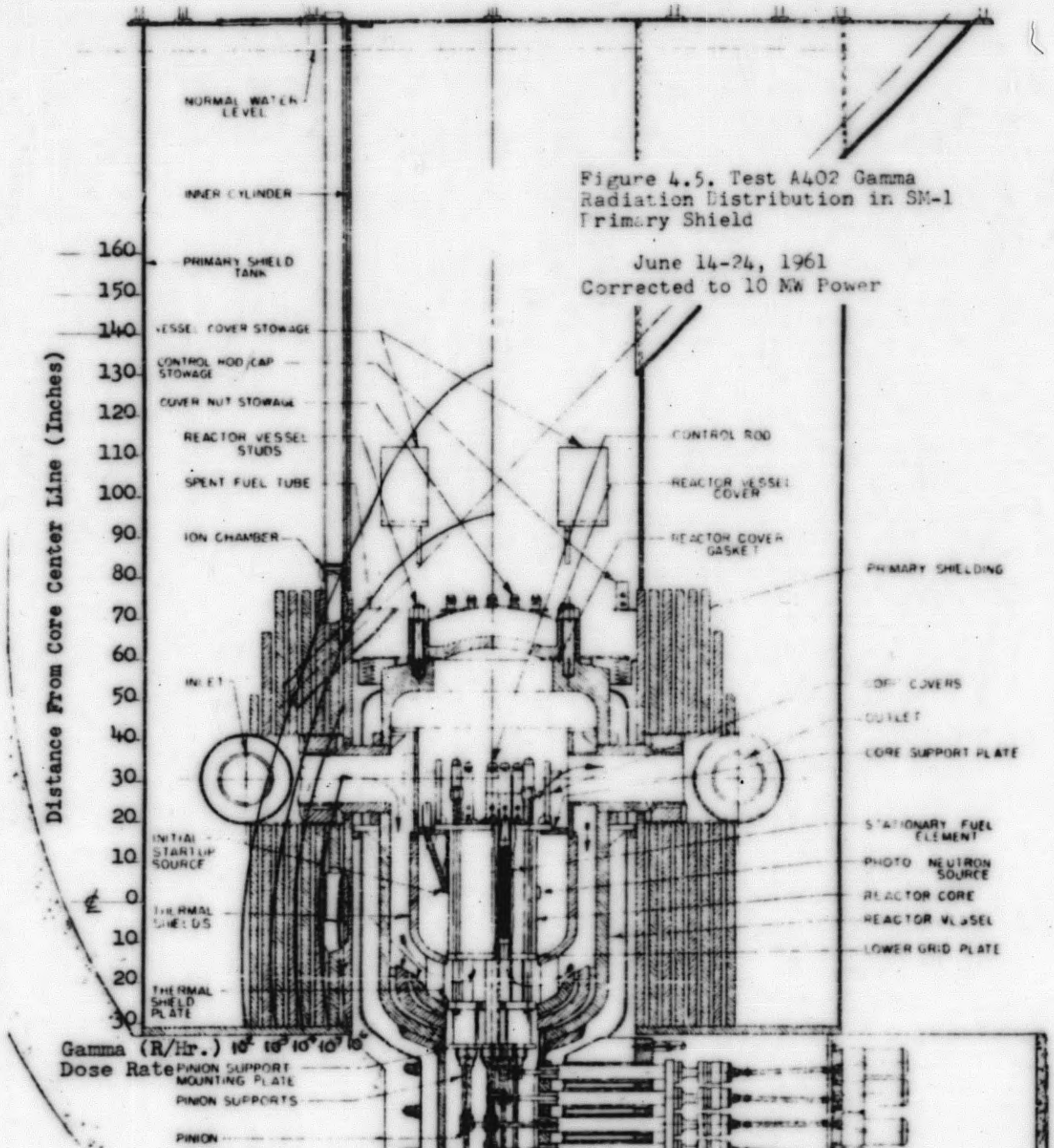

Figure 4.5. Test A402 Gamma Radiation Distribution in SM-1 Primary Shield

June 14-24, 1961
Corrected to 10 MW Power

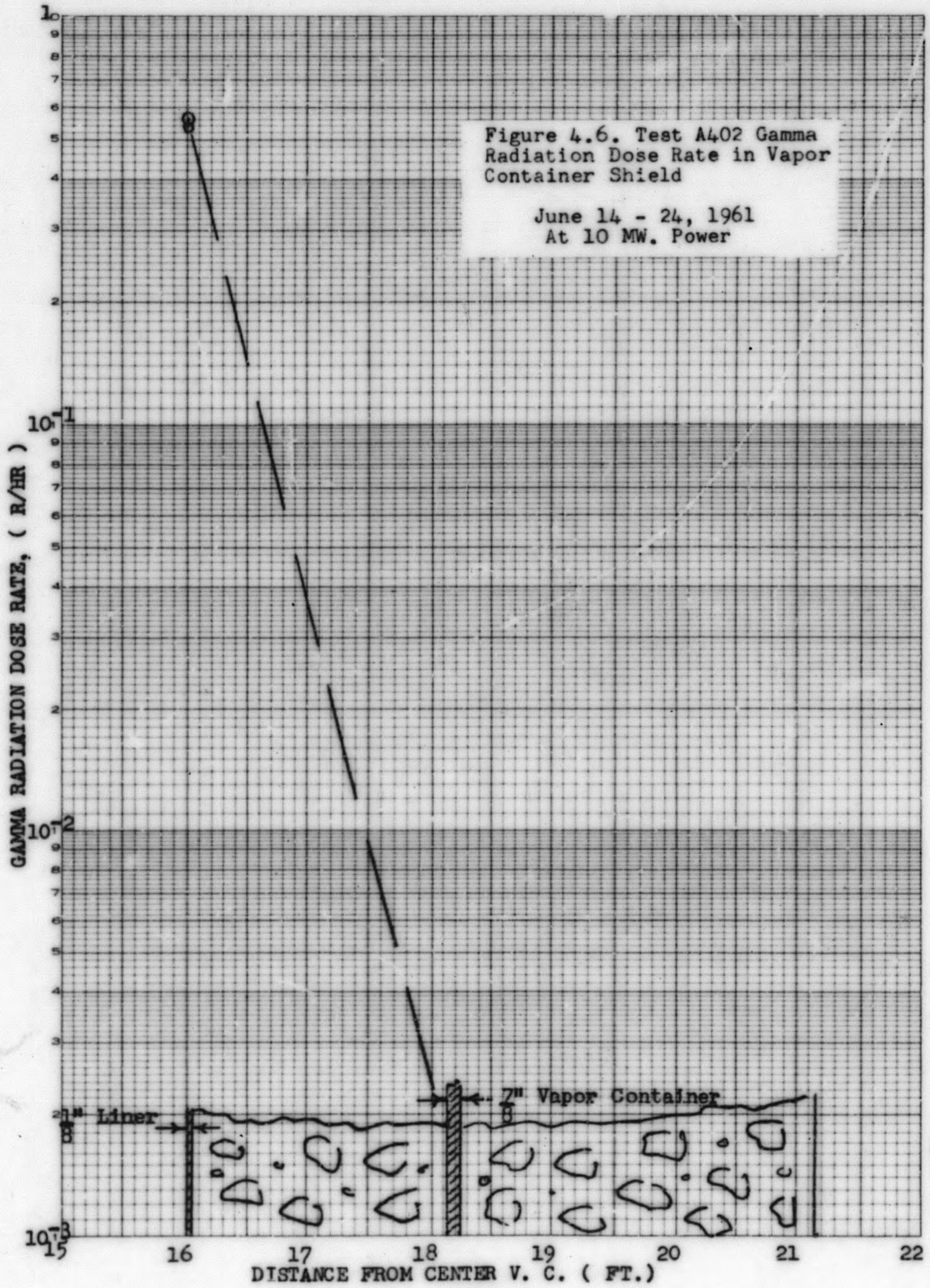
Figure 4.6. Test A402 Gamma Radiation Dose Rate in Vapor Container Shield
June 14 - 24, 1961
At 10 MW. Power
GAMMA RADIATION DOSE RATE, (R/HR)
DISTANCE FROM CENTER V. C. (FT.)
1" Liner
8
7" Vapor Container
8
15
16
17
18
19
20
21
22

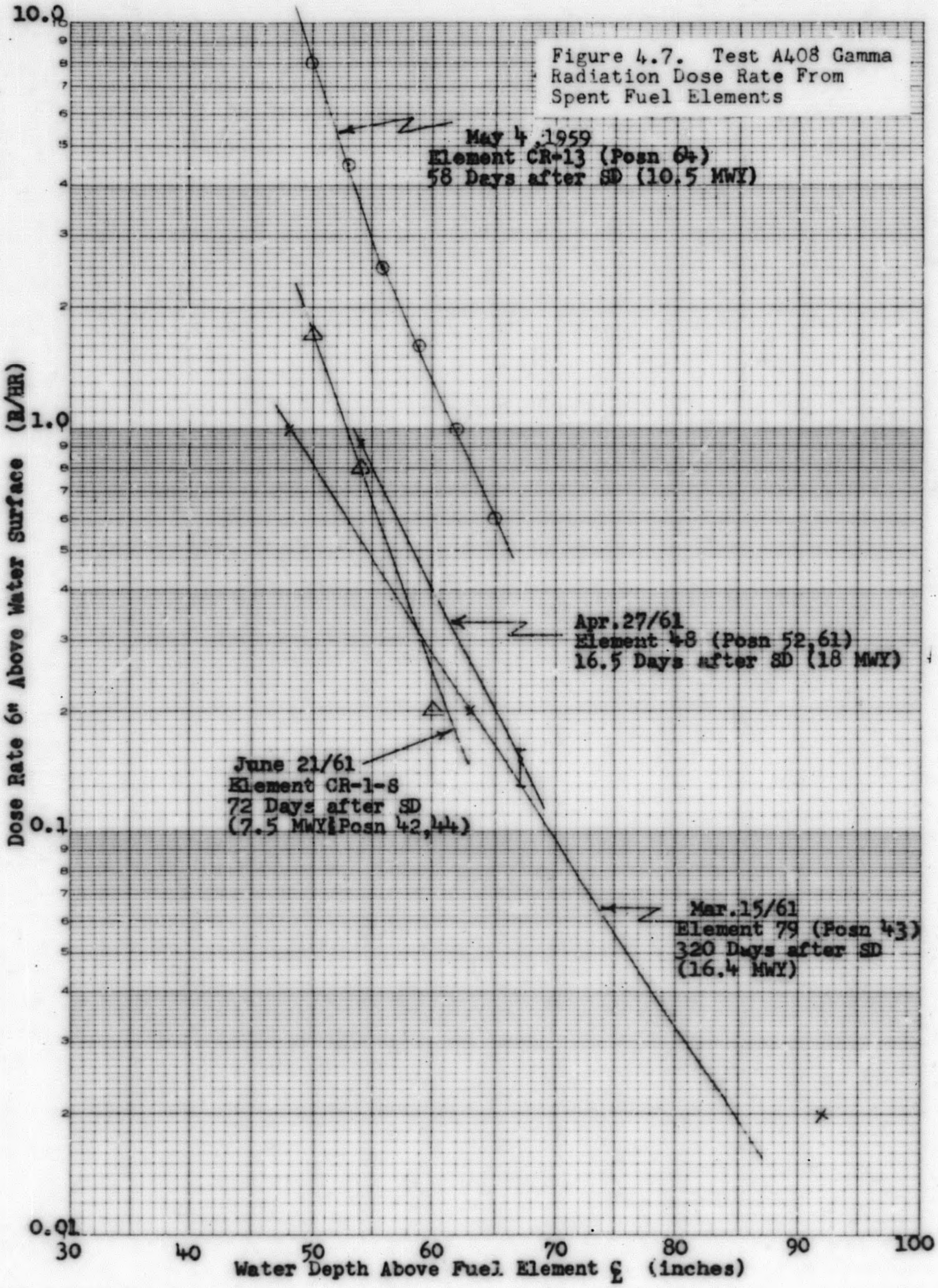
Figure 4.7. Test A408 Gamma Radiation Dose Rate From Spent Fuel Elements
May 4, 1959
Element CR-13 (Posn 64)
58 Days after SD (10.5 MWY)
Apr. 27/61
Element 48 (Posn 52,61)
16.5 Days after SD (18 MWY)
June 21/61
Element CR-1-8
72 Days after SD
(7.5 MWY; Posn 42,44)
Mar. 15/61
Element 79 (Posn 43)
320 Days after SD
(16.4 MWY)
Dose Rate 6" Above Water Surface (R/HR)
10.0
1.0
0.1
0.01
30
40
50
60
70
80
90
100
Water Depth Above Fuel Element ℄ (inches)

5.0 SPENT FUEL ELEMENT RADIATION MEASUREMENTS - TEST A408

The radiation level from spent fuel elements has been measured on several occasions. The element was suspended vertically from the handling tool and raised slowly toward the surface of the water in the spent fuel pit. The dose rate 6 in. above the water surface was measured for two or more readings in the range 0.1 to 2 R/hr; and the water depth was measured by scale markings on the handling tool.

The following elements were monitored:

Date	Element	Position	Years in Reactor	MWYR Irradiation	Days Cooling Time
3-15-61	79	43	3	16.4	320
4-27-61	48	52, 61	4	18.0	16.5
6-21-61	CR-1-S	42, 44	2	7.5	72.0
5-4-59	CR-13	64	2	10.5	58.0

The data for CR-13 was taken from APAE-55[5], and is included in AP Memo-206.[6] The higher dose rates from this element may be attributed to the Co^{60} in the flux suppressor comb at the top of the element. There was no cobalt in the flux suppressor on element CR-1-S.

The dose rates and attenuation curves are shown in Fig. 4.7. They are in reasonable agreement with calculated values. At water depth of approximately 4 ft, the fuel element is effectively a point source and the only significant gamma energies are in the range of 1.0 to 2.0 Mev. These are tabulated as fission product groups III, IV and V in a number of references, and their activity and attenuation can easily be calculated.

6.0 CONCLUSIONS

1. This series of tests complemented earlier measurements and extended knowledge of neutron flux and gamma radiation dose rate throughout the primary shield, shown by the contour plots in Fig. 3.6 and 4.5.

2. Neutron flux and gamma dose rates in the outer shield tank were higher than previously reported by a factor of 3 or more. The previous neutron activations were not converted to absolute flux values. Discrepancy between these and previous measurements is attributed either to systematic errors in power level, activity measurements, or data reduction. It is possible the present measurements were high by 50 percent, or a factor of 1.5 to 2, and the previous were low by 50 percent, or a factor of 2, which would account for the difference, but this is not fully explainable.

3. At the BF_3 counter position in the bottom of the instrument well, the neutron flux is 5.5×10^9 n/cm^2 sec and the gamma radiation level is 8×10^5 R/hr at a power level of 10 Mw. Raising the startup chamber 24 in. will reduce these levels to 3×10^9 n/cm^2 sec and 9×10^4 R/hr, considering the sensitive area for neutrons to be 6 in. from the bottom and for gamma radiation to be the cable connector at the top of the chamber. Raising the chamber 72 in. (the total travel of the lifting mechanism) will reduce the levels to 1.5×10^7 n/cm^2 sec and 1.7×10^2 R/hr. This information will provide understanding of the operation of the BF_3 counter and provide evidence for malfunction analysis.

4. The high gamma radiation levels observed in the rod drive pit can be attributed to activation induced by the high neutron flux in this area.

5. Gamma dose rates measured at 6 in. above the water surface of the spent fuel pit during handling of spent fuel elements agreed reasonably well with calculations.

7.0 RECOMMENDATIONS

7.1 ADDITIONAL SHIELDING TESTS

1. Further studies of the neutron flux and gamma dose rates in the rod drive pit would be of value to compare with past data and thus establish the relative change in these levels. Tests A406 and A407 could be usefully repeated at annual intervals, and the high flux area should be more completely mapped.

2. Additional flast flux measurements using sulfur pellets should be obtained and used to extend this data closer to the reactor vessel and above the vessel head. These data would improve the fast flux mapping for radiation damage studies.

7.2 ANALYSIS OF SHIELDING DATA

1. Analysis of the data contained in this report should be performed. These analyses will aid in insuring the continued adequacy of the SM-1 shielding. Furthermore, the adequacy of the SM-1 shielding should be predicted with regard to future power levels and core loadings which may exceed present design values.

2. Effort should be directed toward identifying radiation sources in the rod drive pit area. Knowledge of these sources is prerequisite to optimum placement of additional shielding if a reduction of radiation levels in the rod drive area is desired.

3. Analysis of the spent fuel shielding data should establish minimum allowable water depths in the spent fuel tank. Minimum water shielding requirements during spent fuel transfer should be determined.

4. Analytical methods for predicting the neutron flux and gamma dose rates at "off-midplane" locations within the primary shield should be developed. The data from tests A401 and A402 should be used to determine the accuracy of the analytical model.

8.0 REFERENCES

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