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MEASURED AND CALCULATED FAST NEUTRON SPECTRA IN A DEPLETED URANIUM AND LITHIUM HYDRIDE SHIELDED REACTOR

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16. Abstract

Measurements of MeV neutron spectra were made at the surface of a lithium hydride and depleted uranium shielded reactor. Four shield configurations were considered: these were assembled progressively with cylindrical shells of 5-centimeter-thick depleted uranium, 13-centimeter-thick lithium hydride, 5-centimeter-thick depleted uranium, and 3-centimeter-thick depleted uranium. Measurements were made with a NE-218 scintillation spectrometer; proton pulse height distributions were differentiated to obtain neutron spectra. Calculations were made using the two-dimensional discrete ordinates code DOT and ENDF/B (version 3) cross sections. Good agreement between measured and calculated spectral shape was observed. Absolute measured and calculated fluxes were within 50 percent of one another; observed discrepancies in absolute flux may be due to cross-section errors.

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SUMMARY

Measurements of MeV neutron spectra were made at the surface of a cylindrical shield arrangement surrounding a small (25.4-cm diam), cylindrical uranyl fluoride (UO_2F_2) - water solution reactor. Four shield configurations were considered: These were assembled consecutively with cylindrical shells of 5-centimeter-thick depleted uranium, 13-centimeter-thick lithium hydride, 5-centimeter-thick depleted uranium, and 3-centimeter-thick depleted uranium. Measurements were made using a NE-218 scintillation spectrometer. Proton recoil pulse height distributions were differentiated to obtain neutron spectra. Calculations of leakage flux were made with the two-dimensional discrete ordinates code DOT using ENDF version 3 cross sections. Comparisons of experimental and calculated results on an absolute basis show some deviations, with the deviations growing with increasing shield thickness. This discrepancy may be due to cross-section errors. However, excellent agreement between measured and calculated spectral shapes was observed.

INTRODUCTION

The design of a minimum weight shield for a nuclear power reactor requires confidence in calculations and cross-section data. To check neutron transport calculations, a number of transmission experiments were conducted using both small neutron sources and the NASA Zero-Power Reactor Facility. This series of experiments first examined geometrically clean spherical configurations of lead and other materials (refs. 1 and 2) and then cylindrical configurations of lead and water (ref. 3).

The present report describes calculations and measurements made for the cylindrical reactor surrounded by alternate layers of depleted uranium and lithium hydride. Calculations were made using the two-dimensional discrete ordinates transport code

DOT (refs. 4 and 5). Neutron cross sections were from ENDF/B (version 3) files; the basic data were collapsed to a 43-group multigroup library using SUPERTOG-III (ref. 6). Measurements were made using a 1.22-centimeter inside diameter by 1.27-centimeter long NE-218 scintillator. The measurements of proton recoil spectra were differentiated using the TUNS code (ref. 7) to obtain neutron spectra.

EXPERIMENTAL SETUP

Facility

The test facility used in these experiments is the NASA Zero Power Reactor Facility. The uranyl fluoride - water solution reactor in this facility is 25.4 centimeters in diameter, and, depending on the fuel concentration and reflector material, can be operated at heights up to 76 centimeters. The reactor is licensed to operate at a maximum of 100 watts thermal (total neutron source strength of about 8×10^{12} neutrons/sec). The reactor vessel is 0.278 centimeter thick and is made of AISI 316 stainless steel. The reactor is surrounded by a 244-centimeter-diameter by 104-centimeter-deep cylindrical tank; shielding materials may be arranged about the reactor in this tank.

Present Experimental Setup

The present experimental setup is illustrated schematically in figure 1. Exact dimensions and material descriptions are given in table I. The reactor is surrounded by 91-centimeter-high cylindrical shells of water (5 cm), depleted uranium (5 cm thick), lithium hydride (13 cm thick) followed by two more of depleted uranium (5 cm and 3 cm thick). Midplane measurements of the neutron transmission were made for these four configurations:

(1) Core plus water plus the first 5-centimeter-thick depleted uranium layer

(2) Configuration 1 plus a 13-centimeter-thick lithium hydride layer

(3) Configuration 2 plus a 5-centimeter-thick depleted uranium layer

(4) Configuration 3 plus a 3-centimeter-thick depleted uranium layer.

The thicknesses of these layers correspond roughly to those considered for space power reactor shields. The midplane measurements were made with the detector at the outer surface of each configuration.

Shield Material Fabrication

The depleted uranium shells were rolled from plate and canned with 0.19 centimeter (75 mil thick) stainless steel. The lithium hydride shells were cast in segments using techniques developed in connection with the Atomic Energy Commission's SNAP reactor program. The lithium hydride was internally reinforced using a stainless-steel honeycomb matrix. About a 94-percent theoretical density of lithium hydride was achieved in those castings. Details of this lithium hydride casting may be found in reference 8.

MEASUREMENTS

Spectrometer

Measurements of the neutron spectra emerging from the shield assembly were made at the horizontal midplane using a 1.22-inside-diameter by 1.27-centimeter-long NE-218 scintillator. The scintillator was optically coupled to an RCA 8575 photomultiplier tube using a 1.3-centimeter light pipe. A modified Owens type of pulse shape circuit (as described in ref. 9) was used. This circuit is used with a two-parameter pulse-height analyzer to separate proton recoil (neutron-induced scattering events) and electron recoil (gamma-ray-induced scattering events) occurring within the scintillator. A spectrometer block diagram is illustrated in figure 2. The proton recoil region of the two-parameter analyzer data was summed using the PREJUD code (ref. 10); the output data from PREJUD was then differentiated using the TUNS code (ref. 7) to obtain neutron spectra. Reference 7 also contains more details on this experimental technique.

Power Calibration

To compare calculated results with the experimental measurements on an absolute basis, both were normalized to correspond to a reactor thermal power level of 1 watt. The reactor power was calibrated by mapping the thermal flux within the reactor with dysprosium foils, which had previously been calibrated against indium foils. The total fission rate was then calculated from the thermal neutron flux level determined by these foil measurements. This was then correlated with integral counting data taken over the period of measurement using a helium-3 proportional counter. The helium-3 proportional counter data was then used to determine reactor power levels for all other measurements.

Experimental Error

Systematic errors. - The uncertainty in the absolute power level of the reactor is estimated to be ±15 percent. Because of the nature of the method of determining power level, this relative error will be approximately constant throughout the present series of experiments and is one possible source of systematic error. The relative power level uncertainty is less than 5 percent.

Another source of systematic error is introduced through the detector efficiencies used in the spectrum unfolding code. The efficiency data used was that for a slightly different detector and was measured without a light pipe between the detector and the photomultiplier tube. One test measurement, however, showed little difference in efficiencies between the detector systems; thus, only a small error is anticipated from this uncertainty.

Random errors. - Counting statistics introduce a random error to the measurements. These are represented by the error bars appearing in the plots of experimentally measured neutron spectra and are an inverse measure of the count rate observed.

CALCULATIONS

The leakage flux per unit energy in the four experimental configurations described earlier was calculated using the two-dimensional discrete ordinates code DOT. DOT was run in a reactivity-calculation mode to account for neutron production in the outer uranium layers. A symmetric S-8 (48 angle) quadrature and P-3 scattering approximations were used. The experiment was mocked up for calculation as described in table I. Table I also lists the number of radial mesh intervals considered. Midplane symmetry was assumed; 27 axial intervals were used in the calculation. Table II lists boundaries for the 43-group split used in the calculations. The top 30 groups correspond to a lethargy split of $\Delta u = 0.1$. The resonance and thermal groups were necessary in the calculation because of the neutron production in both U^{235} and U^{238} in the depleted uranium shield layers.

Multigroup cross sections used were assembled from ENDF/B (version 3) data using the SUPERTOG-III code. Data for several elements were taken from GAM-II files. The sources of cross-section data used is listed in table III. Macroscopic cross sections were assembled using atom densities listed in table IV.

COMPARISON OF MEASURED AND CALCULATED SPECTRA

The neutron leakage flux from each of the four experimental configurations is shown in figure 3. Flux per unit energy is shown for the interval of 0.9 to 14 MeV. To provide a comparison on a uniform basis, both calculated and measured results were normalized to correspond to a power level of 1 watt (reactor fission rate of 3.1×10^{10} fissions/sec).

Several stages of postprocessing were necessary to convert the DOT-calculated multigroup fluxes to those shown in figure 3. First, the surfaces angular fluxes were converted to a scalar flux at the surface using the equation

$$\varphi = \sum_{\substack{\text{All} \\ \text{angles}}} w_i \psi_i$$

where φ is the group scalar flux, w_i is the weight (solid angle) associated with ψ_i, and ψ_i is the i^{th} angular flux. Next, this surface multigroup scalar flux was converted to flux per unit energy E by dividing it by the energy group width, that is,

$$\varphi(E) = \frac{\varphi}{\Delta E}$$

Finally, because the optics and the electronics of the instrumentation system introduce statistical variations, a Gaussian smearing of data is performed by the instrumentation. To provide a realistic comparison between calculation and experiment, the calculated results must also be smeared by this Gaussian function. Symbolically, this is

$$\varphi_s(E)dE = \int_0^\infty \varphi(E')G[R(E'), E', E]dE'\, dE$$

where $\varphi(E')dE'$ is the actual flux of neutrons in dE' about E'; $G[R(E'), E', E]dE$ is the probability (a Gaussian) that a particle of energy E' will appear to the instrumentation as a particle in dE about E, where G is of the form $A/R(E') \exp(-B\{(E' - E)/[R(E')]\})^2$ and A and B are constants such that $\int_0^\infty G\, dE = 1$, namely, $A = \sqrt{\ln 2/\pi}$ and $B = \ln 2$; $R(E')$ is the resolution (more precisely, the half width at half maximum) of the instrumentation for particles of energy E'; and $\varphi_s(E)dE$ is the "smoothed" neutron flux in dE about E, which should be comparable to what the instrumentation sees.

The resolution R(E) of the present instrument is taken to be half that used for the 5- by 5-centimeter (2- by 2-in.) NE-213 detector (ref. 11). Only the smeared calculated fluxes are shown in figure 3.

In all cases, the measured and calculated spectral shapes (fig. 3) show excellent agreement. Differences in absolute magnitude greater than the uncertainty in the power calibration appear in the results for configuration 4. There also appears to be an increase in the difference between measured and calculated results with increasing shield thickness. Some possible reasons for the differences include:

(1) Cross sections in error. A 5 percent error in, say, the total cross section of depleted uranium would result in a 20 percent error in calculated transmission through the 13 centimeters of depleted uranium considered here. Because differences are greater when lithium hydride is present, lithium hydride cross sections are more suspect.

(2) Cracks and voids in the cast lithium hydride (as described in ref. 8), which could permit higher transmission than calculated for homogeneous material.

(3) Inaccurate transport calculations due to inadequate spatial and angular mesh and/or scattering order. But this is presently discounted; an S_{12} angular quadrature was used in one calculation and showed only a few percent difference from the S_8 calculations.

The increasing discrepancy with added thickness can most reasonably be attributed to cross section errors. Because of the good agreement in spectral shape, the inelastic cross section is not suspect. This leaves the total and/or elastic scattering cross section in doubt. But a definitive determination of such a discrepancy cannot be determined from an integral experiment such as this.

On a positive note, the calculated results are all within 50 percent of the measured results. Thus, an engineering design tool, the calculative method (and cross sections) presently used is accurate to within this value.

CONCLUDING REMARKS

Measurements and calculations of MeV neutron spectra emerging from a depleted uranium and lithium hydride shielded reactor have been made. In general, the agreement between experiment and calculation is good. Spectral shapes are in excellent agreement. Comparisons made on an absolute basis are within 50 percent for all cases considered. Differences in spectra of the order of 50 percent were observed in the configuration using the thickest shield, namely, 13 centimeters of depleted uranium and 13 centimeters of lithium hydride. This difference is possibly attributable to inhomogeneities (cracks and voids) in the cast lithium hydride or to erroneous cross sections.

Latest ENDF (version 3) cross sections for lithium 6 and 7, hydrogen, uranium 235 and 238, and iron were used in the calculations.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, July 2, 1973,
503-05.

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TABLE I. - REGION DESCRIPTION

Region	Region material description	Outer radius of region, cm	Number of radial mesh intervals used in transport calculation
1	Reactor core	12.70	11
2	Reactor vessel	13.018	1
3	Water	18.085	5
4	Stainless steel	18.276	1
5	Depleted uranium	23.326	5
6	Stainless steel	23.517	1
7	Void	24.079	1
8	Stainless steel	24.270	1
9	Lithium hydride	37.147	13
10	Stainless steel	37.338	1
11	Void	37.897	1
12	Stainless steel	38.088	1
13	Depleted uranium	43.091	5
14	Stainless steel	43.282	1
15	Void	43.713	1
16	Stainless steel	43.904	1
17	Depleted uranium	46.926	3
18	Stainless steel	47.117	1

TABLE II. - ENERGY GROUP SPLIT USED IN CALCULATIONS

[Upper energy boundary of group 1 is 14.92 MeV.]

Group	Lower energy boundary of group, eV	Group	Lower energy boundary of group, eV
1	13.50×10^{6}	26	1.11×10^{6}
2	12.21	27	1.00
3	11.05	28	.91
4	10.00	29	.82
5	9.05	30	.74
6	8.19	31	.55
7	7.41	32	.41
8	6.70	33	.11
9	6.07	34	15.03×10^{3}
10	5.49	35	3.35
11	4.97	36	582.95×10^{0}
12	4.49	37	101.30
13	4.07	38	29.02
14	3.68	39	10.68
15	3.33	40	3.06
16	3.01	41	1.13
17	2.73	42	.41
18	2.47	43	.00
19	2.23		
20	2.02		
21	1.83		
22	1.65		
23	1.50		
24	1.35		
25	1.22		

TABLE III. - SOURCES OF CROSS SECTION DATA USED IN PRESENT CALCULATION

Element	ENDF material number
Hydrogen	1148
Lithium-6	1115
Lithium-7	1116
Oxygen	1134
Fluorine	(GAM-II data used)
Chromium	(GAM-II data used)
Iron	1180
Nickel	(GAM-II data used)
$Uranium^{235}$	1157
$Uranium^{238}$	1158

TABLE IV. - ATOM DENSITIES OF MATERIALS USED IN PRESENT CALCULATION

Material	Element or isotope	Density, $atoms/cm^3$
Core	Hydrogen	0.06639×10^{24}
	Oxygen	.033419
	Fluorine	.00022236
	$Uranium^{235}$	.00010362
	$Uranium^{238}$	.00000756
Stainless steel	Chromium	0.01559×10^{24}
	Iron	.060638
	Nickel	.009745
Lithium hydride	Hydrogen	0.056837×10^{24}
	Lithium-6	.004217
	Lithium-7	.052620
Depleted uranium	$Uranium^{235}$	0.000109×10^{24}
	$Uranium^{238}$	.047200

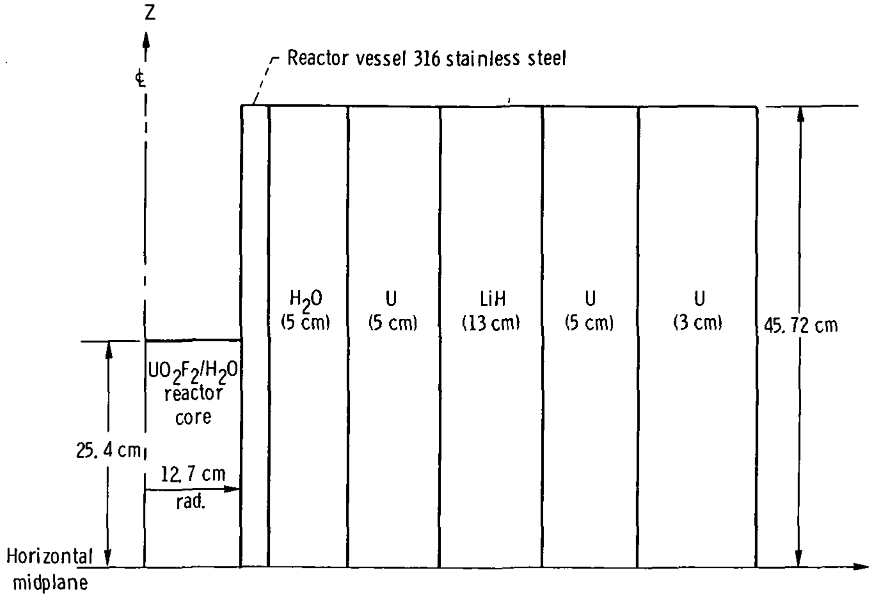


Figure 1. - Sketch of experimental setup. Region above horizontal midplane is shown. Dimensions indicated are approximate and are only for general orientation. The exact dimensions are given in table I.

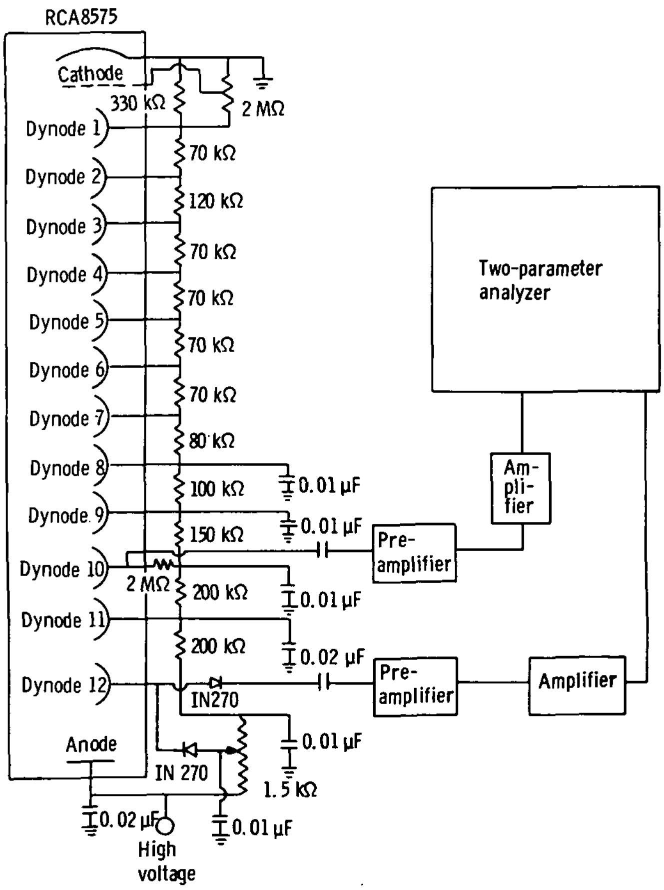
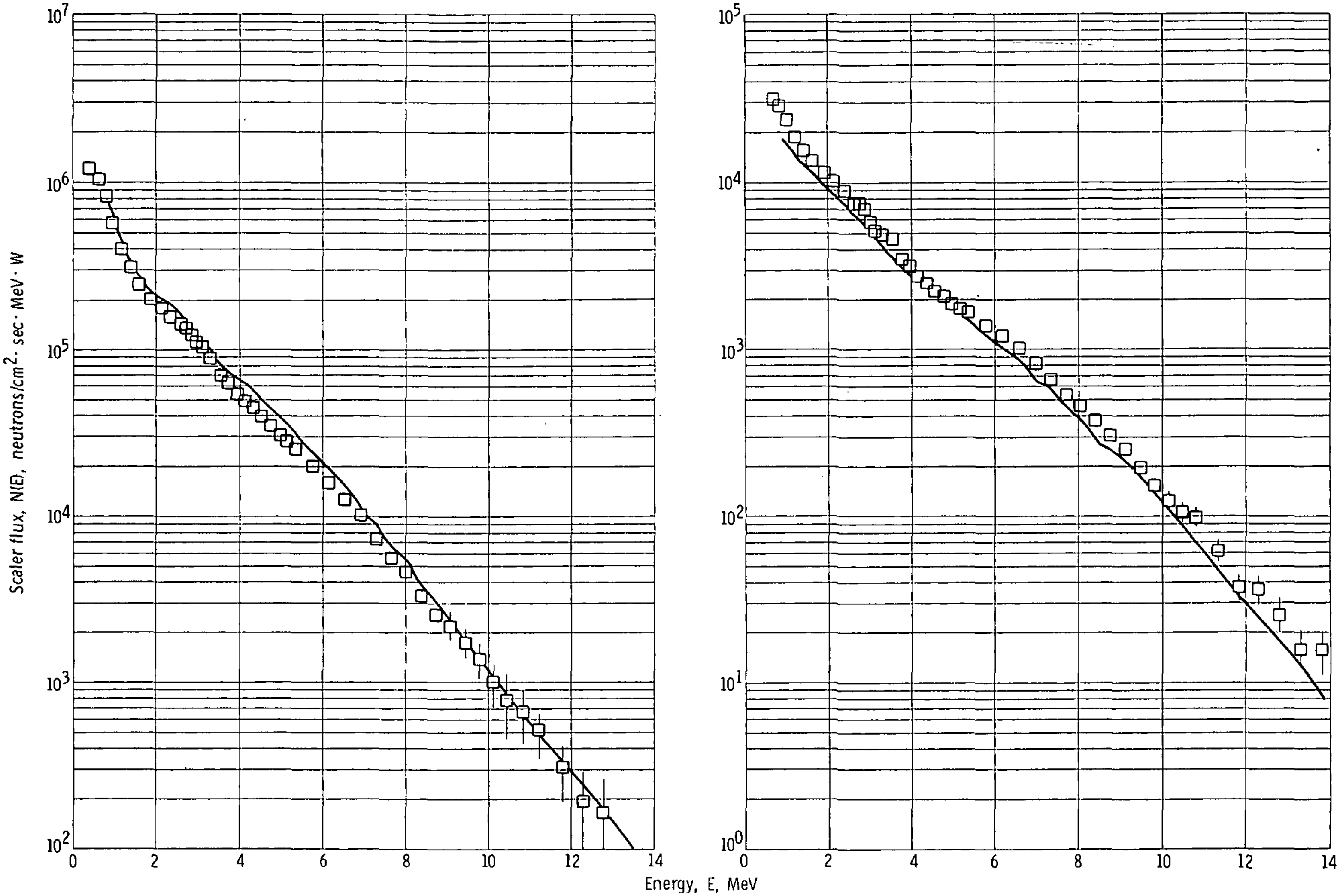


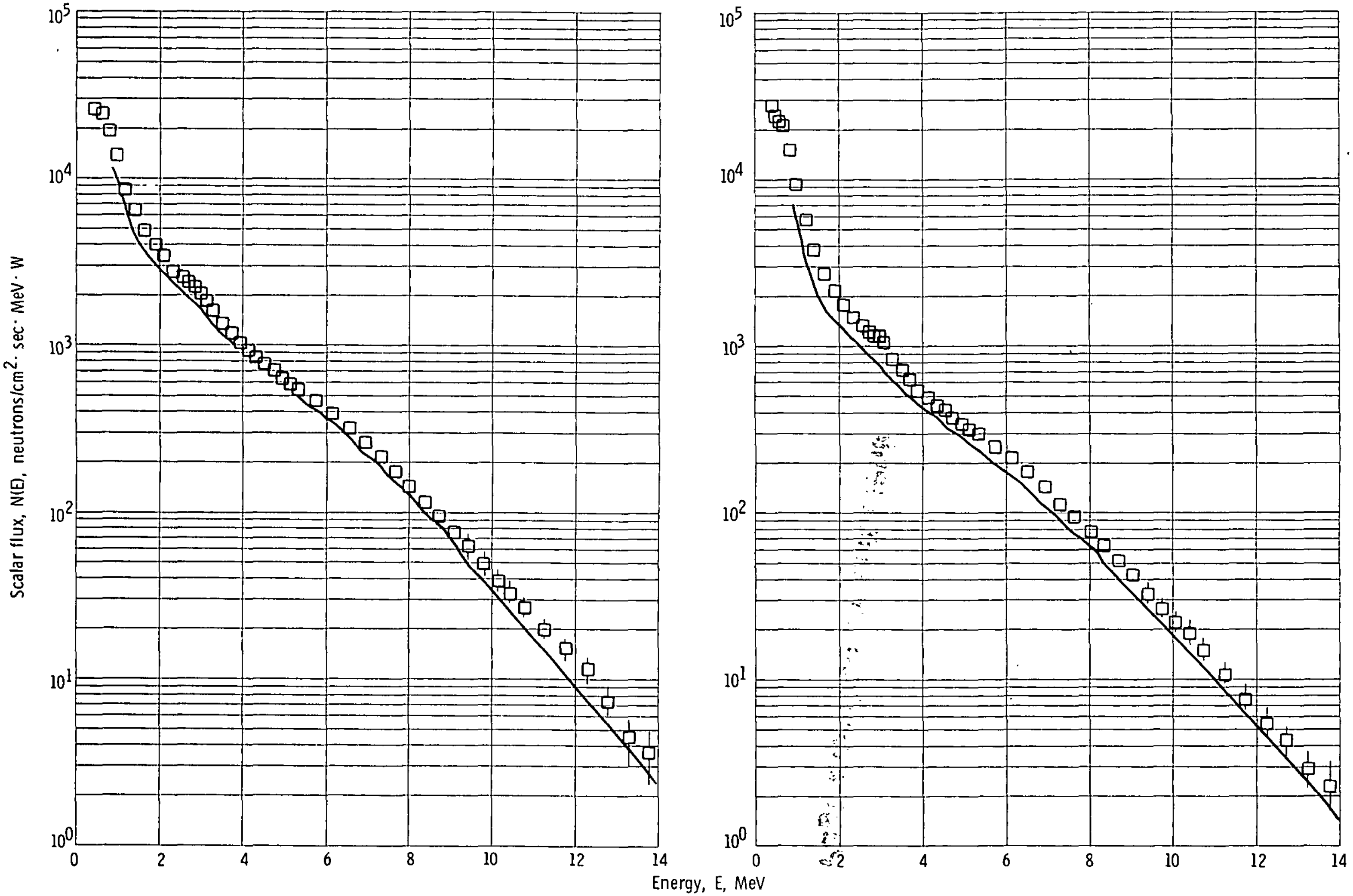
Figure 2. - Spectrometer block diagram and photomultiplier tube base circuit.



(a) Shield configuration: 5 centimeters of depleted uranium.

(b) Shield configuration: 5 centimeters of depleted uranium plus 13 centimeters of lithium hydride.

Figure 3. - Calculated and measured neutron flux at midplane of shield assembly.



(c) Shield configuration: 5 centimeters of depleted uranium plus 13 centimeters of lithium hydride plus 5 centimeters of depleted uranium.

(d) Shield configuration: 5 centimeters of depleted uranium plus 13 centimeters of lithium hydride plus 5 centimeters of depleted uranium plus 3 centimeters of depleted uranium.

Figure 3. - Concluded.

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