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OPERATION BREN

AN EVALUATION OF SIMPLE IRON-WATER
RADIATION SHIELDS AND RADIATION
MEASUREMENTS WITHIN CONCRETE-LINED
AND -CAPPED PITS

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OPERATION BREN: PROGRAM 3 AN EVALUATION OF SIMPLE IRON-WATER RADIATION SHIELDS AND RADIATION MEASUREMENTS WITHIN CONCRETE-LINED AND -CAPPED PITS

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Abstract

Program 3 of Operation BREN consisted in a study of the performance of biological shields composed of various arrangements of steel, water, and borated polyethylene in spherical, cubical, and parallelepipedal geometries. Studies were also made of the radiation levels within concrete-lined and -capped pits, 4 ft in diam and 8 ft deep. The principal radiation source was the ORNL Health Physics Research Reactor, an unshielded, unreflected, 90 wt% U—10 wt% Mo, cylindrical bare-metal assembly. It was supported at heights ranging from 27 to 1500 ft by a hoist car riding outside of a 1527-ft-high steel tower. A 1200-curie Co^{60} gamma-ray source, similarly supported, was substituted for the reactor for some measurements. The work reported includes measurements of fast-neutron and gamma-ray dose rates and thermal-neutron fluxes in air and within the various shields, using the reactor source, measurements of gamma-ray dose rates in air from the Co^{60} gamma-ray source, and measurements of gamma-ray pulse-height spectra within the pits, also employing the Co^{60} source.

Acknowledgments

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1. Introduction

During the spring and summer of 1962 an extensive series of nuclear radiation experiments was carried out at the Atomic Energy Commission's Nevada Test Site. These experiments were designated as Operation BREN (Bare Reactor Experiment, Nevada) since the primary radiation source employed was the ORNL Health Physics Research Reactor (HPRR), an unshielded, unreflected, bare metal assembly with capability for both steady-state and "burst" modes of operation. For the BREN tests the HPRR was mounted on the hoist car of a 1527-ft-high steel tower, located in Area 4 of Yucca Flat at the test site.

As a part of the BREN tests, a comprehensive study of the performance of shields having simple spherical, cylindrical, cubical, or parallelepipedal geometry was attempted. This work was sponsored by the U.S. Army Ordnance -- Ballistic Research Laboratories (USAO-BRL), with the field measurements being carried out by members of the ORNL Neutron Physics Division. The bare metal reactor represented a reasonable steady-state approach to the radiation environment produced by a fission-type weapon, and the principal interest lay in experiments in which it was used. An additional series of measurements was made, however, with a 1200-curie Co^{60} gamma-ray source replacing the reactor on the tower.

The experiments performed were of three general classifications. In the first case, designated "free field" data, were the measurements intended to evaluate the overall radiation environment at various distances from the reactor as a function of reactor height. These data were required primarily to serve as reference points for the second series of measurements, those made inside the various shield enclosures.

The third group of measurements comprised those experiments in which the 1200-curie Co^{60} source replaced the reactor. Included in this group were measurements of gamma-ray spectra for various reactor-detector separations.

GROUP 1

Excluded from automatic downgrading
and declassification

2. Test Objectives

Prior to the initiation of the field work, a detailed set of objectives was prepared by representatives of USAO-BRL. This set listed the measurements desired and assigned a relative priority to each. The studies proposed are listed below in the order of the assigned priorities. Group A consists of the measurements for which the reactor served as the source, Group B those employing the Co⁶⁰ source.

Group A:

1. Attenuation by various shields, as a function of ground range, reactor height, and shield orientation with respect to the reactor.
2. The effect of varying the properties of selected wall areas of a shield.
3. The effect on total gamma-ray dose of using borated materials to modify secondary gamma-ray production.
4. The effect of hydrogenous materials used as shield linings.
5. The effect of changing the albedo of one side of a cubical shield.
6. The variation of dose as a function of position within a shield.
7. The result of changing shield shape from cubical to spherical while retaining the same wall configuration and approximately the same interior dimensions.
8. The effect of changing the interior volume of a cubical shield, retaining the same wall configuration.
9. The effect of varying the composition of the ground surface in the vicinity of a shield.
10. The result of removing thermal-neutron-absorbing materials from selected areas of the inside or outside surfaces of a shield.

11. The variation of dose with depth within a cylindrical hole in the ground, as a function of reactor height.

12. Secondary gamma-ray production in thin shields and its variation with changes in shield geometry.

Group B:

1. Same as 1, Group A.
2. Same as 2, Group A.
3. Same as 6, Group A.
4. Same as 8, Group A.
5. Same as 7, Group A.
6. Same as 11, Group A.
7. Same as 9, Group A.

At least a few measurements were completed in nearly all parts of Group A. Group B measurements, with the exception of parts 1 and 6, had to be abandoned because the 1200-curie Co^{60} source proved to be of insufficient intensity.

3. Test Site

An aerial view of the test site is shown in Fig. 3.1, in which the camera is looking roughly southeast. The 1527-ft-high steel tower which supported the reactor and Co^{60} source is at the lower right. More detail is provided by the plan view of Fig. 3.2. Level, graded areas or "pads," clear of any obstruction, were prepared at distances of 170, 500, 1125, 1500, 2000, and 2175 ft from the tower for placement of the shields. The adjacent access road accommodated support vehicles, and a strip 8-ft wide was bladed at the other end of the pads for placement of instrument cables. Instruments and personnel were located in a trailer and a vanette stationed 2250 ft from the tower in an 11-ft-deep excavation, shown in Fig. 3.3. Sandbags were placed along the rim of the excavation toward the tower to reduce the radiation striking the trailer and vanette.

The intention in constructing the pads at several distances was to obtain the greatest possible variation in slant range and angle of

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Fig. 3.1

Fig. 3.2

Fig. 3.3

incidence of the radiation. Use of the shorter range pads, however, was predicted on reactor runs of short duration, so that shields could be exchanged or modified during reactor shutdowns. Shortly after the start of the BREN program it became clear that most reactor runs would be for relatively long times. Therefore the shields were moved to the 2175-ft pad, at which point the radiation level was low enough to permit changes in the shield configurations while the reactor was operating, without incurring the risk of excessive personnel exposures. The low radiation intensity at this distance, however, resulted in only a few counts/min within many of the shield configurations, increasing the running time per data point and reducing the statistical accuracy.

4. Reactor, Co⁶⁰ Source, and Positioning Arrangement

The reactor which served as the principal source in these experiments, the HPRR, has been described elsewhere.¹ Basically it is an unreflected and unmoderated right cylinder of 90 wt% U -- 10 wt% Mo, with the uranium enriched to 93.2 wt% in U²³⁵. Although particularly intended for the production of neutron bursts of the order of 10¹⁷ fissions, it is designed to operate under steady-state conditions at a power level of 1 kW with natural convection cooling, or at 10 kW with forced-air cooling for 10 min. Although the reactor was used in its burst mode for other phases of the BREN operation, the steady-state mode was employed for the present studies.

The 1200-curie Co⁶⁰ source which was used for the supplementary series of measurements during these tests is shown in Fig. 4.1.

The aluminum hoist car which supported the reactor or source on the steel tower can be seen in Fig. 4.2. The car was designed to ride outside the tower legs in order to minimize shielding by the tower, and permitted positioning the reactor or source at heights ranging from 27 to 1500 ft above the ground. The height of the reactor was accurately

Fig. 4.1

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determined by a simple surveyor's triangulation, the operation being performed in Fig. 4.3.

Reference.

1. F. W. Sanders et al., Operation Plan and Hazards Rept. - Operation BREN, CEX-62.02 (April, 1962), p. 20.

5. Instrumentation

Two types of detection instrument were used for fast-neutron dose-rate measurements. The first was a recoil-proton proportional counter (FND), of the well-known Hurst type. Goldstein¹ has provided a knowledgeable general discussion of such counters, covering their assets and liabilities. The counter is calibrated to give a response proportional to the single-collision dose in tissue. Its low-energy cutoff is at about 200 keV.

The second type of detector was a modified long counter (MLC), consisting of a relatively high pressure (~ 1 atm) BF₃ tube enclosed in a cadmium-covered polyethylene cylinder. This counter was originally developed for close-up, leak-searching of reactor shields. The thickness of polyethylene on the end of the counter is adjusted to give a counter response proportional to the multicollision dose, and the low-energy response of the device extends to the cadmium cutoff.

Unfortunately, a study of the modified long counter, made after the present work was completed (actually instigated by the lack of agreement between FND and modified long counter results) has shown that its response is asymmetric and, over much of its sensitive region, quite different from a dose response. Measurements made with this device, therefore, if interpretable at all must be interpreted as weighted by an energy-dependent counter function, rather than by an energy-dependent dose function.

Both fast-neutron detectors were "calibrated" by exposing them to a Po-Be neutron source of known yield and using the count rate observed to compute a count rate to "dose-rate" conversion factor. (The quotation marks embody the cautions of the preceding paragraph.)

Fig. 4.3

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Gamma-ray dose-rate measurements were made with anthracene crystal scintillation dosimeters (ASD). One of these instruments was the dosimeter described by Mustafa and Blosser.² The rest of the instruments were intended to essentially be duplicates of the first, with only minor modification of the housing to improve weather resistance. Apparently, however, some unintended variations were introduced during their construction, with the result that the data from the newer ASD's seems to imply a response dependent on both energy and intensity.

Gamma-ray dosimeters were calibrated by exposing them to a known-intensity Ra^{226} gamma-ray source and using the observed count rate to compute a count-rate to dose-rate conversion factor. Day-to-day calibration was against small Cs^{137} sources mounted in caps which fitted over the crystal end of the detector housing or which, in the case of the newer dosimeters, fitted the protective shell some distance from the crystal.

Thermal-neutron flux measurements were made with two types of BF_3 proportional counters. The first was the high-pressure system also used in the modified long counter and described by Mustafa and Blosser.³ This system was used as the standard, since it had been previously calibrated. The second counter differed slightly in its physical dimensions and filling pressure, and was slightly more sensitive.

The dosimeters noted above are customarily integrally packaged with their preamplifiers, but for the present application the detector head was separated from the preamplifier by 54 in of coaxial cable. This was necessary because there was not enough room inside some of the shields for both detector and preamplifier. In addition, with the detector inside and the preamplifier outside a shield, it was not necessary to disturb the temperature equilibrium within the shield when preamplifier maintenance was required. The additional capacitance contributed by the 54-in. cable was not significant.

A gamma-ray spectrometer was used in the final stages of the program to measure the energy distribution as a function of range for several heights of the Co^{60} source.

The spectrometer consisted of a 3-in.-diam, 3-in.-high NaI(Tl) crystal mounted on a DuMont 6363 photomultiplier tube feeding a single-channel pulse-height analyzer.

The values of reactor power used in reducing the data of this experiment were based upon the response of a conventional long counter located 2390 ft from the reactor tower. The long counter was placed so that the axis of the detector was perpendicular to the direct beam radiation. In this manner the geometry of the detector did not vary with respect to changes in reactor height. (Its counts were normalized to a value of flux determined by the activity induced in a sulfur foil exposed on the reactor during the day's run.) The counting system was calibrated against a Po-Be source before and after each run.

The major portion of the difficulties encountered in performing the present experiments and in reducing the data centered around instrument malfunction. The difficulty with the modified long counter has been previously mentioned. Errors in the gamma-ray dose-rate measurements could actually be separated into several categories. Soon after the experiment started the original set of instruments had to be withdrawn for use elsewhere and new equipment installed. Overlooked at the time was a wrong capacitor value at the input of the ASD preamps and this error was not discovered until checks were made at ORNL following the experiment. These checks indicated a nonlinear response from the instrument when the radiation intensity at the detector was varied over wide ranges.

Errors were introduced into the data during long periods of reactor operation as a result of large temperature differences and instrument vibration due to rough handling; both giving rise to gain shifts in the instruments. An attempt was made to correct the data for changes in temperature but it was not feasible for all runs since measurements during any reactor run would be made both inside and outside shields where a considerable difference in temperature existed. It was impossible to correct for changes due to vibration since there was no way of knowing when the shift in gain had occurred.

References

1. Herbert Goldstein, Fundamental Aspects of Reactor Shielding, Addison-Wesley, Reading, Massachusetts (1959), p. 123.
2. T. Mustafa and T. V. Blosser, Neutron Phys. Div. Ann. Progr. Rept., Sept. 1, 1961, ORNL-3193, p. 110.
3. Op. cit., p. 102.

6. Experimental Procedure

The shields within which the data were obtained will be individually described below as the results obtained with a particular shield are reported. However, almost without exception they were ponderous and unwieldy. In order to place the detectors ^{inside the shield}, part of the shield had to be removed, a task requiring the use of a crane. The experimental procedure therefore was predicated upon obtaining the maximum amount of information from a single set up. Usually the thermal-neutron detector and the gamma-ray detector could be operated side-by-side within a shield, but the large size of the modified long counter compelled an individual set up for its use. A typical positioning of instruments inside a shield is shown in Fig. 6.1.

A variety of modifications was possible with most of the shields. Cavities within the shield walls could be empty or filled with plain or borated water, boroxyl normally covering the walls could be incrementally removed, shields could be placed on the ground or on aluminum stands, shields could be rotated, etc. As many variations as possible were made for a single detector placement.

Fig. 6.1

7. Results

7.1. Free-Field Data with HPRR as Source

7.1.1. Fast Neutrons. We have previously noted the difficulties with the data obtained with the modified long counter (MLC). However, the high order of sensitivity possessed by this instrument resulted in the acquisition of statistically good data in many situations where the FND was of little or no use. Therefore it has been decided to present the data obtained with the MLC, in the hope that at least they may be useful in evaluating relative attenuations by the various shields. The numbers associated with these data are those resulting from the application of a conversion factor to the counts per minute data of the instrument scalers. The conversion factor was determined by "calibrating" the counter against a Po-Be source of known strength and relating the counting rate observed to the calculated first-collision tissue dose from the source. For convenience, then, we will call the quantity measured by the MLC a "pseudo dose," and specify the same units as those used for the true, or at least accepted, first-collision tissue dose read by the proton-recoil dosimeter, i.e., $\text{erg}\cdot\text{g}^{-1}\cdot\text{hr}^{-1}\cdot\text{W}^{-1}$, stipulating that the $\text{erg}\cdot\text{g}^{-1}$ portion of the units is simply a convenient substitution for a possibly energy-dependent quantity presently not known.

Fast-neutron dose-rate measurements in air, obtained with the FND's are shown in Table 7.1. Similar data obtained with the MLC's are presented in Table 7.2. Dose rates and "pseudo dose" rates have been plotted as functions of the slant range from reactor to detector in Fig. 7.1. The higher group of curves are the MLC results, the lower group the FND-measured dose rates. The data of both sets seems to fit a relaxation length of roughly 750 ft, but there is obviously a factor of 10 to 15 in absolute value between the two sets of results.

In order to avoid clutter, no errors have been shown on the points in the drawing. The error based on counting rates alone ranges from ± 1 to $\pm 5\%$ for the FND data, and is only a fraction of a percent for the MLC data.

7.1.2. Slow Neutrons. Slow-neutron flux measurements in air are shown in Table 7.3. At locations where both a bare BF_3 and a cadmium-covered BF_3 measurement were obtained, the quantity tabulated is the 2200 m/sec flux, in units of $\text{ft}^2 \cdot \text{neutrons} \cdot \text{cm}^{-2} \cdot \text{sec}^{-1} \cdot \text{W}^{-1}$. In all other cases the quantity is appropriately identified.

The thermal-neutron flux as a function of reactor-detector slant range is shown in Fig. 7.2 for five different reactor heights. Beyond the 2100-ft range the data for the various reactor heights are essentially parallel, and are characterized by a relaxation length of 860 ft.

7.1.3. Gamma Rays The free-field gamma-ray dose rates at various distances from the HPRR are presented in Table 7.4. Only those data in which reasonable confidence is placed are given.

The free-field gamma-ray dose rates are plotted as functions of reactor-detector slant range in Fig. 7.3. All of the data can be fitted by a relaxation length of 944 ft. The bulk of the data of Fig. 7.3, however,

Table 7.1. Fast-Neutron Dose Rates in Air (Multiplied)
by $4\pi R^2$ at Various Distances from the HPRR.
Detector: Proton-Recoil Dosimeter

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times$ Dose Rate (ft ² ·erg·g ⁻¹ _{tissue} ·hr ⁻¹ ·W ⁻¹)	Notes
x 10 ⁴			
<u>Reactor Height: 27 ft</u>			
5A	183	3.93	
5B	183	4.11	
5C	181	4.09	
8E	421	3.48	
5D	526	3.36	
5E	525	3.60	
5F	525	3.54	
5G	525	3.60	
7C	1175	1.76	
7D	1175	1.64	
8A	1196	1.61	
8B	1190	1.61	
8C	1190	1.62	
8D	1183	1.66	
9B	1584	0.961	
9C	1581	0.958	
51A	2230	0.377	
<u>Reactor Height: 170 ft</u>			
53A	239	4.68	
<u>Reactor Height: 300 ft</u>			
10D	1212	2.26	
11A	1234	2.48	
11B	1222	2.27	
9D	1589	1.41	
10A	1593	1.63	
10B	1587	1.58	
10C	1592	1.61	
<u>Reactor Height: 1125 ft</u>			
13A	1644	1.44	
13B	1634	1.35	
13C	1626	1.36	
15A	1645	1.47	
15B	1638	1.55	

Table 7.1 (cont.)

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose Rate}$ (ft ² ·erg·g ⁻¹ _{tissue} ·hr ⁻¹ ·W ⁻¹)	Notes
15C	1637	1.54	
16A	1638	1.66	
18A1	1694	1.41	
18B1	2574	0.522	
19A1	2524	0.508	
20A1	2490	0.523	
25A1	2511	0.503	
42A4	2503	0.470	Detector on ground
44A1	2487	0.557	
<u>Reactor Height: 1500 ft</u>			
48A8	2627	0.545	Detector 180° to reactor
48AG	2626	0.565	Detector on ground
48AF	2628	0.548	
48AH	2627	0.555	Detector 0° to reactor
48A9	2627	0.537	Detector 90° to reactor
48A7	2627	0.558	Detector 270° to reactor

a. Normalized to air density of 1.0×10^{-3} g/cm³.

Table 7.2. "Pseudo" Fast-Neutron Dose Rates in Air (Multiplied
by $4\pi R^2$) at Various Distances from the HP RR.
Detector: Modified Long Counter

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2$ x Pseudo-Dose Rate (ft ² .erg.g ⁻¹ .hr ⁻¹ .W ⁻¹)	Notes
x 10 ⁵			
<u>Reactor Height: 27 ft</u>			
4A	185	3.21	
5A	183	2.89	
5B	183	3.06	
5C	181	3.11	
4B	539	3.41	
4C	538	3.33	
4D	533	3.40	
4E	533	3.44	
5D	526	3.41	
5E	525	3.38	
5F	525	3.38	
5G	525	3.35	
6A	532	3.21	
6B	529	3.18	
6C	527	3.21	
6D	1182	2.14	
6E	1181	2.16	
7A	1200	2.14	
7B	1189	2.17	
7C	1175	2.15	
7D	1175	2.13	
8A	1196	2.10	
8B	1190	2.12	
8C	1190	2.10	
8D	1183	2.13	
8E	1182	2.13	
9B	1584	1.37	
9C	1582	1.37	
51A	2226	0.620	
52B	2165	0.432	1/2 in. boroxyl around detector, which is 2 ft above ground
52B	2165	0.523	1/4 in. boroxyl around detector, which is 2 ft above ground
52B	2165	0.657	No boroxyl; detector 2 ft above ground
51A	2763	0.320	
51A	3073	0.211	
51A	3589	0.106	
51A	4102	0.0539	

Table 7.2 (cont.)

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times \text{Pseudo-Dose Rate}$ (ft ² .erg.g ⁻¹ .hr ⁻¹ .W ⁻¹)	Notes
<u>Reactor Height: 170 ft</u>			
53A	239	4.78	
<u>Reactor Height: 300 ft</u>			
10D	1212	3.10	
11B	1222	3.06	
11C	1222	2.95	
14A	1218	3.03	
14B	1204	3.04	
9D	1589	2.06	
10A	1593	2.12	
10B	1587	2.13	
10C	1592	2.10	
11A	1620	2.04	
11B	1605	2.05	
11C	1605	2.00	
11D	1604	2.04	
11E	1613	1.98	
45A	2243	0.863	
45B	2263	0.879	
45B2	2263	0.877	
51BG	2213	0.879	Detector on ground
51B	2213	0.940	
51B	2738	0.498	
51B	3038	0.342	
51B	3540	0.173	
51B	4042	0.0881	
<u>Reactor Height: 500 ft</u>			
49B1	2075	1.31	
49B2	2246	0.04	
41A	2292	0.989	
41BG	2285	0.980	Detector on ground
49B3	2762	0.542	
46C	3173	0.315	
49B4	3555	0.186	
49B5	4058	0.0954	
41A	2292	0.976	
<u>Reactor Height: 1125 ft</u>			
15B	1638	3.00	

Table 7.2 (cont.)

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times \text{Pseudo-Dose Rate}$ (ft ² .erg.g ⁻¹ .hr ⁻¹ .W ⁻¹)	Notes
15B	1930	2.29	
17B5	2559	1.02	
18B2	2574	0.961	
18B4G	2575	0.882	Detector on ground
19A1	2524	1.05	
20A1	2490	1.13	
20A2G	2489	1.02	Detector on ground
44B4	2926	0.640	
44B3	3204	0.442	
44B1	3674	0.232	
44A	4172	0.117	
<u>Reactor Height: 1500 ft</u>			
50A	2103	1.99	
50A	2479	1.25	
40A1G	2684	0.868	Detector on ground
40A2	2684	0.929	Detector 24 in. above ground
40A1G	2684	0.883	Detector on ground
40A2	2684	0.943	Detector 24 in. above ground
47A1	2634	1.03	
47A2	2634	1.12	Detector 30 ft above ground
47A3	2638	0.0209 ^b	Detector pointed 180° from reactor
47A4	2638	0.0274	Detector at 90° to reactor
47A5	2638	0.0477	Detector 0° to reactor
50A	2620	1.01	
48A	2627	1.07	Detector hor. and 0° to reactor
48A	2627	1.11	Detector hor. and 90° to reactor
48A	2623	1.14	Detector hor. and 180° to reactor
48A	2623	1.13	Detector hor. and 270° to reactor
48A	2623	1.16	Detector pointing toward ground
48A	2623	1.07	Detector pointing toward sky
48A	2623	1.00	Detector pointing toward reactor
48A	2620	1.07	Detector hor. and 0° to reactor
48A	2620	1.09	Detector hor. and 0° to reactor
48A	2620	1.14	Detector hor. and 90° to reactor
48A	2620	1.19	Detector hor. and 180° to reactor
48A	2620	1.18	Detector pointing toward ground
48A	2620	1.09	Detector pointing toward sky
48A	2620	1.04	Detector pointing toward reactor
48A	2620	1.17	Detector pointing away from reactor
50A	3059	0.572	
50A	3321	0.403	
46A2	3450	0.351	

Table 7.2 (cont.)

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times$ Pseudo-Dose Rate (ft ² ·erg·g ⁻¹ ·hr ⁻¹ ·W ⁻¹)	Notes
50A	3769	0.219	
50A	4224	0.118	

- a. Normalized to air density of 1.0×10^{-3} g/cm³.
- b. For these data the detector was shielded within Geometric 3, Configuration 3, so that only the front face was exposed to the radiation field.

Fig. 7.1

Table 7.3. Slow-Neutron Fluxes in Air (Multiplied by $4\pi R^2$)
at Various Distances from the HPRR

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times \text{Flux}^b$ (ft ² ·neutrons. cm ⁻² ·sec ⁻¹ ·W ⁻¹)	Notes
x 10 ⁷			
<u>Reactor Height: 27 ft</u>			
4A	185	1.12	
5A	183	0.997	
5B	183	0.113	Cadmium-covered BF ₃ flux
5C	181	1.10	
5A3	183	1.13	
5B3	183	0.155	Cadmium-covered BF ₃ flux
5C3	181	1.11	
4B	539	1.55	
4C	538	1.51	
4D	533	1.52	
4E	533	0.191	Cadmium-covered BF ₃ flux
5D	526	1.53	
5E	525	0.182	Cadmium-covered BF ₃ flux
5F	525	0.189	" " " "
5G	525	1.50	
6A	532	1.63	
6B	529	0.200	Cadmium-covered BF ₃ flux
6C	527	0.202	" " " "
4E	533	0.265	" " " "
5D3	526	1.68	
5E3	525	0.268	Cadmium-covered BF ₃ flux
5F3	525	.273	" " " "
5G3	525	1.65	
6A3	532	1.66	
6B3	529	0.268	Cadmium-covered BF ₃ flux
6C3	527	0.268	" " " "
6D	1182	0.166	Cadmium-covered BF ₃ flux
6E	1181	0.167	Cadmium-covered BF ₃ flux
6F	1181	1.22	
7A	1200	1.34	
7B	1189	1.31	
7C	1175	.167	Cadmium-covered BF ₃ flux
7D	1175	1.28	
7E	1175	.158	Cadmium-covered BF ₃ flux
8A	1196	1.24	
8B	1190	1.28	
8C	1190	.162	Cadmium-covered BF ₃ flux
8D	1183	.165	" " " "
8E	1182	1.26	
6D3	1182	0.223	Cadmium-covered BF ₃ flux
6E3	1181	0.221	" " " "
6F3	1181	1.27	

Table 7.3 (cont.)

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times \text{Flux}^b$ (ft ² ·neutrons· cm ⁻² ·sec ⁻¹ ·W ⁻¹)	Notes
7A3	1200	1.29	
7B3	1189	1.27	
7C3	1175	0.221	Cadmium-covered BF ₃ flux
7D3	1175	1.32	
7E3	1175	.224	Cadmium-covered BF ₃ flux
8A3	1196	1.27	
8B3	1190	1.29	
8C3	1190	0.219	Cadmium-covered BF ₃ flux
8D3	1183	0.224	" " " "
8E3	1182	1.32	
9B	1584	0.857	
9C	1582	0.814	
9B3	1584	0.856	
9C3	1582	0.824	
51A	2224	0.334	Thermal (2200 m/sec) flux
51A	2761	0.184	" " "
51A	3059	0.129	" " "
51A	3576	0.0625	" " "
51A	4083	0.0322	" " "
<u>Reactor Height: 300 ft</u>			
10A	1213	1.82	
10B	1209	0.244	Cadmium-covered BF ₃ flux
10C	1213	0.238	" " " "
10D	1212	1.86	
11A	1234	1.73	
11B	1222	1.73	
11C	1222	1.70	
12A	1227	0.230	Cadmium-covered BF ₃ flux
12B	1202	1.65	
9D	1589	1.30	
9D3	1589	1.24	
10A3	1593	1.30	
10B3	1587	0.240	Cadmium-covered BF ₃ flux
10C3	1592	0.237	" " " "
10D3	1592	1.33	
45A1	2243	0.364	Thermal (2200 m/sec) flux: on ground
45B1	2263	0.410	" " " " "
51B	2213	0.461	" " " " "
51B	2737	0.287	" " " " "
51B	3038	0.195	" " " " "
51B	3539	0.0989	" " " " "
51B	4040	0.0478	" " " " "

Table 7.3 (cont.)

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times \text{Flux}^b$ (ft ² ·neutrons. cm ⁻² ·sec ⁻¹ ·W ⁻¹)	Notes
<u>Reactor Height: 500 ft</u>			
49E	2084	0.707	Thermal (2200 m/sec) flux
49E	2255	0.565	" " "
41B1	2285	0.617	On ground
41B2	2286	0.650	
41B3	2286	0.0855	Cadmium-covered BF ₃ flux
41B	2286	0.563	Thermal (2200 m/sec) flux
49E	2773	0.308	" " "
46C12	3173	0.194	" " "
49E	3570	0.106	" " "
49B	4057	0.0539	" " "
<u>Reactor Height: 1125 ft</u>			
13A	1937	1.46	
14C	1897	1.47	
25A1	2511	0.749	
25B1	2529	0.104	Cadmium-covered BF ₃ flux
24A1	2490	0.674	Thermal (2200 m/sec) flux
18B	2574	0.620	" " "
18B	2574	0.600	" " "
44B4	2925	0.361	" " "
44B3	3204	0.250	" " "
44B1	3674	0.134	" " "
44A23	4172	0.0663	" " "
<u>Reactor Height: 1500 ft</u>			
40A	2684	0.556	
40A1	2684	0.0810	Cadmium-covered BF ₃ flux
40A2	2684	0.599	
50A	2098	1.11	Thermal (2200 m/sec) flux
50A	2474	0.728	" " "
50A	2617	0.578	" " "
47A1	2633	0.575	" " "
47A	2620	0.511	" " "
50A	3057	0.351	" " "
46AU	3450	0.297	" " "
46AH	3450	0.210	" " "
50A	3766	0.131	" " "
50A	4224	0.0668	" " "

a. Normalized to air density of 1.0×10^{-3} g/cm³.

b. All fluxes are bare BF₃ measurements unless specifically noted otherwise.

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Fig. 7.2

Table 7.4. Gamma-Ray Dose Rates in Air (Multiplied by $42R^2$)
at Various Distances from the HPRR

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose}$ (ft ² ·mr·hr ⁻¹ ·W ⁻¹)	Notes
x 10 ⁴			
<u>Reactor Height: 27 ft</u>			
5ABC	182	13.1	
5DEFG	525	9.37	
6ABC	529	10.1	
4D+E	533	8.95	
7C	1175	5.26	
7D	1175	5.73	
7E	1175	5.50	
6EF	1181	5.70	
6D	1182	6.02	
8D	1183	6.03	
7B	1189	5.46	
8B	1190	5.85	
8C	1190	5.91	
8A	1196	5.76	
7A	1200	4.99	
4A	1210	22.3	Questionable: low count rate
4B	1210	6.82	
4C	1210	5.97	
9C	1582	3.80	
9B	1584	3.83	
8E	1681	3.60	
52A	1996	2.33	
52A	2171	1.98	
52A	2178	1.98	
52A	2695	1.13	
52A	2995	0.815	
52A	3494	0.495	
52A	3994	0.288	
<u>Reactor Height: 300 ft</u>			
12B	1202	8.14	Questionable: poor calibration
11B	1222	6.85	
11C	1222	6.89	
11A	1234	7.02	Questionable: instrument troubles
12A	1227	6.73	
9D	1589	4.89	
10BC	1590	4.89	
10A	1593	4.88	
52C	2009	3.27	
52C	2179	3.04	

Table 7.4 (cont.)

Reactor Run No.	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose}$ (ft ² ·mr·hr ⁻¹ ·W ⁻¹)	Notes
52C	2183	2.76	
45B1	2263	2.41	
52C	2697	1.57	
52C	2991	1.26	
52C	2993	1.21	
52C	3487	0.648	
52C	3981	0.381	
<u>Reactor Height: 500 ft</u>			
49B	1587	5.54	
49B	2068	3.49	
49B	2234	2.97	
41B2	2286	3.67	Questionable: data record uncertain
41B3	2287	3.87	" " " "
49B	2743	1.71	
49B	3037	1.28	
46C	3174	1.08	
49B	3530	0.783	
49B	4023	0.487	
<u>Reactor Height: 1125 ft</u>			
16A	1637	6.97	
15B	1638	7.46	
13A	1937	6.28	
25A1	2511	3.71	
25B1	2524	2.96	
18B3	2574	2.45	
<u>Reactor Height: 1500 ft</u>			
47A2	2618	3.25	In air, 30 ft off ground
47A3	2633	3.20	
47A4	2633	3.09	
47A5	2633	3.11	
47A1	2634	3.23	
47A6	2635	3.05	In air, 30 ft off ground
47A7	2638	3.05	
40A	2684	3.04	
48A	3060	2.00	
48A	3323	1.51	
46A	3450	1.36	
48A	4228	0.592	

a. Normalized to air density of 1.0×10^{-3} g/cm³.

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Fig. 7.2

was obtained with the questionable ASD's, which are suspected to be somewhat intensity-dependent. Therefore the relaxation length determined by these data, 944 ft, is probably best accepted as a lower limit, and may be low by as much as 30 or 40%. As was the case with the fast-neutron and slow-neutron data, the free-field gamma-ray dose increases smoothly with reactor height, nearly doubling as the reactor is raised from 300 to 1500 ft.

7.2. Dose Rates and Fluxes Within Shields

7.2.1. Geometric No. 1 (Large Cube). The shield identified as Geometric No. 1 was a large cube having interior dimensions of 40 x 40 x 40 in. and identical, 16-3/8-in.-thick walls. The walls, from interior to exterior, consisted of 4 in. of iron, a 6-in. void (called the "inner tank"), 1/2 in. of iron, a 4-1/2-in. void ("outer tank"), 1 in. of iron, 1/4 in. of boron-loaded polyethylene — "Borox" — which was segmentally removable, and 1/8 in. of aluminum. A photograph of this shield is shown as Fig. 7.4.

The usual height above ground of the bottom of the shield, on the aluminum stand shown in the photograph, was 14-1/2 in., making the center of the cube 50-1/2 in. above the ground. Various modifications or configurations of the shield walls were made. Those for which usable data were obtained are shown in the diagram of Fig. 7.5.

Fast-neutron dose rates measured within Geometric No. 1 with the FND are given in Table 7.5. It will be noted that in some cases two values are quoted for the same configuration, the one with a vacuum tube integrator and the other with the Perdue-Blosser transistorized integrator. At the time when these experiments were initiated the transistorized integrator was, relatively speaking, an innovation. Therefore, as a check upon its results, the time-tested vacuum tube integrator from the ORNL Lid Tank Shielding Facility (LTSF) was also used.

Fast-neutron "pseudo dose" rates within Geometric No. 1, measured with the MLC, are given in Table 7.6. The sensitivity of the MLC made it possible to investigate a much wider range of configurations than was practical with the FND's.

Slow-neutron fluxes within Geometric No. 1 are shown in Table 7.7. It can be seen from the notes following many of the values that low radiation intensity became a serious problem with these data.

Fig. 7.4

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CONFIGURATION NO.	4 in.	6 in.	4 1/2 in.	1/8 in. ALUMINUM	1/4 in. BOROX	1/2 in. IRON	1 in. IRON
1		BORATED WATER	BORATED WATER				ON
2		BORATED WATER	EMPTY				ON
3		EMPTY	EMPTY				ON
4		PLAIN WATER	PLAIN WATER				ON
5		PLAIN WATER	EMPTY				ON
6		EMPTY	BORATED WATER				ON
7		PLAIN WATER	BORATED WATER				ON
8	IRON	EMPTY	PLAIN WATER	IRON			ON
9		BORATED WATER	PLAIN WATER				ON
10		BORATED WATER	BORATED WATER				OFF
11		BORATED WATER	EMPTY				OFF
13		PLAIN WATER	PLAIN WATER				OFF
14		PLAIN WATER	EMPTY				OFF
15		EMPTY	BORATED WATER				OFF
16		PLAIN WATER	BORATED WATER				OFF
17		EMPTY	PLAIN WATER				OFF
18		BORATED WATER	PLAIN WATER				OFF
12		EMPTY	EMPTY				OFF

Fig. 7.5

Table 7.5. Fast-Neutron Dose Rates (Multiplied by $4\pi R^2$)
Within Geometric No. 1. Radiation Source:
HPRR; detector: FND

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2$ x Dose Rate ($\text{ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$)	Notes
25B1	3	1125	2529	1.32(3) ^b	
25B4	2	1125	2553	5.69(1)	
40B4	2	1500	2737	7.38(1)	
40B4	2	1500	2737	3.31(0) ^c	
51B1	3	300	2214	1.61(3)	
51B1	3	300	2214	1.68(3) ^c	
51B4	3	300	2213	1.25(3)	1/4" boroxo around detector
51B4	3	300	2213	1.25(3) ^c	" " " "
51B5	3	300	2213	9.90(2)	1/2" boroxo around detector
					exc. 1/4" at bottom
51B5	3	300	2213	1.01(3) ^c	
51B9	Free	300	2214	4.89(3)	Free-field measurement;
					on ground outside Geo. 1
51B9	Free	300	2214	5.12(3) ^c	" " " "
53B	2	170	239	3.29(2)	
53B	2	170	239	3.86(2) ^c	

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. Digit in parentheses indicates power-of-ten multiplier; i.e., $1.32(3) \equiv 1.32 \times 10^3$.

c. Indicates value with vacuum tube integrator. All other values with the Perdue-Blosser transistorized integrator.

Table 7.6. Fast-Neutron Pseudo Dose Rate (Multiplied by $4\pi R^2$)

Within Geometric No. 1. Radiation source: HPFR;

detector: modified long counter

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2$ x Pseudo Dose Rate ^b ($\text{ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$)	Notes
51A3	1	27	2228	4.92(1)	
51A5	1	27	2222	4.38(1)	
51A2	2	27	2228	3.07(2)	
52A3	2	27	2171	3.03(2)	Mapping interior: right wall center
				4.15(2)	Front wall center
				3.22(2)	Rear wall center
				3.29(2)	Center point of shield
				2.97(2)	Bottom wall center
				3.59(2)	Bottom center of front wall
				4.00(2)	Top center of front wall
				3.73(2)	Top right corner of front wall
				2.98(2)	Top right corner of center wall
				3.06(2)	Top right corner of rear wall
52B	3	27	2165	2.57(2)	
53C	2	170	238	3.53(3)	
53C	2	170	238	3.43(3)	
14A	1	300	1600	1.65(2)	
14B	1	300	1581	1.86(2)	
12A	3	300	1611	6.19(4)	
12B	3	300	1579	7.14(4)	
12C	3	300	1577	8.21(4)	
12D	3	300	1577	8.47(4)	
45A2	1	300	2242	6.62(1)	
45B1	2	300	2263	4.18(2)	
51B2	3	300	2213	3.46(4)	
51B3	3M	300	2213	2.76(4)	1/4 in. boroxo around detector
51B7	3M	300	2213	2.31(4)	1/2 in. boroxo around detector
					exc. 1/4 in. on bottom
51B6	2M	300	2213	3.18(2)	" " " " "
52C	1	300	2179	6.63(1)	Mapping interior: front wall
					below center
				5.75(1)	Center of bottom wall
				6.97(1)	Center of front wall
				5.71(1)	Center of right wall
				5.74(1)	Right wall: top center
				6.39(1)	Top right corner of front wall
				7.64(1)	Top center of front wall
41A	1	500	2289	6.32(1)	
41B	2	500	2286	4.09(2)	
41A	3	500	2299	3.41(4)	
49E	3	500	2255	3.74(4)	

Table 7.6 (cont.)

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Pseudo Dose Rate}^b$ ($\text{ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$)	Notes
49E	12	500	2256	4.47(4)	
49B1	13	500	2230	7.58(1)	
49B	14	500	2246	5.36(2)	
49E	12M	500	2256	4.25(4)	Modified: boroxo front only
49E	12M	500	2255	3.71(4)	Modified: boroxo all sides
13A	3	1125	1937	7.63(4)	
13B	2	1125	1925	7.59(2)	
13C	1	1125	1916	8.76(1)	
13D	1	1125	1916	9.85(1)	
13E	1	1125	1915	1.02(2)	
17A1	1	1125	2536	5.13(1)	
18B2	1	1125	2574	5.09(1)	
19A1	1	1125	2524	5.41(1)	
19A3	1	1125	2572	5.86(1)	
20A1	1	1125	2490	6.54(1)	Corner toward reactor
29A2	1	1125	2540	5.04(1)	
37A1	1M	1125	2532	3.51(3)	Modified: top inner and outer empty
37A ⁴	1M	1125	2527	1.87(2)	Modified: top outer empty
37A5	1M	1125	2510	1.50(2)	Modified: sides outer empty
37A8	1M	1125	2502	1.09(4)	Modified: sides inner and outer empty
37A9	1M	1125	2502	1.94(2)	Modified: front outer empty
37A12	1M	1125	2495	8.32(3)	Modified: front inner and outer empty
39A1	1M	1125	2497	1.43(3)	Modified: bottom inner and outer empty
39A3	1M	1125	2496	5.99(1)	Modified: bottom outer empty
39A6	1M	1125	2483	5.21(3)	Modified: rear inner and outer empty
39A7	1M	1125	2480	8.60(1)	Modified: rear outer empty
17A2	2	1125	2560	3.57(2)	
18B3	2	1125	2574	4.81(2)	
19A2	2	1125	2562	8.87(2)	
20A2	2	1125	2536	3.85(2)	Corner toward reactor
25B3	2	1125	2542	3.91(2)	
39A11	2	1125	2448	4.59(2)	
39A10	2M	1125	2471	3.76(2)	Modified: bottom outer filled, borated water
42A1	2M	1125	2505	2.85(2)	Modified: outer sides filled borated water
42A2	2M	1125	2504	2.42(2)	Modified: front outer filled borated water
42A3	2M	1125	2504	3.46(2)	Modified: rear outer filled borated water
42A4	2M	1125	2500	3.02(2)	Modified: top outer filled borated water
17B1	3	1125	2560	3.85(4)	
18B4	3	1125	2581	3.56(4)	

Table 7.6 (cont.)

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times$ Pseudo Dose Rate ^b ($\text{ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$)	Notes
20A3	3	1125	2570	3.53(4)	
25B2	3	1125	2542	4.04(4)	Corner toward reactor; det. 45°
24A1	3M	1125	2534	4.07(4)	Corner toward reactor; det. 45°
24A2	3M	1125	2534	4.07(4)	Corner Orientation
43A7	4	1125	2462	5.70(1)	
44A3	5	1125	2471	3.88(2)	
44B7	6	1125	2451	1.10(3)	
44A6	7	1125	2469	5.03(1)	
43A4	8	1125	2462	1.12(3)	
44A2	8	1125	2487	1.08(3)	
43A3	9	1125	2462	5.65(1)	
44A	9M	1125	2487	5.12(2)	Modified: plain water all tanks
49A2	10	1125	2473	6.55(1)	
49A3	11	1125	2467	4.79(2)	
49A10	13	1125	2458	7.24(1)	
44B10	15	1125	2469	1.26(3)	
44B12	16	1125	2480	5.98(1)	
49A7	17	1125	2455	1.34(3)	
49A6	18	1125	2461	6.83(1)	
40A5	1	1500	2685	3.86(1)	
46A1	1M	1500	2720	4.64(1)	Modified: boroxo off top
46A4	1M	1500	2720	4.26(1)	Modified: boroxo off front
46A5	1M	1500	2720	5.19(1)	Modified: boroxo off back
40A4	2	1500	2685	3.18(2)	
50A	2	1500	2619	8.04(0)	Modified: inner filled borated water, 2-in. borated, 2-in. plain polyethylene cavity lining
50A2	2	1500	2618	4.88(1)	Modified: 2-in. borated polyethylene cavity lining
50A6	2	1500	2617	1.16(2)	Modified: 1-in. " " "
50A7	2	1500	2614	3.53(2)	
50A10	2M	1500	2611	5.07(1)	Modified: 2-in. polyethylene cavity lining
40A1	3	1500	2685	3.05(4)	

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. See text for discussion of units used with modified long counter.

Table 7.7. Slow-Neutron Flux (Multiplied by $4\pi R^2$) Within
Geometric No. 1. Radiation source: HPFR;
detector: BF_3 proportional counter.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Flux}^b$ (ft ² ·neutrons· cm ⁻² ·sec ⁻¹ ·W ⁻¹)	Notes
41A	1	500	2294	2.34(2)	Poor statistics: ~3 net counts/min
41A	2	500	2294	2.75(3)	
41A	3	500	2299	1.79(5)	
15A	1	1125	1938	2.92(2)	~7 net counts/min
15C	2	1125	1929	2.77(3)	
16A	3	1125	1929	4.37(5)	
17C1	1	1125	2564	1.92(2)	~2 net counts/min
17C2	1M	1125	2586	1.13(3)	
17B5	1M	1125	2559	2.28(3)	
18B1	1M	1125	2574	3.15(1)	< 1 net count/min
20A1	1C	1125	2489	3.07(2)	
17B4	2	1125	2559	3.78(3)	
18A1	2	1125	2597	1.40(3)	~13 net counts/min
20A2	2C	1125	2516	1.55(3)	
43A	2M	1125	2474	1.46(3)	
17B23	3	1125	2560	2.61(4)	Thermal flux; C.R. = 1.18
18A2	3	1125	2606	2.72(4)	
20A3	3	1125	2570	1.98(5)	
43A4	4	1125	2462	1.81(2)	~4 net counts/min
43A5	8	1125	2462	5.39(3)	
43A1	9	1125	2462	1.83(2)	
40A6	1	1500	2684	2.94(1)	Thermal flux; C.R. = 1.78 (low counts)
40A3	2	1500	2685	7.16(2)	
40A2	3	1500	2685	4.17(4)	

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. All fluxes are bare BF_3 measurements, except when specifically noted as thermal (2200 m/sec) flux.

A few gamma-ray dose rates were measured within various configurations of Geometric No. 1 with one of the proved ASD's, and are presented in Table 7.8. It should be noted that none of the ASD data of this report have been corrected for counts due to neutrons, an effect which may be as small as $\sim 2\%$ or as great as 15 to 20%.

A final result from the examination of Geometric No. 1 is contained in Figs. 7.6, 7.7, and 7.8. Figure 7.6 displays the variation of dose rate (MLC pseudo dose) within configuration 1 of Geometric No. 1 with azimuthal angle. This plot was obtained by positioning the MLC centrally in the shield, then circling the shield in a horizontal midplane with a Po-Be source, maintaining a constant radial distance of 181 cm from the center of the shield. The dose received through the corners is apparently less than a third of that received when the source was opposite the center of one of the shield faces.

An identical exercise was performed with configuration 3 of Geometric No. 1, this time using an ASD and a radium source. The result is shown in Fig. 7.7. Again the characteristic quatrefoil pattern is evident, with the dose rate through the corners nearly seven times less than that through the center of the face. To complete this experiment, the identical setup was again employed, but the source was circled in a plane projected at 45° from the top edge of the cubical shield. The results are shown in Fig. 7.8. The maximum of the curve corresponds, as it should, to the corner locations or minima of Fig. 7.7, with the minima of Fig. 7.8 being achieved at 45° points, where the maximum line-of-sight shielding must be penetrated.

7.2.2. Geometric No. 2 (Medium Cube). This shield was again a cube, but with interior dimensions of $24 \times 24 \times 24$ in. Its walls, from interior to exterior, consisted of 4 in. of iron, an 11-in. void, 1 in. of iron, $1/8$ in. of aluminum, and $1/4$ in. of Boroxy, a total thickness of $16-3/8$ in. This shield is shown in Fig. 7.9. On its stand, as shown, the bottom was $22-1/2$ in. above the ground, its center $50-1/2$ in.

Data were obtained from this shield only with the MLC, ASD, and BF_3 detectors. Five configurations were tested. Configuration 1 consisted of the shield as described above with the 11-in. void filled with borated

Table 7.8. Gamma-Ray Dose Rates Within Geometric 1
for Various Configurations

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose}$ (ft ² ·mr·hr ⁻¹ ·W ⁻¹)	Notes
				(x 10 ²)	
18A3	1	1125	2577	2.18	
18B1	1	1125	2574	2.08	
18A1	2	1125	2606	3.73	
18A2	3	1125	2594	36.8	
20A1	1C	1125	2489	2.59	Corner orientation
20A2	2C	1125	2516	4.59	Corner orientation; questionable data be- cause of uncertain temperature correction
20A3	3C	1125	2570	43.1	" " " "

a. Normalized to air density of 1.0×10^{-3} g/cm³.

Fig. 7.6

Fig. 7.7

Fig. 7.8

Fig. 7.9

water. Emptying the borated water produced Configuration 2. For Configuration 3 the void was filled with plain water, and Configuration 4 was identical with 1 except that the boroxo was removed from all sides. To produce Configuration 5 the water was emptied from 4.

The fast-neutron pseudo dose rates inside Geometric No. 2 are presented in Table 7.9. Slow-neutron fluxes are shown in Table 7.10. Again the slow-neutron data suffers from count rates in some cases barely above background.

Gamma-ray dose rates are given in Table 7.11 for Configurations 1 and 2 only.

7.2.3. Geometric No. 3 (Small Cube). Geometric No. 3 had interior dimensions of 6 x 6 x 6 in. Its walls consisted of 4 in. of iron on the inside, followed by an 11-in. void, 1 in. of iron, 1/8 in. of aluminum, and 1/4 in. of Boroxo. Because of the small size of the cavity, the only data obtained with this shield were three fast-neutron dose rates measured with the FND and a few gamma-ray dose measurements. With the void filled with borated water, a $4\pi R^2$ fast-neutron dose rate of $1.97 \times 10^1 \text{ ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$ was observed at a slant range of 2548 ft. Emptying the water from the void increased the fast-neutron dose rate to a value of 2.18×10^3 . The reactor height was 1125 ft.

The gamma-ray data is presented in Table 7.12.

7.2.4. Geometric No. 4 (Thick-Walled Sphere). This shield, shown in Fig. 7.10, had an inner diameter of 24 in., and its walls, from interior to exterior, consisted of 4 in. of iron, a 4-1/4-in. void, 1/4 in. of iron, a 6-1/2 in. void, 1 in. of iron, and 1/4 in. of Boroxo. Separate controls were provided so that either top or bottom hemispheres could be individually filled with water. The seven configurations of this shield which were tested are diagrammed in Fig. 7.11.

The fast-neutron dose rates measured within Geometric No. 4 by the FND are shown in Table 7.13, while the fast-neutron pseudo dose rates measured in Geometric No. 4 by the MLC are presented in Table 7.14. Table 7.15 shows the slow-neutron fluxes measured by a bare BF₃ detector within Geometric

Table 7.9. Fast-Neutron Pseudo Dose Rate (Multiplied by $4\pi R^2$)
 Within Geometric No. 2. Radiation source: HPFR;
 detector: modified long counter

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2$ x Pseudo Dose Rate (ft ² .erg. g ⁻¹ . hr ⁻¹ .W ⁻¹)	Notes
41B	1	500	2343	7.67(1)	
41B2	2	500	2286	4.47(4)	
15A	1	1125	1938	1.13(2)	
15C	1M	1125	1929	1.33(2)	Corner orientation
16A	1M	1125	1929	1.26(2)	Corner orientation
14C	2	1125	1897	1.07(5)	
18A2	1M	1125	2605	5.13(1)	Corner orientation
24A3	1	1125	2522	5.98(1)	
25B1	1	1125	2536	6.36(1)	Corner orientation
44A1	1M	1125	2487	7.09(1)	Boroxy off top
44A2	1M	1125	2487	6.85(1)	Boroxy off front
44A	1M	1125	2487	6.69(1)	Boroxy off bottom
44A	1M	1125	2486	6.91(1)	Boroxy off back
44A	1M	1125	2478	6.82(1)	Boroxy off sides
44B3	1M	1125	2451	6.72(0)	2-in. borated polyethylene cavity lining
17B3	1	1125	2559	7.20(1)	
17B2	2	1125	2560	4.64(4)	
17C2	2M	1125	2586	4.25(4)	Shield oriented 45° left
18A1	2M	1125	2606	4.08(4)	Corner orientation
18B1	2M	1125	2574	4.46(4)	Corner orientation
44B6	4M	1125	2450	7.87(0)	2-in. borated polyethylene cavity lining
44B6	4	1125	2569	7.58(1)	
44B7	5	1125	2487	5.20(4)	
45A1	2	1125	2475	5.07(4)	
46B	2	1125	2533	4.30(4)	Shield on ground
46B	2	1125	2533	4.19(4)	
40A	1	1500	2684	5.56(1)	
40B2	2	1500	2711	3.79(4)	
47A	2M	1500	2634	3.06(4)	Shield over Geo. 8, no bottom
47A	2M	1500	2635	3.76(4)	Shield over Geo. 8, 4-in. iron bottom, boroxy off top
47A	2M	1500	2638	3.50(4)	Shield over Geo. 8, 4-in. iron bottom, boroxy on top
47A	2M	1500	2638	3.15(4)	Shield over Geo. 8, 4-in. borated polyethylene bottom, boroxy on top

a. Normalized to air density of 1.0×10^{-3} g/cm³.

Table 7.10. Slow-Neutron Flux (Multiplied by $4\pi R^2$) Within
Geometric No. 2. Radiation source: HPFR;
detector: BF_3 proportional counter.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Flux}^b$ (ft ² ·neutrons· cm ⁻² ·sec ⁻¹ ·W ⁻¹)	Notes
12C	1	300	1577	9.84(2)	Poor statistics: ~26 counts/min
12D	1	300	1577	6.48(2)	Poor statistics: ~17 counts/min
41B	2	500	2332	7.25(2)	Poor statistics: ~10 counts/min
41B	2	500	2298	2.30(5)	
13B	1	1125	1925	6.16(2)	Poor statistics: ~10 counts/min
13C	2	1125	1916	5.86(5)	
13D	2	1125	1916	5.88(5)	
13E	2	1125	1915	5.81(5)	
17A1	1	1125	2634	2.87(2)	Poor statistics: ~3 counts/min
18B3	1C	1125	2574	8.25(1)	Corner orientation. Poor statistics: ~3 counts/min
19A1	1C	1125	2574	3.8(0)	Corner orientation. Thermal flux; C.R. = 1.35; poor statistics; ~3 counts/min
20A1	1C	1125	2490	2.05(2)	Corner orientation. Poor statistics: ~2 counts/min
17A2	2	1125	2552	2.53(5)	
17B1	2	1125	2560	1.87(5)	Cadmium-covered BF_3 flux
17A2-B1	2	1125	2556	6.6(4)	Thermal flux: C.R. = 1.35
18B2	2C	1125	2574	2.25(5)	Corner orientation
19A2	2C	1125	2548	1.98(5)	Corner orientation; cadmium-covered flux
	2C	1125	2561	2.7(4)	Thermal flux (from above): C.R. = 1.14
20A2	2C	1125	2500	2.69(5)	Corner orientation
40B1	2	1500	2708	2.27(5)	Thermal flux; C.R. = 1.34

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. Bare BF_3 measurements except when noted. "Thermal" flux is 2200 m/sec flux.

Table 7.11. Gamma-Ray Dose Rates Within Geometric 2
for Various Configurations

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose}$ (ft ² .mr.hr ⁻¹ .W ⁻¹)	Notes
(x 10 ²)					
13B	1	1125	1925	7.96	
13C	2	1125	1916	125.	
13D	2	1125	1916	130.	
13E	2	1125	1915	127.	
17B1	2	1125	2560	53.3	
18B3	1C	1125	2574	2.55	Corner orientation
18B2	2C	1125	2574	40.8	" "

a. Normalized to air density of 1.0×10^{-3} g/cm³.

Table 7.12. Gamma-Ray Rates Within Geometric 3
for Various Configurations

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose}$ (ft ² .erg.g ⁻¹ .hr ⁻¹ .W ⁻¹)	Notes
(x 10 ²)					
17A2	1	1125	2560	4.05	
18B1	1	1125	2574	3.52	
17B1	1C	1125	2560	3.50	Corner orientation
18B2	1C	1125	2574	3.40	" "
17A1	2	1125	2542	66.7	
18A2	2	1125	2605	54.6	
17B2	2C	1125	2559	66.0	Corner orientation
17C1	4	1125	2559	6.63	
17C3	5	1125	2586	79.4	
18A1	5	1125	2606	81.0	

a. Normalized to air density of 1.0×10^{-3} g/cm³.

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Fig. 7.10

Fig. 7.11

Table 7.13. Fast-Neutron Dose Rates (Multiplied by $4\pi R^2$)
 Within Geometric No. 4. Radiation
 source: HPRR; detector: FND

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2$ x Dose Rate ($\text{ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$)	Notes
4OB1	3	1500	2685	1.77(3)	
4OB1	3	1500	2685	2.11(3) ^b	
4OB2	2	1500	2711	1.30(2)	
4OB2	2	1500	2711	1.34(2) ^b	
41B4	3	500	2298	2.32(3)	Doubtful; long delay following calibration
41B4	3	500	2298	2.68(3) ^b	" " " "
41B	2	500	2309	3.72(2)	" " " "
41B	2	500	2309	3.82(2) ^b	" " " "

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. Value with vacuum tube integrator.

Table 7.14. Fast-Neutron Pseudo Dose Rates (Multiplied by $4\pi R^2$)
in Geometric No. 4. Radiation source: HP RR;
detector: modified long counter.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2$ x Dose Rate (ft ² .erg.g ⁻¹ . hr ⁻¹ .W ⁻¹)	Notes
12B	4M	300	1202	2.57(5)	Bottom half of sphere only. Water in both compartments
12C	4	300	1201	1.61(3)	
12D	4	300	1201	1.60(3)	
41B	1	500	2320	1.37(2)	
41B	2	500	2332	1.96(3)	
41B	3	500	2343	4.32(4)	
13A	4	1125	1644	1.37(3)	Questionable
17B2	1	1125	2559	1.38(2)	
18A2	1	1125	2605	1.46(2)	
17B1	2	1125	2560	2.02(3)	
18A1	2	1125	2606	1.96(3)	
17A2	3	1125	2552	4.52(4)	
18A3	3	1125	2594	4.16(4)	
17A1	4	1125	2536	1.45(2)	
18B1	4	1125	2574	1.29(2)	
17B3	10	1125	2559	2.76(4)	
17B4	11	1125	2558	1.59(4)	
40B4	1	1500	2737	3.97(2)	
40B5	2	1500	2738	1.59(3)	
40B6	3	1500	2738	3.82(4)	

a. Normalized to air density of 1.0×10^{-3} g/cm³.

Table 7.15. Slow-Neutron Flux (Multiplied by $4\pi R^2$) Within
Geometric No. 4. Radiation source: HP RR;
detector: BF_3 proportional counter

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Flux}^b$ (ft ² ·neutrons· cm ⁻² ·sec ⁻¹ ·W ⁻¹)	Notes
20A6	1	1125	2591	2.35(2)	Poor statistics: ~3 counts/min
20A5	2	1125	2570	9.99(3)	
20A4	3	1125	2553	2.31(5)	
20A3	4	1125	2536	2.84(2)	Poor statistics: ~3 counts/min
25B4	10	1125	2565	1.38(5)	
25B3	11	1125	2542	1.15(5)	

a. Normalized to air density of 1.0×10^{-3} g/cm³.

b. All data from bare BF_3 measurements.

No. 4. Counting statistics within the configurations containing either plain or borated water are extremely poor.

The "questionable" designation on the gamma-ray dose rates shown in Table 7.16 stems from the application of corrections of from 8 to 14% to the raw data based on estimates of instrument temperatures.

7.2.5. Geometric No. 5. Shield No. 5 was a thin-walled sphere with the same interior diameter, 24 in., as the thick-walled sphere. Its wall, however, from interior to exterior, consisted of only 1 in. of iron, a 2-in. void, 1 in. of iron, and 1/4 in. of Borox. Configuration 1 for this shield existed when the void was filled with borated water. Emptying the water produced Configuration 2, while Configuration 3 saw the void filled with plain water. For Configuration 4 the lower hemisphere was filled with borated water and the upper hemisphere was empty, while for Configuration 5 the upper was filled with borated water and the lower was empty. Fast-neutron dose rates within Configurations 1 through 4 are shown in Table 7.17. Each measurement was made simultaneously with the transistorized integrator and the LITSF vacuum tube integrator. The results are essentially in agreement within the experimental errors.

Table 7.18 gives the fast-neutron pseudo dose rates measured with the MLC. Some feeling for the way the fast-neutron flux is distributed can be gained from a comparison of the data from Configurations 2, 4, and 5. The pseudo dose rate within 2 (void empty) is roughly 40% greater than the dose when the configuration's lower hemisphere is filled with borated water, but is about double the dose rate observed when its upper hemisphere contained the borated water.

Slow-neutron fluxes within the thin-walled sphere are shown in Table 7.19. The apparent anomaly in the values for Configurations 2 and 4 (the flux with borated water in the lower half of the shield is greater than that with the void empty) is probably only an indication of the errors in the data.

Table 7.20 gives the gamma-ray dose rates measured within Geometric No. 5.

Table 7.16. Gamma-Ray Dose Rates Within Geometric 4
Various Configurations

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose}$ (ft ² .mr.hr ⁻¹ .W ⁻¹)	Notes
				(x10 ²)	
20A6	1	1125	2591	31.8	Questionable: Dubious temperature data
20A5	2	1125	2570	39.2	" " "
20A4	3	1125	2553	86.1	" " "
20A3	4	1125	2516	44.6	" " "
18B4	4 Mod	1125	2581	217.	Top of Geo. No. 4 removed

a. Normalized to air density of 1.0×10^{-3} g/cm³.

Table 7.17. Fast-Neutron Dose Rates (Multiplied by $4\pi R^2$)
Within Geometric No. 5. Radiation source:
HPRR; detector: FND

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose Rate}$ (ft ² .erg.g ⁻¹ .hr ⁻¹ .W ⁻¹)	Notes
40B6	1	1500	2738	8.21(2)	
40B6	1	1500	2738	9.42(2) ^b	
40B5	2	1500	2738	2.66(3)	
40B5	2	1500	2738	2.91(3) ^b	
41A1	1	500	2299	1.25(3)	
41A1	1	500	2299	1.31(3) ^b	
41A2	4	500	2292	2.88(3)	Questionable data: 4-hr delay after cal.
41A2	4	500	2292	3.04(3) ^b	" " " " "
41A3	2	500	2292	3.65(3)	" " " " "
41A3	2	500	2292	3.96(3) ^b	" " " " "
41B	3	500	2285	1.08(3)	Questionable data: 5-hr delay after cal.
41B	3	500	2285	1.10(3) ^b	" " " " "

a. Normalized to air density of 1.0×10^{-3} g/cm³.

b. Value with vacuum tube integrator.

Table 7.18. Fast-Neutron Pseudo Dose Rates (Multiplied by $4\pi R^2$)
in Geometric No. 5. Radiation source: HPRR;
detector: modified long counter.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2$ x Dose Rate (ft ² .erg.g ⁻¹ .hr ⁻¹ .W ⁻¹)	Notes
41B	1	500	2287	1.40(4)	
41B	2	500	2286	6.57(4)	
14C	2	1125	1610	5.99(4)	
17C2	1	1125	2586	1.44(4)	
17C4	2	1125	2586	6.41(4)	
17B4	3	1125	2559	2.13(4)	
17C1	4	1125	2559	4.63(4)	
17C3	5	1125	2586	3.20(4)	
40B1	1	1500	2685	1.20(4)	
40A	2	1500	2684	6.21(4)	

a. Normalized to air density of 1.0×10^{-3} g/cm³.

Table 7.19. Slow-Neutron Flux (Multiplied by $4\pi R^2$) Within
Geometric No. 5. Radiation Source: HPRR;
detector: BF₃ proportional counter.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2$ x Flux ^b (ft ² .neutrons.cm ⁻² .sec ⁻¹ .W ⁻¹)	Notes
19A3	1	1125	2536	3.40(5)	
19A1	2	1125	2524	5.87(5)	
19A5	2	1125	2572	5.58(5)	
19A2	4	1125	2548	5.93(5)	
19A4	5	1125	2572	5.04(5)	

a. Normalized to air density of 1.0×10^{-3} g/cm³.

b. All data from bare BF₃ measurements.

Table 7.20. Gamma-Ray Dose Rates Within Geometric 5
Various Configurations

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose}$ (ft ² ·mr·hr ⁻¹ ·W ⁻¹)	Notes
				(x 10 ²)	
19A3	1	1125	2560	103.	
19A5	2	1125	2572	139.	
19A	2	1125	2524	148.	
19A	2	1125	2548	138.	
19A2	4	1125	2548	130.	
19A4	5	1125	2572	140.	

a. Normalized to air density of 1.0×10^{-3} g/cm³.

7.2.6. Geometric No. 6. Shield No. 6 was a hollow-walled, rectangular parallelepiped, with interior dimensions of $8\frac{1}{2} \times 17 \times 17$ in. Its base was a 2 or 4 in. thick plate, covered on the outside with $\frac{1}{4}$ in. of Boroxy. All other walls, from interior to exterior, consisted of 1 in. of iron, a 2-in. void, 1 in. of iron, and the usual $\frac{1}{4}$ in. of Boroxy. The 12 wall configurations possible are diagrammed in Fig. 7.12, while a photograph of the shield is shown in Fig. 7.13. Fast-neutron dose rates are shown in Table 7.21, slow-neutron fluxes in Table 7.22. No gamma-ray dose rates were measured.

7.2.7. Geometric No. 7. This shield was a rectangular parallelepiped, of the same interior dimensions as No. 6, but having 4-in.-thick solid iron walls covered with $\frac{1}{4}$ in. of boroxy. Both 2-in. and 4-in.-thick iron bases were available, and the possible configurations were: (1) boroxy on, 4-in.-thick base; (2) boroxy off, 4-in.-thick base; (3) boroxy on, 2-in.-thick base; and (4) boroxy off, 2-in.-thick base. Fast-neutron dose rates within this shield are shown in Table 7.23. Although several measurements are listed as doubtful because of a delay between calibration and data taking, the results seem to be consistent with the expected effects of the variations in the shield.

Slow-neutron fluxes in Geometric No. 7 are shown in Table 24. A comparison of the data within Configuration 4 with that within Configuration 3 shows that the boroxy reduces the slow-neutron flux by about a factor of 2.

No gamma-ray dose rates were obtained in Geometric No. 7.

7.2.8. Geometric No. 8. "Shield No. 8" actually identified a pair of shields, each a cylindrical, concrete-lined hole in the ground. One of these holes was dug 1125 ft from the reactor tower, the second at 2175 ft. Each was 8 ft deep and 4 ft in diameter, with a precast concrete lip placed flush with the ground surface. Rectangular concrete slabs, 6 and 12 in. thick, were available for capping the holes. Figure 7.14 shows the top of one of the holes. The wing-like objects attached to the axial pipe were used to support the detectors.

CONFIGURATION NO.	1 in.	2 in.	1 in.	1/4 in. BOROX	
1	IRON	BORATED WATER	IRON	ON	2-in. BASE PLATE
2	IRON	EMPTY	IRON	ON	
3	IRON	PLAIN WATER	IRON	ON	
4	IRON	BORATED WATER	IRON	OFF	
5	IRON	EMPTY	IRON	OFF	
6	IRON	PLAIN WATER	IRON	OFF	4-in. BASE PLATE
7	IRON	BORATED WATER	IRON	ON	
8	IRON	EMPTY	IRON	ON	
9	IRON	PLAIN WATER	IRON	ON	
10	IRON	BORATED WATER	IRON	OFF	
11	IRON	EMPTY	IRON	OFF	
12	IRON	PLAIN WATER	IRON	OFF	

Fig. 7.12

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Fig. 7.13

Table 7.21. Fast-Neutron Dose Rates (Multiplied by $4\pi R^2$)
Within Geometric No. 6. Radiation source:
HPRR; detector: FND.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose Rate}$ ($\text{ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$)	Notes
20A2	5	1125	2489	3.85(3)	
20A3	4	1125	2493	1.22(3)	
20A4	1	1125	2516	1.14(3)	
20A5	1C	1125	2536	1.14(3)	Oriented 45° left of zero line
20A6	1C	1125	2553	1.10(3)	Same as A5 but detector at 0° Doubtful data. ^b
20A7	2C	1125	2570	3.00(3)	Doubtful data. ^b
42A1	2	1125	2445	3.41(3)	
42A1	2	1125	2445	3.76(3) ^c	
42A2	3	1125	2504	1.21(3)	
42A2	3	1125	2504	1.32(3) ^c	
42A3	6	1125	2504	1.36(3)	
42A3	6	1125	2504	1.43(3) ^c	
49A2	7	1125	2473	1.25(3)	
49A2	7	1125	2473	1.25(3) ^c	
49A3	8	1125	2470	3.36(3)	
49A3	8	1125	2470	3.41(3) ^c	
49A6	9	1125	2464	1.17(3)	
49A6	9	1125	2464	1.24(3) ^c	
49A7	12	1125	2461	1.34(3)	
49A7	12	1125	2461	1.38(3) ^c	
49A10	11	1125	2455	4.07(3)	Doubtful data. ^b
49A10	11	1125	2455	4.28(3) ^c	" "
49A11	10	1125	2454	1.38(3)	" "
49A11	10	1125	2454	1.49(3) ^c	" "

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. Data doubtful because of long delay (> 2 hr) between calibration and measurement.

c. Value with vacuum tube integrator.

Table 7.22. Slow-Neutron Flux (Multiplied by $4\pi R^2$) Within
Geometric No. 6. Radiation source: HP RR;
detector: BF_3 proportional counter.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Flux}^b$ (ft ² ·neutrons. cm ⁻² ·sec ⁻¹ ·W ⁻¹)	Notes
42A11	1	1125	2492	2.58(5)	Thermal flux: C.R. 2.64
42A10	1C	1125	2494	4.12(5)	Corner orientation
42A7	2	1125	2497	5.57(5)	
42A9	2C	1125	2491	4.96(5)	Corner orientation
42A6	3	1125	2497	1.05(7)	
42A13	4	1125	2490	2.34(5)	
42A8	5	1125	2496	1.34(6)	
42A5	6	1125	2503	1.88(6)	
49A1	7	1125	2473	3.90(5)	
49A4	8	1125	2448	5.29(5)	
49A5	9	1125	2464	6.17(5)	Thermal flux: C.R. = 5.11
49A12	10	1125	2455	2.47(5)	Thermal flux: C.R. = 2.39
49A9	11	1125	2455	6.13(5)	Thermal flux: C.R. = 2.09
49A8	12	1125	2461	8.63(5)	Thermal flux: C.R. = 5.46

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. All fluxes are bare BF_3 measurements except when specifically noted as thermal (2200 m/sec) flux.

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Table 7.23. Fast-Neutron Dose Rates (Multiplied by $4\pi R^2$)
Within Geometric No. 7. Radiation
source: HPFR; detector: FND.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2$ x Dose Rate ($\text{ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$)	Notes
20A8	1	1125	2591	2.11(3) ^b	Doubtful
42A5	1C	1125	2500	2.20(3)	Corner alignment
42A5	1C	1125	2500	2.45(3) ^b	" "
42A6	1	1125	2500	2.17(3)	
42A6	1	1125	2500	2.51(3) ^b	
42A7	2	1125	2496	2.71(3)	Questionable, delay after calibration
42A7	2	1125	2496	3.15(3) ^b	" " "
42A8	3	1125	2494	2.32(3)	" " "
42A8	3	1125	2494	2.67(3) ^b	" " "
46B2	4	1125	2533	2.79(3)	
46B2	4	1125	2533	2.96(3) ^b	

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. Value with vacuum tube integrator.

Table 7.24. Slow-Neutron Flux (Multiplied by $4\pi R^2$) Within
Geometric No. 7. Radiation source: HP RR;
detector: BF_3 proportional counter.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Flux}^b$ ($\text{ft}^2 \cdot \text{neutrons} \cdot$ $\text{cm}^{-2} \cdot \text{sec}^{-1} \cdot \text{W}^{-1}$)	Notes
42A2	1	1125	2504	1.03(5)	Thermal flux; C.R. = 1.42
42A4	1C	1125	2503	7.00(4)	Thermal flux; C.R. = 1.30; corner orientation
42A1	3	1125	2505	3.60(5)	
46B	4	1125	2533	6.12(5)	
46B	4	1125	2533	7.12(5)	

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

b. All fluxes are bare BF_3 measurements.

Fig. 7.14

No fast-neutron data were obtained within the hole at 1125 ft. Thermal (2200 m/sec) fluxes and bare BF_3 fluxes are plotted as a function of distance below the surface in Fig. 7.15, for various conditions of reactor height and shield configuration. The values written on some of the curves are cadmium ratios, defined as the ratio of the bare BF_3 flux to the cadmium-covered BF_3 flux. The ratio is seen to increase with increasing depth in the hole and with increasing thickness of the concrete hole cap. The slopes of the flux-vs-depth curves also seem to increase with increased thickness of the concrete cap.

Fast-neutron dose rates observed in the hole at 2175 ft are plotted vs depth below ground, for various reactor heights, in Fig. 7.16.

Fast-neutron pseudo dose rates measured by the MLC within the hole at 2175 ft are plotted vs depth below ground level in Fig. 7.17. The sensitivity of the MLC made possible measurements below as much as 12 in. of concrete. With the reactor at 1125 ft, a 6-in. slab of concrete attenuates the pseudo dose by a factor of about 6 at a depth of 5 ft in the hole, while a 12-in. thickness of concrete attenuates the unshielded dose by a factor of ~ 44 . It is also noticeable in these results that for the unshielded case reactor height had no apparent effect except for the lowest height. The curves for 300, 500, 1125, and 1500 ft are essentially identical.

7.2.9. The OTAC Pod. The Ordnance Tank-Automotive Command (OTAC) Radiological Pod has been elsewhere described.¹ The experiments with this shield did not involve any modifications to the pod, but rather consisted of measurements of flux and dose at various locations within the shield. These locations are noted in the tabulations of data. Fast-neutron dose rates within the shield were so low as to preclude employment of the FND. The fast-neutron pseudo dose rates observed within the OTAC pod are shown in Table 7.25.

Only two slow-neutron flux measurements, both representing bare BF_3 data, were made within this shield. With the pod at a slant range of

Fig. 7.15

Fig. 7.16

Fig. 7.17

Table 7.25. Fast-Neutron Pseudo-Dose Rates (Multiplied by $4\pi R^2$)
in the OTAC Pod. Radiation source: HPFR;
detector: modified long counter.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose Rate}$ ($\text{ft}^2 \cdot \text{erg} \cdot \text{g}^{-1} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$)	Notes
12A	1	300	1227	1.27(2)	
11E	2	300	1229	2.00(2)	
11D	3	300	1222	1.31(2)	
13C	1	1125	1626	1.36(2)	
13B	2	1125	1634	1.80(2)	
13D	3	1125	1626	1.34(2)	
15A	4	1125	1645	1.52(2)	
16A	5	1125	1638	1.56(2)	
15C	6	1125	1637	1.93(2)	
13E	3M	1125	1626	1.30(2)	Detector 41-3/8 in. forward of rear wall

a. Normalized to air density of $1.0 \times 10^{-3} \text{ g/cm}^3$.

1222 ft, the $4\pi R^2$ flux at a point 77-1/4 in. from the rear wall was $2.44 \times 10^3 \text{ ft}^2 \cdot \text{neutrons} \cdot \text{cm}^{-2} \cdot \text{sec}^{-1} \cdot \text{W}^{-1}$. With a slant range of 1229 ft, the $4\pi R^2$ flux at 16 in. from the rear wall was $6.38 \times 10^3 \text{ ft}^2 \cdot \text{neutrons} \cdot \text{cm}^{-2} \cdot \text{sec}^{-1} \cdot \text{W}^{-1}$. For both measurements the reactor height was 300 ft, and the slant range has been normalized to an air density of $1.0 \times 10^{-3} \text{ g/cm}^3$. This normalization is responsible for the slight difference between the two ranges.

Some gamma-ray dose rate data were obtained inside the OTAC pod, are shown in Table 7.26. The free-field dose rate at the location of the pod was about 6.8 or $6.9 \times 10^4 \text{ ft}^2 \cdot \text{mr} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$ for a reactor height of 300 ft, slightly higher, 7.0 to $7.5 \times 10^4 \text{ ft}^2 \cdot \text{mr} \cdot \text{hr}^{-1} \cdot \text{W}^{-1}$, for a reactor height of 1125 ft. The dose rates shown thus represent attenuations of roughly a factor of 50 by the pod.

7.3. Measurements Using 1200-Curie Co⁶⁰ Source

We have previously noted that the intensity of the Co⁶⁰ source was too low to enable measurements within the various shields. Free-field traverses were possible, however, and the results are presented in Fig. 7.18. There does not seem to be a single exponential slope which fits all data. The relaxation length approximately fitting the lowest of the curves, that resulting from the 27-ft source height, is 608 ft, with the relaxation length apparently becoming slightly longer with increased source height. The rollover strongly evident in the early part of the curves for 1125 ft and 1500 ft source heights is believed due to the fact that the detector is seeing the shadow of the source container. During operation the container remained in position on the hoist, the source merely being lifted a few inches above it.

A comparison of the apparent relaxation length resulting from the 1.17- and 1.33-MeV Co⁶⁰ gamma rays with the much longer relaxation length resulting from the free-field measurements with the reactor as the source bluntly emphasizes the important contribution of the high-energy (3.3 to 10.8 MeV) air-capture nitrogen gamma rays to the free-field dose rate.

Some gamma-ray dose rate traverses employing the Co⁶⁰ source were feasible in Geometric No. 8, the 4-ft-diam, 8-ft-deep holes 1125 and 2175 ft

Table 7.26. Gamma-Ray Dose Rates (Multiplied by $4\pi R^2$) Within
the OTAC Pod. Radiation source: HPFR.
Detector 21-1/2 in. above floor of pod.

Run No.	Configuration No.	Reactor Height (ft)	Slant Range ^a (ft)	$4\pi R^2 \times \text{Dose}$ (ft ² .mr.hr ⁻¹ .W ⁻¹)	Notes
12B	1	300	1202	1.17(3)	Center of detection 49-3/4 in. from rear wall
12C	1	300	1201	1.73(3)	" " " " "
12D	1	300	1201	1.12(3)	" " " " "
14A	1	300	1218	1.03(3)	" " " " "
14B	1	300	1204	9.63(2)	Poor calibration
11D	2	300	1222	1.01(3)	Center of detection 81 in. from rear wall
11E	3	300	1229	1.72(3)	Center of detection 19 3/4 in. from rear wall
13A	1	1125	1644	1.36(3)	Center of detection 50 in. from rear wall
15A	1	1125	1645	1.92(3)	Pod rotated 90° to left
15C	1	1125	1637	1.34(3)	Center of detection 50 in. from rear wall
14C	3	1125	1610	1.08(3)	Center of detection 19-3/4 in. from rear wall

a. Normalized to air density of 1.0×10^{-3} g/cm³.

Fig. 7.18

from the source tower. A number of these, made in the 2175-ft distance hole, are shown in Fig. 7.19. Data obtained at a source height of 1500 ft were statistically meaningless because of the low source intensity.

Vertical traverses made in the 1125-ft-distant hole are shown in Fig. 7.20. The Co^{60} source was positioned at 27, 300, and 1125 ft for these data. Two runs are shown for the 1125-ft height. The lower of the pair represents the dose rate measured by one of the unmodified Blosser-Mustafa dosimeters; the higher of the curves was measured with one of the newer dosimeters. The difference was first thought to reflect a difference in the response of the two dosimeters. However, as shown by the curves discussed below, differences in the position of the center of detection of the two instruments were mainly responsible. The design of the housing of the newer instrument placed its center of detection ~ 24 cm toward the reactor from the center of the hole, whereas the center of detection of the old instrument was about even with the center. Each was also ~ 12 cm north or south of center. To evaluate the importance of these offsets, a series of horizontal traverses was run within the 1125-ft-distant hole. Figure 7.21 shows such a traverse, made with the Co^{60} source at a height of 300 ft. The plane in which the dosimeter was traversed was 26-1/4 in. below the surface of the ground. Positive abscissas are toward the source. A difference in dose rate by a factor of 6 appears in going from front to back of the uncapped hole. With the source at 1125 ft, traverses from front to back of the 1125-ft-distant hole were made at various depths within the hole. These curves are shown in Fig. 7.22, positive abscissas again being plotted toward the source. At depths of 6 and 26-3/4 in., the dosimeter can "see" the source over much of the traverse. At the 50-1/2 and 62 in. depths the source is never on the line-of-sight.

To complete the descriptions of the radiation field within the hole, traverses at right angles to the source-hole axis were run at 26-3/4 and 50-3/4 in. below ground level. These data are shown in Fig. 7.23.

7.4. Gamma-Ray Spectra (Co^{60} Source)

The effect of distance and source height upon the spectrum of gamma rays emitted by the 1200-curie Co^{60} source was investigated with the NaI(Tl)

Fig. 7.19

Fig. 7.20

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Fig. 7.21

Fig. 7.22

Fig. 7.23

spectrometer previously mentioned. Figure 7.24 shows a number of such spectra, measured at horizontal distances of 1125 and 2175 ft from the source tower and with source heights ranging from 27 to 1125 ft. The spectrum from a small Co^{60} calibration source, measured at a distance of 5.7 cm, is included for comparison purposes. The latter curve has been increased by a factor of 5 in order to bring it closer to the others.

The Co^{60} gamma-ray spectrum in free-field at 1125 ft and a source height of 1125 ft is compared to the spectrum observed at a depth of 26-3/4 in. in the 4-ft-diam, 8-ft-deep hole in Fig. 7.25. Some of the energy-degraded gamma rays seem to disappear at the shallow depth, resulting in comparatively more prominent 1.17- and 1.33-MeV peaks.

As a reminder, Configuration 3 of Geometric No. 1 consisted of the large cubical shield with boroxyl on but with all voids empty. The spectrum from the 1200-curie Co^{60} source was measured within this configuration for a variety of source heights at 1125 ft from the source tower and for a source height of 1125 ft at 2175 ft from the tower. The resulting curves are plotted in Fig. 7.26. In contrast to the free-field data earlier shown, which in general showed the two prominent peaks at 1.17 MeV and 1.33 MeV plus an almost smooth continuum of degraded gamma rays, the spectra within the shield show a ragged succession of low-energy peaks. Many of these peaks are roughly traceable from one curve to another, although small shifts in energy seem to occur. It should be noted, however, that count rates were generally low, ranging from a maximum of about 10^3 to a low of zero, making much of the data statistically unreliable.

Reference

1. R. E. Maerker, ORNL-3173 (August, 1961) (classified).

8. Errors

It is effectively an impossible task to assign valid and defensible limits of error to the data reported above. As we have earlier noted, the FND and BF_3 detectors suffered generally from low counting rates due to the low intensity at 2175 ft from the source tower. The MLC counting rates were generally fairly high, but the response of this instrument was directional and energy dependent. The questions posed as to the possible energy

Fig. 7.24

Fig. 7.25

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Fig. 7.26

and intensity dependence of the ASD's remain unanswered. A large amount of effort, however, has been expended to make the data reported as good as possible. The raw data have been examined, piece by piece, in conjunction with the instrument logs, and corrections have been made for changes in calibration of the detectors due to temperature shifts, background fluctuations, etc. Perhaps the most practical index of reliability lies in the multiple measurements which were made for many situations. By using this criterion, the bulk of the data seems to fall within limits of error of ±10%.

APPENDIX A

Calculation of Corrections to BREN Data

Data of the BREN experiment were obtained over a wide range of air temperatures, air densities, and reactor powers. In order to place the data within an internally consistent framework, corrections for these effects were computed and applied.

The bases for the corrections are the meteorological data recorded by the Yucca Flat station, temperature records at the BREN site, and the reactor power monitor data obtained from the master monitor established by BREN Program 1. All of these data have been reported elsewhere.¹

The initial step in the computation of the corrections was the calculation of the barometric pressure of the BREN site, using the Yucca Flat data. The following equation, a rearrangement of a standard formula,² was used:

$$P_1 = P_2 \left[\frac{A(1 + Bt) - H}{A(1 + Bt) + H} \right] \quad (1)$$

where

P_1 = barometric pressure at BREN site,

P_2 = barometric pressure at Yucca Flat,

$A = (1.6 \times 10^6)/30.48$: a constant from the original formula with a cm-to-ft conversion factor added,

$B = 4 \times 10^{-3}$: a temperature conversion constant from the original formula,

t = mean temperature ($^{\circ}\text{C}$) of the air at Yucca Flat and BREN,

$H = 316 \text{ ft} + (H_S + H_D)/2$,

H_S = source height above pad of reactor tower,

H_D = detector height above pad of reactor tower,

316 ft = difference in elevation between BREN and Yucca Flat.

The variation of neutron or gamma-ray relaxation length with air temperature and barometric pressure is given by

$$L_1 = L_0 \left(\frac{P_0 T_1}{T_0 P_1} \right) \quad (2)$$

in which L, T, and P are respectively the relaxation length, the air temperature in °K, and the barometric pressure.

For convenience in future manipulation, it was decided to normalize all data to an air density of 1.0 g/liter. From a reactor run during which the air density actually was 1.0 g/liter, values, for fast neutrons, of $L_0 = 738.5$ ft, $P_0 = 25.17$ in. of Hg, and $T_0 = 291^\circ\text{K}$ were obtained.

The master power monitor earlier noted was a conventional long counter originally located 2390 ft horizontally from the reactor tower. The count rate of the monitor was normalized to a reactor power of 1 W by comparison with the activity induced in sulfur plaques attached to the reactor during a fairly large number of its runs. The sulfur activity was related to the reactor power by a single run, No. 37A, during which the reactor power was measured by the method described by Mihalcz.³ Monitor counts were corrected to standard air density (1.0 g/liter) from the relation:

$$M_0 = M_1 \exp - \left[R_M (L_0^{-1} - L_1^{-1}) \right] \quad (3)$$

where

M_0 = monitor counts for an air density of 1.0 g/liter,

M_1 = actual count of monitor for a specified run,

R_M = slant range from reactor to monitor.

Late in the series of BREN experiments the master monitor was moved to a position 3,000 ft horizontally from the reactor tower. The monitor

counts were corrected for this shift by employing the following formula:

$$M_0 = M_1(3000) \left(\frac{R_{M(3000)}}{R_M} \right)^2 \exp \left[L_0^{-1} (R_{M(3000)} - R_M) \right] \quad (4)$$

where $M_1(3000)$ = monitor counts at the 3,000-ft position,

$R_{M(3000)}$ = slant range from the reactor to the monitor at the 3,000-ft position.

By using experimentally determined values of L_0 , P_0 , and T_0 recorded above, a quantity $G(H_S)$, the monitor counts per minute per watt, was computed for each source height, and is tabulated below. The errors shown are the standard deviation of the mean of several reactor runs; where no error is shown only a single run was available. The values shown are specified for a reactor operational time of 10 min from 1/e of full power to shutdown, with the reactor achieving full power on a period generally of about 30 sec.

Table A.1. The Quantity $G(H_S)$ for Various Reactor Heights

H_S , Reactor Height (ft)	$G(H_S)$ (counts.min ⁻¹ .W ⁻¹)
27	16.0 ± 6%
170	21.3
300	22.6 ± 3%
500	23.8 ± 6%
750	25.9
1125	21.0 ± 4%
1500	15.9 ± 6%

Finally, the dose rate of interest was computed from:

$$\text{Dose} = \frac{4\pi R_D^2 G(H_S) D}{(M_0/M_1)}$$

where

R_D = the slant range from the source to the detector,

D = the measured dose rate divided by the monitor count rate.

The development outlined above was written as an "ALGOL" code for the CDC-1604A computer, and all data were reduced by this code. For the Co^{60} source data, where obviously there were no monitor counts, the same code was applicable by simply setting $G(H_S)$ and M_0/M_1 equal to unity.

References

1. J. A. Auxier, F. F. Haywood, and L. W. Gilley, General Correlative Studies, Operation BREN, CEX-62.03.
2. C. D. Hodgman (ed.), Handbook of Chemistry and Physics, 34th ed., p. 2577, Chemical Rubber Publishing Co., Cleveland, Ohio (1952).
3. J. T. Mihalczo, Super Prompt Critical Behavior of an Unmoderated, Unreflected Uranium-Molybdenum Assembly, ORNL-TM-230 (May, 1962).

APPENDIX B

Analyses of Shielding Materials

Two types of borated polyethylene were used in the experiments detailed above. The first, called "Boroxyl," contained 22.67 wt% of boron, according to an analytical analysis made in July, 1962. A later spectrographic analysis gave a value of 21.0 wt% and showed small contaminations by various common elements. Of these, aluminum, 0.2 wt%; calcium, 0.42 wt%; and iron, 0.42 wt%, were the most plentiful.

The second type of borated polyethylene, used only with a few configurations and designated in the tabulations of results as simply "borated polyethylene," contained 0.87 wt% boron according to the analytic determination, or 0.93 wt% boron according to the spectrographic analysis.

Two analyses of the borated water used for constructing some of the configurations gave values of the boron content essentially in agreement: 1.39 and 1.41 wt%.

Analyses of the concrete pipe used to line the holes called Geometric No. 8, the dirt in the region, and the concrete used in the shields covering Geometric No. 8 are shown in Table B-1 below. Pipe sample no. 1 and the

Table B.1. Composition of Concretes and Dirt

Constituent	Composition (wt%)			
	Concrete Pipe			Concrete Slab
	Sample 1	Sample 2	Dirt	
H	0.12	0.48	0.70	0.12
O	42.8	37.0	43.5	43.8
C	9.03	4.30	0.16	9.26
Ca	24.4	20.0	1.50	33.9
Si	4.87	13.6	32.6	2.53
Mg	8.12	2.55	0.61	1.92
Al	1.11	4.35	6.32	1.11
Fe	0.74	1.95	1.78	0.60
CO ₃	45.1	21.5	0.81	46.3
Moisture	2.12	1.75	1.00	2.89

concrete slab were fabricated at ORNL; pipe sample no. 2 was from concrete pipe obtained at the Nevada site. The estimated density of the ORNL concrete, calculated from the mix recipe, is 2.3 g/cm^3 , that of the Nevada concrete is not known.

An analysis of the steel used in the construction of the geometrics is given in Table B-2.

Table B.2. Composition of Steels Used in Geometrics

Element	Amount Present (wt%)		
	1-in.-thick Plate	4-in.-thick Plate	1/2-in.-thick Plate
Fe	99.1	99.2	99.12
C	0.20	0.20	0.12
Si	0.06	0.21	0.20
S	0.034	0.021	0.028
Al	0.0056	0.0015	0.0052
Ca	0.062	0.0074	0.0040
Cr	0.025	0.018	0.025
Cu	0.0044	0.014	0.0053
Mg	0.0056	0.00052	0.00046
Mn	0.40	0.34	0.39
Ni	0.025	0.012	0.012

The density of the 1-in.-thick plate was 7.848 g/cm^3 and in addition to the elements tabulated above it contained 90 ppm of phosphorus. The 4-in.-thick plate had a density of 7.827 g/cm^3 , with 75 ppm phosphorus, while the density of the 1/2-in.-thick plate was 7.769 g/cm^3 . Its phosphorus content was 95 ppm.

Most of the analyses recorded above were made by G. R. Wilson of ORNL with others due to C. A. Pritchard and M. Murray also of ORNL.