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EXPERIMENTAL STUDIES OF NEUTRON
ATTENUATION IN NATURAL LITHIUM
HYDRIDE SHIELDS

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EXPERIMENTAL STUDIES OF NEUTRON
ATTENUATION IN NATURAL LITHIUM
HYDRIDE SHIELDS

By

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ABSTRACT

Results of a program to measure the attenuation of fission neutrons in lithium hydride at the SNAP Shield Test Facility are presented, with the methods used.

Experimental and analytical corrections to the data are given, to allow reduction to the infinite medium case.

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I. INTRODUCTION

Upon completion of the SNAP Shield Test Facility (STF) in 1962, one of the first tasks undertaken was the measurement of the attenuation of fission neutrons in lithium hydride. The program was initiated to provide the SNAP shielding program with clean geometry configuration data for testing analytical techniques and to provide good attenuation coefficient information for shield design. The attenuation coefficients extracted from these data are obtained from assumed kernels of the form $e^{-\mu r}/4\pi r^2$, this form being that utilized most frequently in SNAP shielding codes.

This report presents the experimental arrangement and equipment used in the LiH attenuation measurement program performed at the Shield Test Facility (STF), as well as the data taken. The measurements were carried out in a slab geometry, with a fission plate providing the neutron source.

The experimental program provided a basic series of measurements of neutron attenuation from which the attenuation coefficient, μ , was obtained over the range of shield thicknesses of interest to current SNAP shields.

II. DESCRIPTION OF EXPERIMENTAL PROGRAM

A. FACILITY

The facility consists of a 50-kw water cooled uranium-zirconium hydride reactor, a neutron thermal column, a fission source plate, and associated equipment. The uranium fission source plate is normally located on a wall of the underground shield test room in front of the 5-ft-square by 4-ft-long graphite thermal column, where all the shield measurements are performed (see Figure 1). A shield test carriage is used in conjunction with the fission source plate to support the shields during the experiments. Both the fission source plate and the carriage can be positioned remotely with the aid of a closed-circuit TV system. The fission plate can be driven into a lead shielding cask allowing personnel to assemble the shield experiments in a low radiation field.

B. TEST ARRANGEMENT

1. Fission-Source Plate

The fission plate is composed of a 24-in.-diameter by 0.060-in.-thick uranium disc (93.17% U^{235}) with a total U^{235} content of 7.8 kg. The thermal flux incident on the plate fits the equation $\phi = \cos(\pi R/50.8)$, where R is in inches (R = 0 is the center of the plate). The uranium disc is sandwiched between a 1-1/8-in. aluminum plate on the thermal column side and 1/8 in. aluminum and 1/8 in. boral (35% by weight B_4C , 65% Al) on the shield side. (See Figure 2.) The plate power was measured to be 59.9 watts corresponding to fast flux of 9.4×10^8 n/cm² sec at the center.² Figure 3 shows the fast neutron distribution at the surface of the fission plate as measured with sulfur threshold detectors.

2. Shield Test Carriage

The shield test carriage is a self-propelled cart 8 ft 6 in. in width and 10 ft in length. It is designed to hold test shields up to 5 ft square and 8 ft 6 in. in length. The carriage can be remotely operated from the reactor console and viewed continuously on television (see Figure 4).

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Figure 1. STF Architectural Sections and Details

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Figure 2. Fission Plate

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Figure 3. Fast Neutron Distribution at the Surface of the Fission Plate

A 7-ft-square by 1-ft-thick borated ordinary concrete "donut" can be fastened to the front of the carriage, if so desired, by the experimenter. A stepped 3-ft-square opening exists in the center of the donut and 1-3/8-in.-deep slots on the front of the donut provide space for the fission-plate-driven mechanism. The purpose of the donut is to remove the degraded neutrons that emerge from the edges of the fission plate and thermal column; these neutrons might otherwise interfere with a given experimental measurement. Test slabs can be placed within the center of the donut without interfering with the other test slabs which may be on the shield test carriage.

3. Lithium Hydride Shield Slabs

The lithium hydride shield slabs used in the experiments are made up of stepped blocks canned in aluminum, and were fabricated at Oak Ridge National Laboratory (ORNL). The blocks contain a stainless steel honeycomb which runs perpendicular to the neutron beam. The stainless steel content is specified to be less than 2-1/2% by weight; however, no information was supplied by ORNL as to the actual stainless steel content. The central block in each slab is

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Figure 4. Shield Test Carriage

approximately 15 in. square. The density and chemical composition of the central blocks as well as the maximum and minimum values for the other blocks are listed below.

TABLE 1
DENSITY OF LiH THREE-FOOT SLABS

Slab Thickness (in.)	Lithium Hydride Thickness (in.)	Aluminum Cladding Thickness (in.)	Density (Central Block) (gm/cc)
1	0.8768	0.1232	0.755
1	0.8768	0.1232	0.755
1-1/2	1.3768	0.1232	0.755
2	1.8768	0.1232	0.756
3	2.8768	0.1232	0.757
3	2.8768	0.1232	0.760

Densities of the other blocks ranged from 0.747 to 0.760. The central blocks for each of the 3-ft slabs came from the same hydriding batch, analysis of which is shown below.

TABLE 2
COMPOSITION OF LiH THREE-FOOT SLABS

Component	Central Block	Other Blocks	
		Maximum	Minimum
Li/LiH	0.8616	0.8703	0.8591
H (%)	12.37	12.47	12.14
OH	0.94	1.58	0.70
CO ₃	0.20	0.23	0.01
Cl (%)	0.07	0.38	0.01
Na (ppm)	44	89	35
Ba	20	50	10
Ca	375	641	213
Cr	8	15	1
Co	<20	--	--
Fe	60	300	<20
K (ppm)	33	89	18
Li ⁶ (%)	7.62	7.66	7.46

Density values for the five-foot slabs are shown in Table 3.

TABLE 3
DENSITY OF LiH FIVE-FOOT SLABS

Slab Thickness (in.)	Lithium Hydride Thickness (in.)	Aluminum Cladding Thickness (in.)	Density (Central Block) (gm/cc)
2	1-3/4	0.25	0.751
4	3-3/4	0.25	0.751
6	5-3/4	0.25	0.752
7	6-3/4	0.25	0.750
7	6-3/4	0.25	0.753

Densities of the other blocks ranged from 0.748 to 0.759. The central blocks came from two different hydriding batches, the analysis of which is shown in Table 4.

TABLE 4
COMPOSITION OF LiH FIVE-FOOT SLABS

Component	Central Block (2 in., 4 in., 7 in.)	Central Block (5 in., 7 in.)	Other Blocks	
			Maximum	Minimum
Li/LiH	0.8694	0.8648	0.8709	0.8535
H (%)	12.49	12.27	12.59	12.27
OH ↓	0.44	1.08	2.22	0.44
CO ₃ ↓	0.07	0.06	0.17	<0.01
Cl (%)	0.03	0.03	0.06	0
Na (ppm)	68	96	143	26
Ba ↓	<10	<10	--	--
Ca ↓	341	161	771	101
Cr ↓	1	12	12	<1
Co ↓	<20	<20	--	--
Fe ↓	<20	60	100	<20
K (ppm)	27	16	99	14
Li ⁶ (%)	7.35	7.50	7.69	7.31

C. DETECTORS

1. General

The fast neutron attenuation measurements were performed with a Hurst type dosimeter and a neptunium-237-lined fission chamber. The thermal and resonance neutron attenuation measurements were performed with gold and indium foils, a BF_3 counter, and a uranium-235 fission chamber. In all measurements, except those with foils, an approximately constant counting rate was maintained by raising the reactor power as the shield thickness was increased.

2. Hurst Dosimeter

The Hurst dosimeter used in these experiments is polyethylene-lined and cyclopropane-filled, with a built-in plutonium alpha source for calibration.³ A discussion of the limitations of the Hurst dosimeter for this type of measurement is found in Reference 4. Reference 5 discusses many of the fine points in the use of this chamber, such as temperature effects, gamma sensitivity, determination of the center of detection, and directional properties. Figure 5 shows the relative sensitivity as a function of energy.³ Data from a

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Figure 5. Hurst Dosimeter Response Function

Figure 6. Np^{237} Fission Chamber Response Function

measurement is a pulse height spectrum from a multichannel analyzer. Each pulse from the counter represents an amount of energy deposited in the counter proportional to the pulse height. The energy deposited in the chamber E_c is given by:

$$E_c = K \int_0^{\infty} C_e \cdot r \, dr$$

where C_e is the number of counts in the r th channel and K is determined by the alpha source calibration. This integral is evaluated for each shield thickness and the result is plotted versus shield thickness.

3. Neptunium-237 Fission Chamber

The neptunium fission chambers used in these experiments are standard EIA type 6376 chambers with a 2 mg/cm^2 coating of Np^{237} replacing the standard coating material. The chambers were manufactured by Lionel Electronics Labs using neptunium obtained from ORNL. The chambers were also sensitive to thermal neutrons as the coating material contained 5060 ppm plutonium and 340 ppm uranium. The chambers were surrounded by 2.18 gm/cm^2 of boron carbide to reduce the thermal neutron contribution. Figure 6 shows the relative sensitivity as a function of energy with and without boron calculated from cross-section data.⁶ The chambers were shielded from the fission plate by 3 in. of lead to reduce the background from gamma-induced fissions. Np^{237} has a photofission threshold of 4.99 Mev and a cross section of 20 millibarns at 6 Mev.⁷

4. Thermal Detectors

The U^{235} fission chamber and BF_3 proportional counter used in these experiments are standard chambers manufactured by Reuter-Stokes, Inc. The gold and indium foils used are 0.75 in. in diameter by 0.005 in. thick with a purity of 99.99%. They were beta-counted using a 2-in.-diameter by 0.5-mm-thick anthracene crystal as a detector. No attempt was made to determine the absolute intensity of the fast and thermal fluxes, or their ratios.

III. EXPERIMENTAL RESULTS

A. THERMAL NEUTRON MEASUREMENTS

The centerline attenuation of thermal and resonance neutrons in lithium hydride was measured using a BF_3 proportional counter, a U^{235} fission chamber, and bare gold and indium foils. The experimental geometry is shown in Figure 7. The shield-detector separation distance "A" for the two counters was approximately 1 in. The foils were placed between the lithium hydride slabs of a 3-ft-thick shield. A 2-ft concrete shield followed the lithium hydride.

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Figure 7. Experimental Setup — Centerline Attenuation Measurements

Figure 8 shows the results of these measurements. The break in the two chamber measurements is due to wall-scattered neutrons as is the tail on the foil curve. The gold foil measurements gave the same shape curve as the indium

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Figure 8. Centerline Distribution of Thermal Neutrons in LiH

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Figure 9. Transverse Map of Thermal Neutrons in LiH

results. Figure 9 shows a transverse map of thermal neutrons in lithium hydride using indium foils. Leakage neutrons leaving the shield are scattered from the walls of the test vault and some are returned to the shield and detector. At large shield thicknesses, these wall-scattered neutrons predominate at the detector and mask out the neutrons which come directly from the shield. Figure 9 indicates that for thermal neutrons the wall scatter background is equivalent to the leakage from 40 to 45 cm of LiH. These results indicate that measurements beyond 40 cm must be made with a collimator to minimize the effects of wall scatter.

B. FAST NEUTRON MEASUREMENTS

1. Measured Distribution

The centerline attenuation of fast neutrons in lithium hydride was measured in four different geometries: two fixed source-detector geometries with source-detector separations of 16 in. and 72 in., and two fixed shield-detector geometries with shield-detector separations of 2 in. and 44 in. The experimental arrangement is shown in Figure 7. The amount of lead used for each experiment is indicated along with the results of the measurements, shown in Figure 10. The lead was used to reduce the effects of gammas on the Hurst detector, but it was found not to be necessary in the range of thicknesses measured in these experiments. The tails on the curves are caused again by wall-scattered neutrons. The experimental uncertainty associated with each point is within 10%. This was determined by measuring a single configuration at several different reactor power levels on different days.

An attempt was made to determine experimentally the geometry corrections in the fixed shield-detector experiments. A measurement of dose as a function of distance from the fission plate was made using the Hurst dosimeter. The fixed shield-detector measurements were divided by the appropriate geometry factor at each point. Figure 11 shows the results of this reduction. The slope of the curves for 2-in. shield detector separation is 0.170 cm^{-1} , and 0.156 cm^{-1} for the 44-in. shield-detector separation. The slope of the fixed source-detector experiment with a 72-in. separation is 0.16 cm^{-1} and 0.17 cm^{-1} for the 16-in. separation.

Figure 10. Centerline Attenuation of Fast Neutrons in
LiH - Hurst Dosimeter, Four Geometries

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Figure 11. Experimental Geometry Correction to the Fixed Shield-Detector Separation Measurement

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 Figure 12. Attenuation of Fast Neutrons in LiH- Np^{237}
 Fission Chamber

An attempt was made to measure the fast neutron attenuation in lithium hydride with a Np^{237} fission chamber. Figure 12 compares the results of this measurement to those made with the Hurst dosimeter. The effects of shielding the chamber from thermal neutrons and gamma rays are also shown in Figure 12 for the end point. As the usefulness of this type of detector is severely limited by its sensitivity to thermal neutrons and gamma rays, care should be taken to understand this behavior before it is used.

2. Aluminum Correction

As the lithium hydride shields are clad in aluminum (the aluminum making up approximately 5% of the total shield thickness), an experiment was performed to determine the effects of the aluminum on the fast neutron attenuation results. The experimental setup is shown in Figure 13. The detector was not moved during this experiment, i. e., the source-detector distance was constant. The 3-ft LiH slab configuration was constant throughout the experiment also; only the 5-ft LiH slabs were changed. From 0 to 5 in. of aluminum was placed between two sections of LiH, also in varying thicknesses. The results of these measurements are shown in Figure 14. The data indicates an attenuation coefficient for aluminum of $\sim 0.07 \text{ cm}^{-1}$.

3. Transverse Leakage Corrections

Two experiments were performed to determine the effects of transverse leakage on the experiments described in Section III-B-1. The centerline attenuation of fast neutrons measured with a Hurst dosimeter, in 3-ft-square slabs of LiH with and without concrete surrounding the shields, is shown in Figure 7. The same experiment was performed with 5-ft-square slabs against the fission plate without concrete. The results of these measurements are shown in Figure 15. The results indicate that there is no correction for transverse leakage for centerline attenuation measurements; i. e., the 3-ft slabs appear to be effectively infinite in the transverse direction for centerline measurements.

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Figure 13. Experimental Setup — Aluminum Effects Experiment

Figure 14.
Attenuation of Fast Neutrons
in Aluminum Immersed
in LiH

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Figure 15.
Transverse Leakage
Correction Experiment

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IV. RESULTS OF ANALYSIS

A. CALCULATIONS OF THE ATTENUATION COEFFICIENT

The data from the experiments were reduced in a manner suitable for determining the attenuation coefficient (μ) associated with the kernel $e^{-\mu r}/4\pi r^2$. Reduction of the data was facilitated by use of the SMAC-II computer code,⁸ developed for this purpose. This program calculates the flux at a detector position from a disc source having a radial source distribution with azimuthal symmetry, attenuated by an infinite slab of thickness t . The flux at P (see Figure 16) is determined from:

$$\phi = \int_{\theta=0}^{2\pi} \int_{r=0}^R S(r) \frac{e^{-\mu\rho}}{4\pi\rho^2} r \, dr \, d\theta$$

where

$$\rho^2 = r^2 + t^2$$

Fluxes are calculated for several thicknesses t , corresponding to the slab shield thicknesses used. An estimate of μ is assumed, and varied until a least-squares fit of calculated $\phi(t)$ to the experimental is minimized. In other words, SMAC-II calculates the experiment, varying the attenuation coefficient, until a best fit is obtained. This program was utilized in all cases in which the source-detector separation was a variable.

Aluminum attenuation corrections, when considered in the calculations, were found to have little effect on the fitted coefficients. The results are summarized in Table 5.

The aluminum correction experiments and the transverse leakage correction experiments were suggested because early measurements, viz the 2-in. shield-detector separation, indicated such wide variations from accepted relaxation lengths for LiH that the above effects were judged to be important. However, as described above, these effects were not found to be significant.

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Figure 16. SMAC II Geometry

It is interesting to note that thermal neutron measurements through LiH at a 1-in. shield-detector separation distance led to an attenuation coefficient very close to that of the fast neutron measurements for a 2-in. separation.

B. INTERPRETATION

It is apparent, from the above, that a simple point kernel approach is not adequate in predicting the effectiveness of LiH as a shield material. However, F.H. Clark of ORNL has indicated that the results observed may be explained by assuming that a varying anisotropy of the emergent flux exists for LiH as a function of thickness.⁹ Therefore, a surface correction should be applied to ray tracing codes in which LiH calculations are performed. Based upon several preliminary calculations, it appears that this surface correction is of a $\cos^x \theta$

TABLE 5
CALCULATED AND EXPERIMENTAL
FIT TO ATTENUATION
COEFFICIENTS

Source-Detector Separation Distance (d) Constant		
d (in.)	Experiment	Calculated
16	0.17 cm^{-1}	0.17 cm^{-1}
72	0.16 cm^{-1}	0.16 cm^{-1}
Shield-Detector Separation Distance (A) Constant		
A (in.)	Experiment	Calculated
2	0.17 cm^{-1}	0.169 cm^{-1}
44	0.156 cm^{-1}	0.158 cm^{-1}

form; with x a slightly varying function of shield thickness may be adequate. However, this function x has not yet been determined, but appears to lie in the range $3 \lesssim x \lesssim 6$.

V. CONCLUSIONS

Values have been obtained for the attenuation coefficient of fission neutrons in lithium hydride using a shielding kernel of the form described above. The values are not unique, but depend upon the separation between the shield and the detector due apparently to angular distribution effects. In determining neutron fluxes at positions close (<10 cm) to the shield surface an attenuation coefficient of 0.169 cm^{-1} is applicable; at large distances from the shield (>150 cm from the shield) a coefficient of 0.158 cm^{-1} is indicated. A better approach would consist of modifying the treatment of neutron current at the shield surface to include angular distribution.¹⁰

Previous values for the lithium hydride attenuation coefficient have been reported by H. Goldstein¹¹ in which a value of 0.139 cm^{-1} was obtained for the infinite medium point source case by the moments method. More recently F. H. Clark¹² calculated a value of 0.156 cm^{-1} for a disc source in a truncated cone with a collar geometry (essentially slab geometry) using Monte Carlo methods. All of these values have been reduced to a standard density of 0.755 g/cm^3 .

Additional experimental studies are needed to provide a direct measurement of the angular distribution rather than inferring one from the difference between two measurements of the attenuation coefficient. Information on the behavior of the neutrons at greater depths in the shield would also be of value in understanding the problem. A measurement of the neutron spectra is needed at various depths in the shield to complete the study.

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APPENDIX

DATA FOR FIGURE 8. Centerline Distribution of Thermal Neutrons in LiH

INDIUM FOILS (Centerline)

Foil Position Following:*		LiH Thickness (cm)	Relative Intensity
Number	Size (in.)		
Fission Plate		0	2.07×10^6
1	1	2.23	4.70×10^6
1	1	4.55	3.82×10^6
1	2	9.22	1.41×10^6
1	3	16.5	3.48×10^5
1	3	23.8	8.50×10^4
1	6†	38.4	5.12×10^3
1	7†	55.5	1.70×10^2
1	7†	72.6	8.3
1	4†	82.1	4.1
1	2†	86.6	4.1
12-in. concrete†			71
12-in. concrete†			1330

*All slabs listed were in place when measurements were made; this column indicates the position of the foil in the shield.

†5-ft slabs

BF₃ AND FISSION CHAMBER (Centerline)

Slab Array		LiH Thickness (cm)	Relative Intensity	
Number	Size (in.)		BF ₃	Fission Chamber
-	-	0	1.3×10^6	1.0×10^6
2	1	4.55	1.8×10^6	1.37×10^6
2 1	1 2	9.22	8.1×10^5	6.3×10^5

* 5-ft² slabs

DATA FOR FIGURE 8. Centerline Distribution of Thermal Neutrons in LiH
(Continued)

BF₃ AND FISSION CHAMBER (Centerline)
(Continued)

Slab Array		LiH Thickness (cm)	Relative Intensity	
Number	Size (in.)		BF ₃	Fission Chamber
2 1 1	1 2 3	16.5	2.0 x 10 ⁵	1.55 x 10 ⁵
2 1 2	1 2 3	23.8	4.8 x 10 ⁴	3.7 x 10 ⁴
2 1 2 1	1 2 3 2*	28.3	1.55 x 10 ⁴	1.3 x 10 ⁴
2 1 2 1	1 2 3 6*	37.8	2.6 x 10 ³	2.9 x 10 ³
2 1 2 1 1	1 2 3 6* 4*	48	1.5 x 10 ³	1.5 x 10 ³
2 1 2 1 1	1 2 3 6* 4*	55.5	1.3 x 10 ³	1.3 x 10 ³
2 1 2 1 2	1 2 3 6* 7*	72	1.2 x 10 ³	1.1 x 10 ³

* 5-ft² slabs

DATA FOR FIGURE 9 — Transverse Map of Thermal Neutrons in LiH

RADIAL THERMAL NEUTRON FLUXES (Indium Foils)

Radial Position in Shield (in. from centerline)	Relative Intensity				
	Axial Position in Shield				
	9.2 cm LiH	23.8 cm LiH	37.8 cm LiH	54.9 cm LiH	72.4 cm LiH
-29.5			2.15×10^3	940	620
-27			5.8×10^2	140	71
-24			5.8×10^2	32	8.6
-21			8.2×10^2	26	3.9
-18			1.1×10^3	34	3.1
-17.5	3.7×10^5	4.4×10^4			
-15	4.3×10^5	2.8×10^4			
-12	8.4×10^5	3.4×10^4	1.7×10^3	64	3.2
-8	14.7×10^5	5.2×10^4			
-6			2.9×10^3	106	5.8
-4	17.1×10^5	6.8×10^4			
0	18.4×10^5	7.3×10^4	3.4×10^3	112	7.1
4	17.8×10^5	6.9×10^4			
6			2.9×10^3	106	4.1
8	14.1×10^5	5.3×10^4			
12	7.0×10^5	3.2×10^4	1.7×10^3	62	3.6
15	4.1×10^5	2.5×10^4			
17.5	3.6×10^5	3.9×10^4			
18			1.18×10^3	37	2.7
21			9.0×10^2	30	4.5
24			6.2×10^2	31	10
27			5.19×10^2	115	68
29.5			2.2×10^3	880	600

DATA FOR FIGURE 10 — Centerline Attenuation of Fast Neutrons in
LiH — Hurst Dosimeter, 4 Geometries

FIXED SHIELD-DETECTOR GEOMETRY

Slab Array		LiH Thickness (cm)	Relative Intensity	
Number	Size (in.)		2-in. Separation	44-in. Separation
-	-	0	1.000	1.000
2	1	4.55	0.228	-
2 1	1 2 }	9.22	0.076	0.21
2 1 1	1 2 3 }	16.53	0.0151	-
2 1 2	1 2 3 }	23.84	0.0034†	0.020
2 1 2 1	1 2 3 2* }	28.27	0.00133	-
2 1 2 1 1	1 2 3 2* 4* }	37.81	0.000198	0.0017
2 1 2 1 1	1 2 3 2* 7* }	45.43	0.000048	-
2 1 2 1 1 1	1 2 3 2* 7* 4* }	54.95	0.000014	-

* Five-foot square slabs used in these positions

† Sulphur foil against slab for this thickness gave 0.0028

DATA FOR FIGURE 10 — Centerline Attenuation of Fast Neutrons in
LiH — Hurst Dosimeter, 4 Geometries (Continued)

FIXED SOURCE-DETECTOR GEOMETRY

Slab Array		LiH Thickness (cm)	Relative Intensity	
Number	Size (in.)		6-ft Separation	16-in. Separation
-	-	0	1.000	1.000
2	1	4.55	0.536	-
2 1	1 2 }	9.22	0.248	0.22
2 1 1	1 2 3 }	16.5	0.0733	-
2 1 2	1 2 3 }	23.8	0.0259	0.015
2 1 2 1	1 2 3 2* }	28.3	0.0119	-
2 1 2 1	1 2 3 6* }	37.8	0.00239	0.0014
2 1 2 1 1	1 2 3 6* 4* }	48	0.000718 0.000575†	-
2 1 2 1 1	1 2 3 6* 7* }	55.5	0.000456 0.000280†	-
2 1 2 1 2	1 2 3 6* 7* }	72	0.000265	-

* 5-ft² slabs

† Fission plate shielding increased, reducing wall-scattered neutrons

DATA FOR FIGURE 11 — Experimental Geometry Correction to
the Fixed Shield Detector-Separation Measurement

LiH Thickness (cm)	Relative Intensity	
	B/A	C/A
0	1	1
4.55	0.293	---
9.22	0.124	0.238
16.5	0.034	---
23.8	0.0102	0.0275
28.3	0.00478	---
37.8	0.00098	0.0027

Distance From Fission Plate (in.)	Relative Intensity (No Lithium Hydride)
0	1
6	0.404
12	0.201
18	0.113
24	0.0752
36	0.0376
48	0.0239

Note: Curves B and C are same as Figure 10,
Fixed Shield-Detector Geometry.

DATA FOR FIGURE 12 — Attenuation of Fast Neutrons in
LiH-Np²³⁷ Fission Chamber

Slab Array		LiH Thickness (cm)	Relative Intensity		
Number	Size (in.)		Bare Chamber	B ₄ C-Covered	3 in. Pb + B ₄ C
-	-	0	1.000	1.000	1.000
2	1	4.55	0.310	---	---
2 1	1 2 }	9.22	0.116	---	---
2 1 1	1 2 3 }	16.5	0.0265	---	---
2 1 2	1 2 3 }	23.8	0.0079	0.0051	0.0031

DATA FOR FIGURE 14 — Attenuation of Fast Neutrons in
Aluminum Immersed in LiH

Shield Configuration			Relative Intensity
Front - LiH (5)* (in.)	Center - Al (6)* (in.)	Rear - LiH (7)* (in.)	
0	0	2	1.000
	1	2	0.768
	3	2	0.508
0	5	2	0.366
0	5 void	2	0.396
0	0	4	0.380
0	1	4	0.295
0	3	4	0.211
0	5	4	0.149
0	0	2 + 4	0.170
0	1	2 + 4	0.141
0	3	2 + 4	0.0981
0	5	2 + 4	0.0708

*Region in Figure 13

DATA FOR FIGURE 14 — Attenuation of Fast Neutrons in
Aluminum Immersed in LiH (Continued)

Shield Configuration			Relative Intensity
Front - LiH (5) * (in.)	Center - Al (6) * (in.)	Rear - LiH (7) * (in.)	
4	0	6	0.0401
4	1	6	0.0350
4	3	6	0.0276
4	5	6	0.0217
2 + 4	0	6	0.0244
2 + 4	1	6	0.0227
2 + 4	3	6	0.0178
2 + 4	5	6	0.015

* Region in Figure 13

DATA FOR FIGURE 15 — Transverse Leakage Correction
Experiment

3-ft slabs (no concrete)

Slab Array		LiH Thickness (cm)	Relative Intensity
Number	Size (in.)		
-	-	0	1.000
1	2	4.77	0.231
1 2	2 1	9.22	0.079
1 2 1	2 1 3	16.53	0.016
1 2 2	2 1 3	23.84	0.00371

Note: Data for 3-ft and 5-ft slabs with concrete same as Figure 10,
Fixed Shield Detector Separation 2 in.

DATA FOR FIGURE 15 – Transverse Leakage Correction
Experiment (Continued)

5-ft slabs (no concrete)

Slab Array		LiH Thickness (cm)	Relative Intensity
Number	Size (in.)		
-	-	0	1.000
1	4	9.54	0.0729
1	7	17.25	0.0138
1	6 }	24.1	0.00318
1	4 }		
1	6 }	28.6	0.00123
1	4 }		
1	2 }		

Note: Data for 3-ft and 5-ft slabs with concrete same as Figure 10,
Fixed Shield Detector Separation 2 in.