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VARIATION OF A FISSION NEUTRON FLUX AND SPECTRUM FROM A FAST REACTOR MEASURED OVER LARGE DISTANCES

Authors

Stephens, Lloyd D.
Aceto, Henry.

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University of California

Ernest O. Lawrence Radiation Laboratory

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Lawrence Radiation Laboratory
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ABSTRACT

Variations over large distances in the numerical flux density, the energy flux density, and the spectrum from a point source of fission neutrons were determined experimentally.

Experimental work was carried on at the Nevada Test Site, using the Oak Ridge National Laboratory Health Physics Research Reactor as the source of neutrons. The reactor height could be varied between 27 and 1500 ft. (8 and 457 meters) and the horizontal measurements extended from 250 to 1500 yd (228 to 1371 meters) from the tower base. The reactor was operated at power levels from 1 to 1250 W.

Numerical flux densities were measured by the use of BF_3-filled proportional counters surrounded by various thicknesses of paraffin. Each detector therefore has its peak sensitivity in a different energy region of the given spectrum. Numerical flux densities were also measured by indium-foil activation. The foils were exposed in right circular cylinders of cadmium filled with paraffin.

Energy flux density was measured by the use of a polyethylene-lined proton-recoil proportional counter. If the ratio between the latter detector and a BF_3 detector is taken, a measure of the average neutron energy is thus obtained.

Nuclear emulsions were exposed to the reactor neutron flux at several locations. From an analysis of the proton recoil tracks, the neutron spectrum can be inferred.

Results of this study provide much information on the varitions of the neutron flux and spectrum as measured over large distances from a source. An analysis of the data also reveals the contribution to the total measured flux due to scattered neutrons, as well as the relaxation length of the given spectrum.

VARIATION OF A FISSION NEUTRON FLUX AND SPECTRUM FROM A FAST REACTOR MEASURED OVER LARGE DISTANCES*

Lloyd D. Stephens and Henry Aceto, Jr.†

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University of California
Berkeley, California
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INTRODUCTION

A primary function of the Health Physics Department at Lawrence Radiation Laboratory is determination of the radiation fields around the 6.3-GeV proton synchrotron and the 730-MeV synchrocyclotron. In nearly all instances the neutron leakage is a far greater hazard than the gamma radiation. Our responsibilities also require that we properly assess the hazards involved and make appropriate recommendations for shielding and machine operation. With these functions in mind, we became interested in the following study.

Neutrons from a source propagated to large distances in air arrive at the detector or point of interest by two paths. The direct neutron flux, composed of neutrons that have not suffered any scattering event, with only air attenuation, can easily be determined by a simple calculation. Neutrons arriving at the detector or point of interest through a devious path due to scattering in air, ground, or possibly some of each, are not as easily calculated.

The expected total neutron flux at any point as determined from one or two measurements made at given points is one of our concerns and a method for predicting this value is one of the objectives of this study. The total neutron flux is not only effected by the detector-to-source distance but also by the height of the source above the ground.

The variations in the neutron spectrum at the large distances involved are very slight and behave according to a predicted pattern [1, 2]. All of the measurements made during this experiment bear out this fact.

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† Present address University of Texas, Southwestern Medical School, Dallas, Texas.

From the data presented in this report one can predict the neutron flux at distances as large as one mile (1.4 km) from the source and various source heights. The experimental work was carried on at the Nevada Test Site, with the Oak Ridge National Laboratory Health Physics Research Reactor used as the source of neutrons [3]. The reactor was secured to a hoist car which was mounted on a 1530-ft tower (465-meters). The reactor height could be varied between 27 and 1500 ft (8 and 457 meters), and the horizontal measurements extended from 250 to 1500 yd (228 to 1371 meters) from the tower base. The reactor was operated at power levels from 1 to 1250 W.

DETECTORS

Four kinds of detectors were used in this experiment. The first type was the BF_3 proportional counter. Six of these were used, one being bare (unmoderated), while the other five were moderated by five different thicknesses of paraffin. The moderators were enclosed in cadmium jackets; therefore the peak sensitivity of each detector was in a different energy region of the given spectrum[4].

A second type of detector that was used was an argon-CO_2-filled proportional counter that counts pulses caused by recoiling protons due to neutrons interacting in the polyethylene lining in the counter [4, 5]. With a 1/8-in.-thick lining of polyethylene, the efficiency of the chamber is nearly proportional to energy for all neutrons of energies from 0.03 to 20 MeV. When used in this manner the counter actually measures energy flux instead of particle flux. For the size of counter used here, one count was nearly equal to 15 MeV/cm^2 of neutrons through the chamber. If we adjust the discriminator to a proper level, the gamma pulses in the detector are rejected.

The third detector was a right circular cylinder of 0.030-in.-thick cadmium 6 in. in diameter by 6 in. high filled with paraffin. Indium foils 1 in. in diam. and 0.005 in. thick were placed in the center of the detector and exposed to the moderated fast neutrons. The β^--activity was counted with an end-window β-γ counter. The foil counter used here gave essentially the same background and counting rate [(counts/min-g) per ($1\ n/cm^2$-sec)] as the gas flow proportional counter, as previously reported [6].

Nuclear-track emulsions were the fourth type of detector used. The neutron energies given by these films were, in our opinion, not realistic values, because the efficiency decreases rapidly below 500 to 600 keV.

Because of the reduced sensitivity in the region 0-500 keV, the average neutron energies as given by our emulsion were in the 1.6 MeV region. This value is not consistent with any other data we obtained [7].

COLLECTION OF DATA

Our detectors were mounted on a light aluminum framework and normally were 6 ft above the ground with the long axis at 90 deg to the reactor. Other orientations and heights were used at each location to determine whether the neutron flux showed any anisotropy. The paraffin-filled cadmium cylinders were placed at 250-yd intervals to obtain simultaneous measurements of the variation of the neutron flux. Our measurements were obtained at as many reactor heights, horizontal distances and power levels as were consistent with the experimental needs of the Program-1 dose measurements.

Before data were taken on any given day, the response of each of the BF_3 counters and the polyethylene-lined counter was checked by using a Pu-Be source. A standard check for each electronic counting circuit was also done each day. The overall gain of each amplifier system was adjusted to within 0.1 dB. All discriminators were checked each day as an added precaution to insure accuracy of measurements. The β^- counting system was checked with a Cs^{137} source. See Fig. 1.

Our equipment was powered by a portable diesel-electric system, and the output of the generator was fed into a "Stabiline" voltage regulator. In addition the outputs of all power and high-voltage supplies were stabilized. Calibrations were checked again at the end of each experimental day.

CALIBRATION OF THE DETECTORS

The BF_3 detectors were calibrated using a Po–mock–fission source and a Pu-Be source. They were exposed at several angles from 0 to 180 deg. One can obtain from these measurements the efficiency of the detectors for an isotropic flux. The efficiency of the detectors as a function of angle θ is shown in Fig. 2. It can be seen that the counting rate as a function of θ has enough variation to give an indication of anisotropy from the reactor.

The response of the polyethylene-lined proportional counter was calibrated by using several neutron sources of known energy and strength. The sources of neutrons used included Po-Li, Po–mock–fission, Po-Be, Pu-Be, and Ra-Be, as well as the d-d and d-t reactions. The neutrons from an Sb-Be neutron source were clearly seen above the gamma rays from the Sb alone; however this point was not used for an actual calibration point. Figure 3 shows the response of this detector as a function of neutron energy.

The indium-foil–paraffin–cadmium neutron detectors were calibrated in a similar manner. The neutron sources were augmented with neutrons of several energies produced by a Van de Graaff accelerator at the U. S. Naval Radiological Defense Laboratory in San Francisco, California. The response of the indium foils is seen in Fig. 4.

NEUTRON FLUX AND SPECTRUM MEASUREMENTS

The neutron flux measured at the various locations was composed of the uncollided neutrons and those that had scattered in the air and (or) ground. The neutron flux was greater than that expected from simple calculations in which a λ value of 210 yd [1] was used in the equation

$$\phi = \frac{Qe^{-D/\lambda}}{4\pi r^2} .$$

From measurements the value of λ that best fits the data is 240 yd. A calculation using this value yields a leakage current of about 1.4×10^{18} n/kWh. A relaxation length of 210 yd gives a calculated flux at a slant range of 1068 yd of about 20 n/cm^2-sec. Our measured value was slightly more than 100 n/cm^2-sec. We attribute this increase--a factor of 5--to the scattered neutrons. Using the λ value of 240 yd, we can properly fit calculated data to the measured data to within 10% at distances from 500 to 1500 yd. Data obtained by using the indium foils also fit this value to nearly the same degree of accuracy (see Figs. 5 and 6).

The effect of increased source height results in an apparent increase in source strength due, we believe, to more efficient irradiation of the ground area around our detectors. We are able to predict the expected neutron flux at a given point if we know the neutron leakage current, λ, and use the graph giving the relative source strength increase for equal slant ranges vs source height (Fig. 7).

The determination of the neutron spectrum is not completed yet. Work is continuing on a solution to this problem. We can make several comments now, however, regarding the neutron spectrum. Beyond the region of about 500 yd the neutron spectrum does not vary, as can be seen from the equal attenuation slopes of all detectors shown in Fig. 8. At distances less than 500 yd the slopes of plots of counting rate vs slant range for the various detectors vary in an erratic fashion, indicating that equilibrium has not yet been attained.

The average energy as determined from the ratio of the energy-flux counter and the particle-flux counter give quite consistent results [8]. The average energies are shown in Fig. 9 for various slant ranges and source heights.

CONCLUSIONS

Some of the things that we can say regarding the experiment as it has been evaluated to date are as follows.

Neutrons from an unmoderated reactor are apparently attenuated less rapidly than those from U^{235} fission weapons. The measured neutron flux at large distances is greater than would be predicted from previous measurements. We can also state that the measured flux is a function of source height, with the neutron flux increasing with source height. This value can be determined approximately from the data given in this report.

Also, there is good agreement with other measurements, which indicates that the neutron spectrum shows little or no variation with distance once an equilibrium has been established.

One of our immediate uses of this information will be to predict neutron radiation levels at our Laboratory boundaries and at large distances within the Laboratory as they are created by the recently improved 6.3-GeV Bevatron.

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One of us (L. D. S.) wishes especially to thank his co-workers in the Health Physics Department at this laboratory for their help during both days and nights when they so ably carried on the necessary duties that are always left behind in an undertaking such as this.

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FIGURE LEGENDS

Fig. 1. Block diagram of electronic counting systems.

Fig. 2. Angular efficiency of moderated BF_3 detectors.

Fig. 3. Polyethylene-lined proton-recoil detector efficiency as a function of incident-neutron energies.

Fig. 4. Response curve for cadmium-paraffin-indium foils as a function of neutron energy.

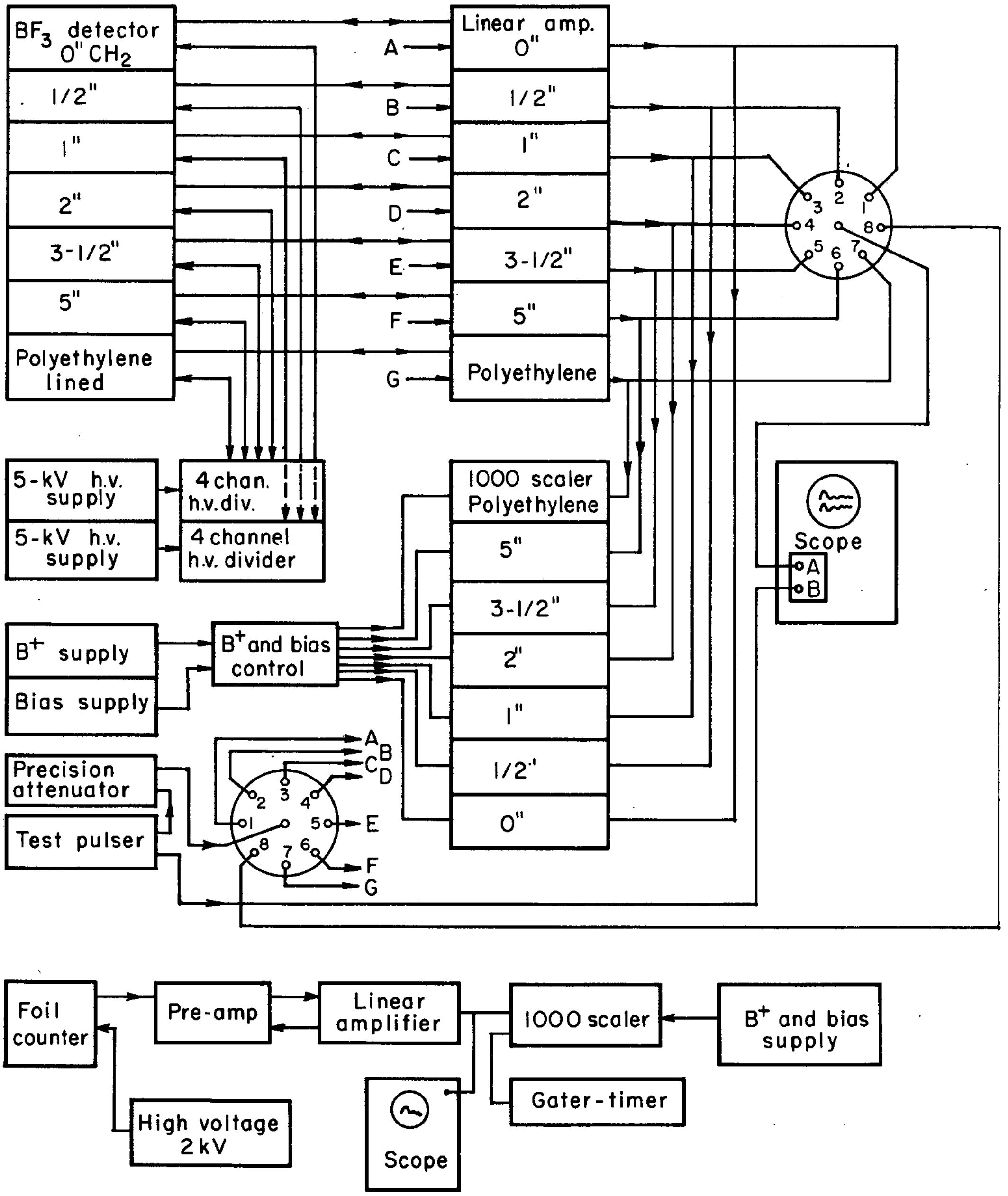
Fig. 5. Neutron flux vs slant range, measured with a BF_3-CH_2 neutron detector. Measured points and points calculated from information in this report are shown.

Fig. 6. Neutron flux vs slant range, measured with the indium-foil-paraffin-cadmium technique described. All points for a given source height were determined simultaneously by using this technique. Several BF_3 points are shown for comparison.

Fig. 7. Relative increase in source strength as a function of the source height for equal slant ranges.

Fig. 8. Counting rate $\times r^2$ vs slant range. The slope of these curves gives the value of $\lambda = 240$ yd.

Fig. 9. Curve of average neutron energies measured at several horizontal distances.



MUB-1404

Fig. 1

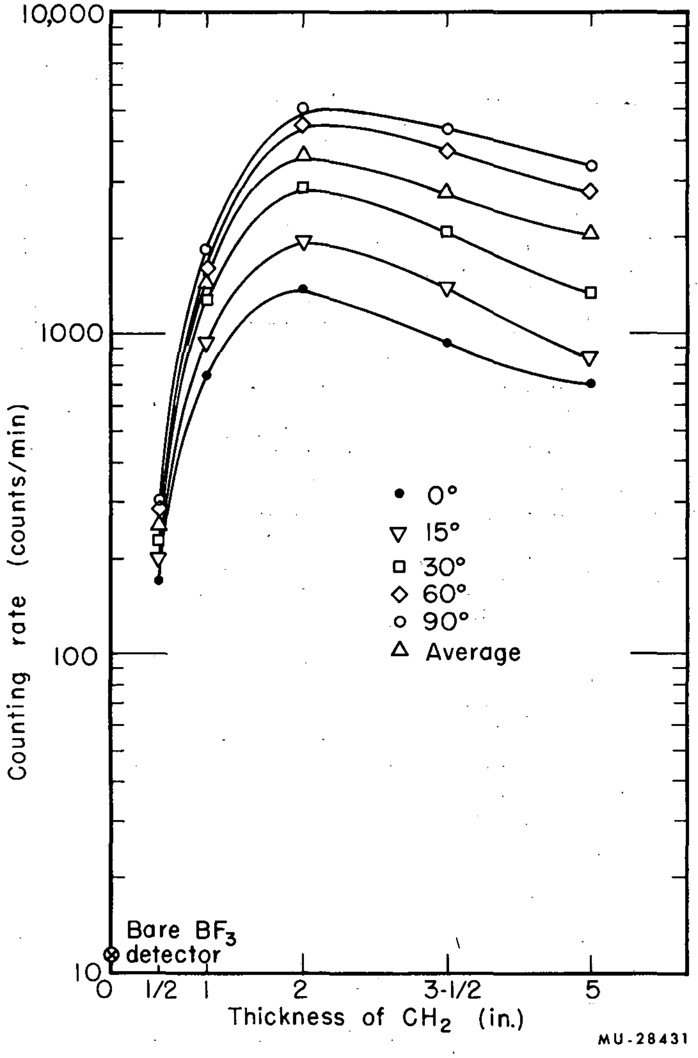

10,000
1000
100
10
Counting rate (counts/min)
0
1/2
1
2
3-1/2
5
Thickness of CH2 (in.)
Bare BF3
⊗ detector
● 0°
▽ 15°
□ 30°
◇ 60°
○ 90°
△ Average
MU-28431

Fig. 2

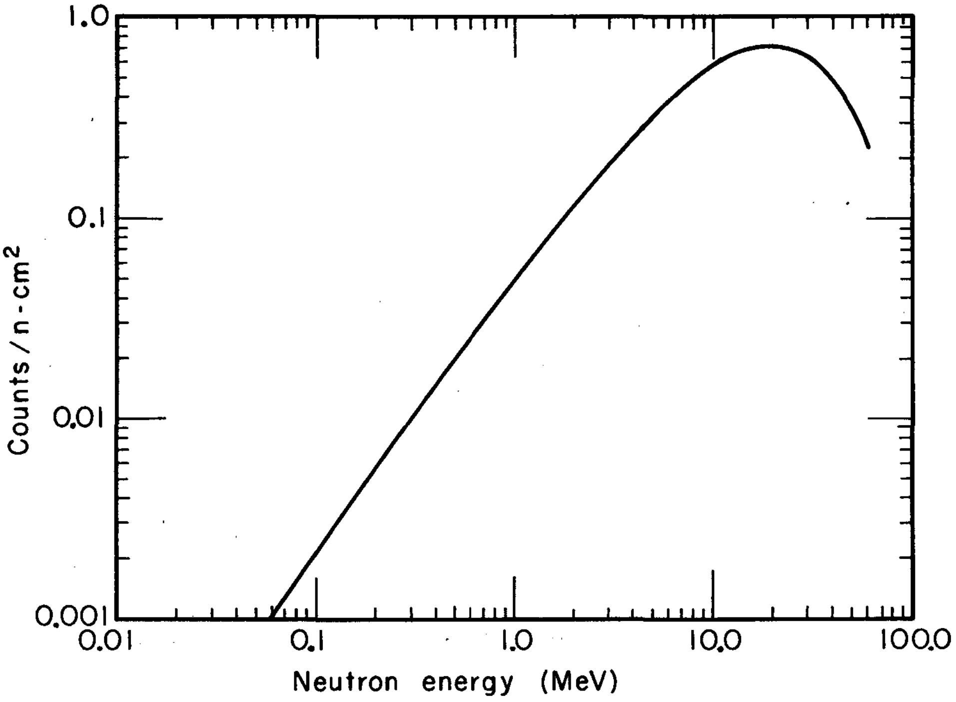


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Fig. 3

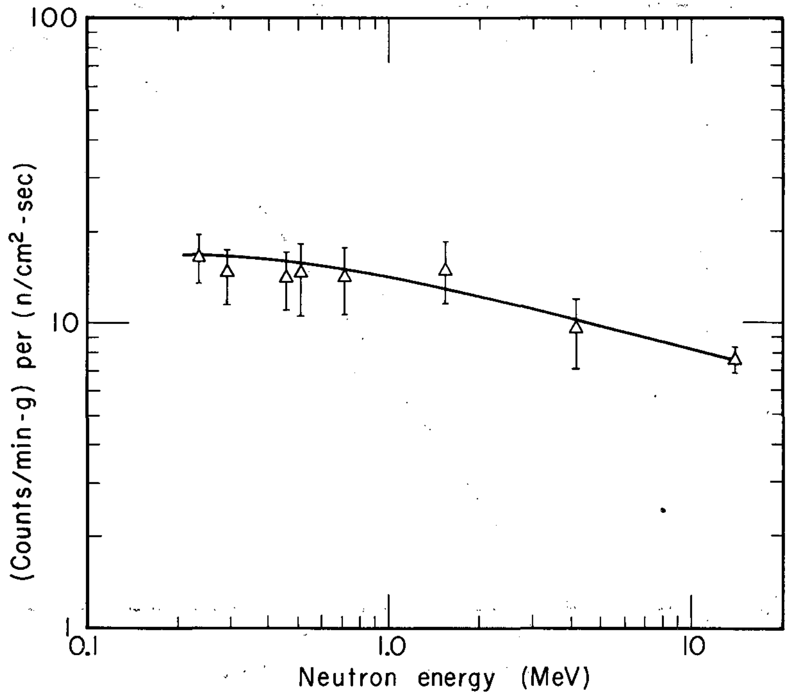

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Fig. 4

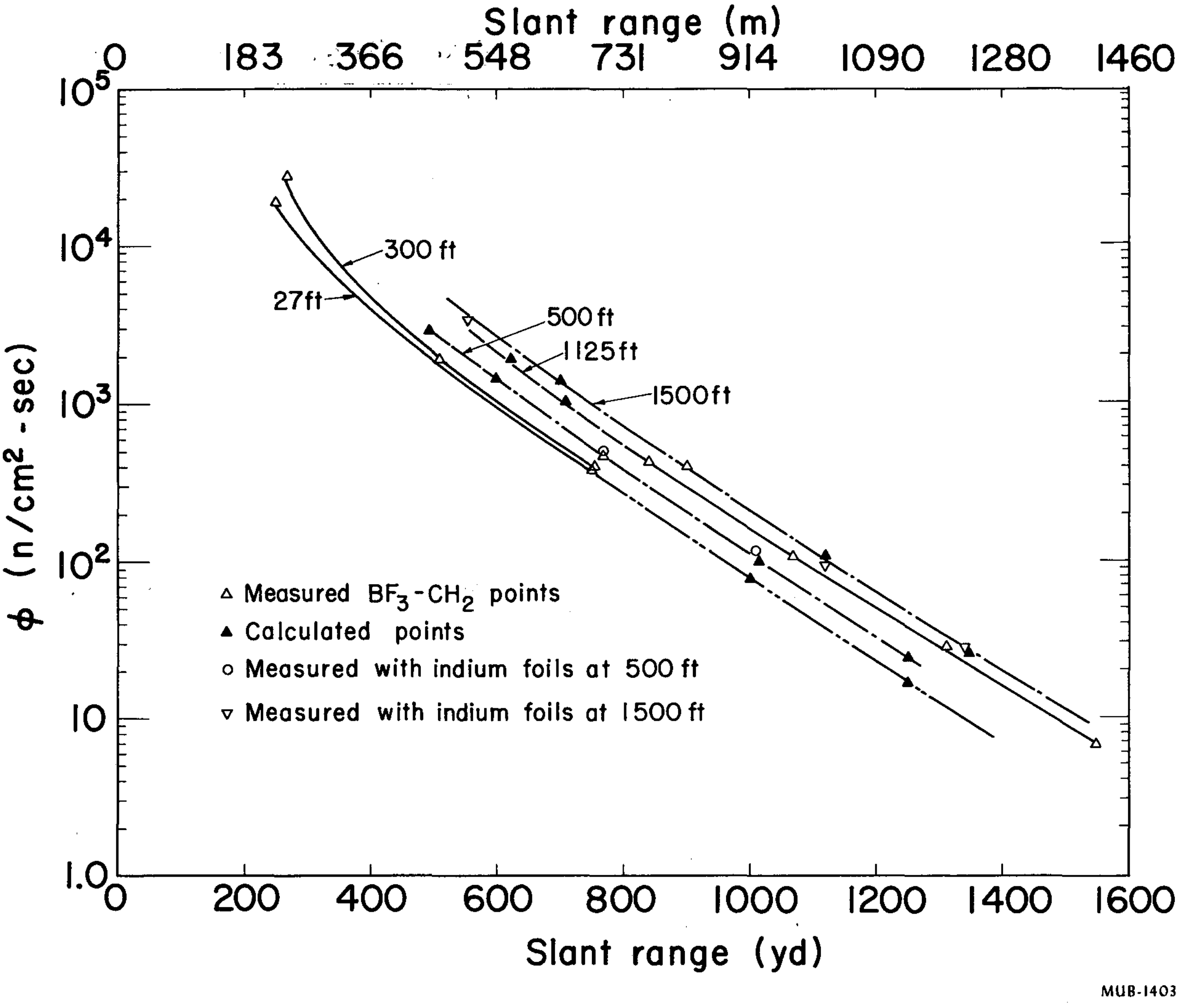

Slant range (m)
0
183
366
548
731
914
1090
1280
1460
10^5
10^4
10^3
10^2
10
1.0
ϕ (n/cm^2 - sec)
300 ft
27 ft
500 ft
1125 ft
1500 ft
△ Measured BF_3-CH_2 points
▲ Calculated points
○ Measured with indium foils at 500 ft
▽ Measured with indium foils at 1500 ft
0
200
400
600
800
1000
1200
1400
1600
Slant range (yd)
MUB-1403

Fig. 5

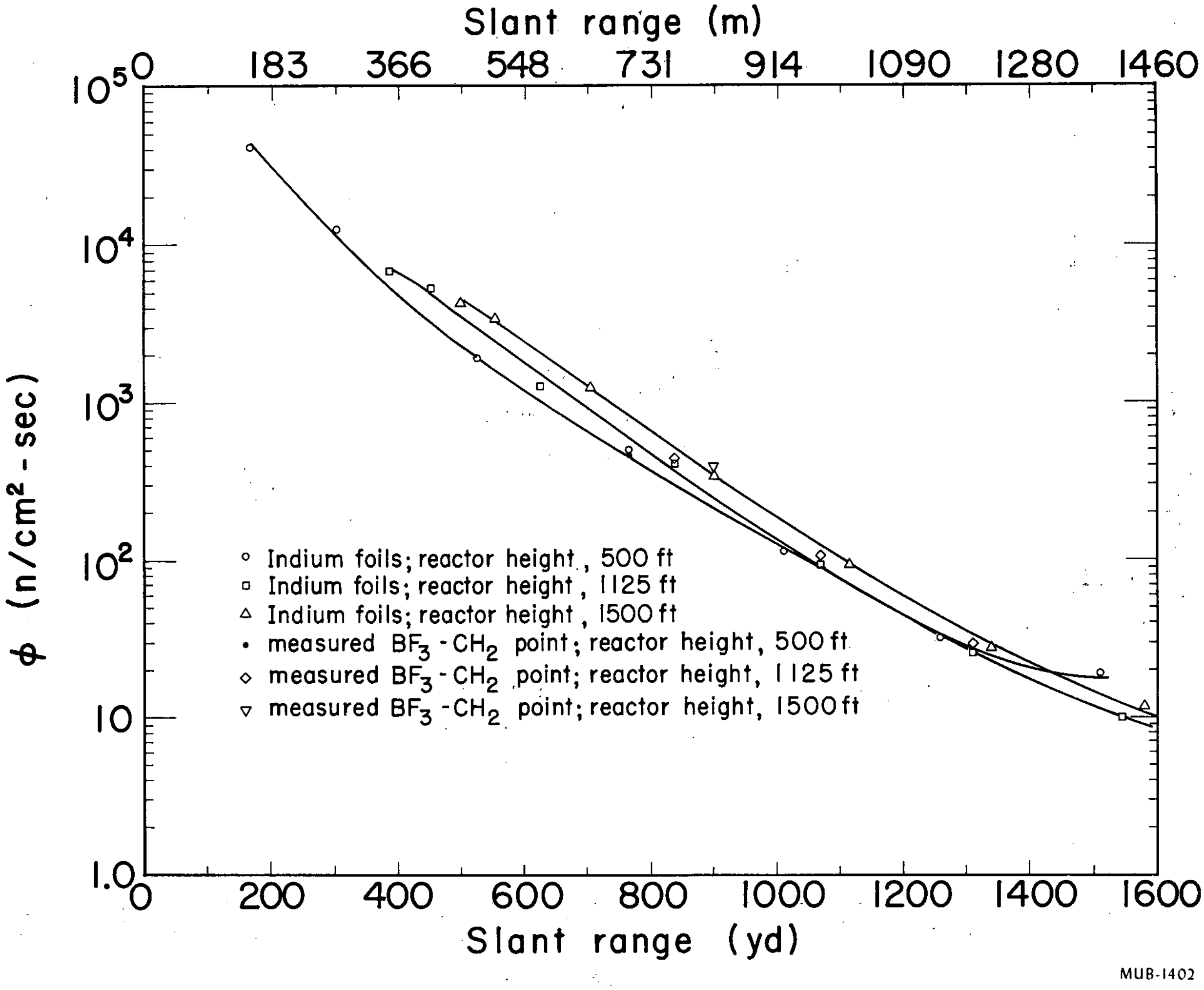
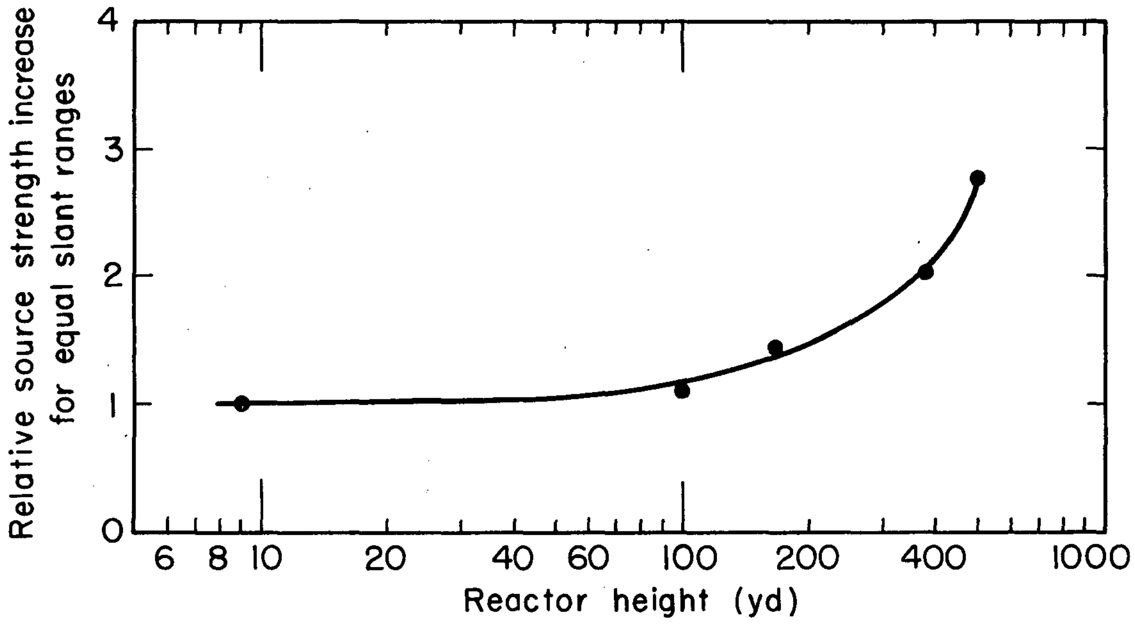
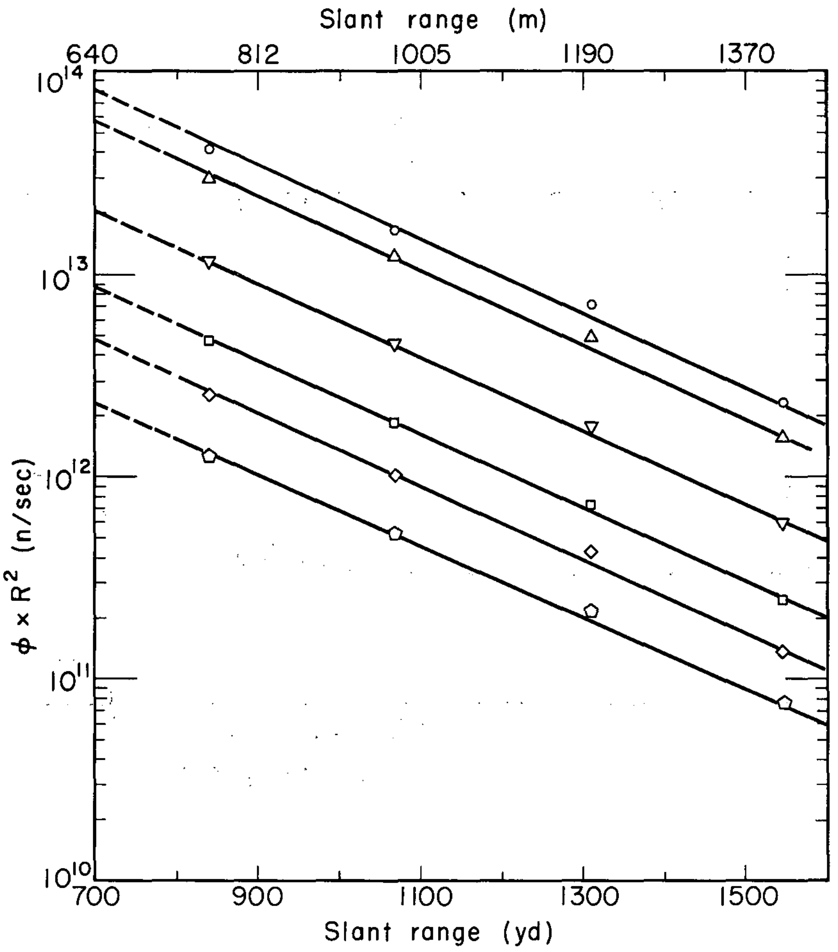

Slant range (m)
0
183
366
548
731
914
1090
1280
1460
10^5
10^4
10^3
10^2
10
1.0
ϕ (n/cm² - sec)
○ Indium foils; reactor height, 500 ft
□ Indium foils; reactor height, 1125 ft
△ Indium foils; reactor height, 1500 ft
• measured BF_3 - CH_2 point; reactor height, 500 ft
◇ measured BF_3 - CH_2 point; reactor height, 1125 ft
▽ measured BF_3 - CH_2 point; reactor height, 1500 ft
0
200
400
600
800
1000
1200
1400
1600
Slant range (yd)
MUB-1402

Fig. 6



MU-28435

Fig. 7



MU-28436

Fig. 8

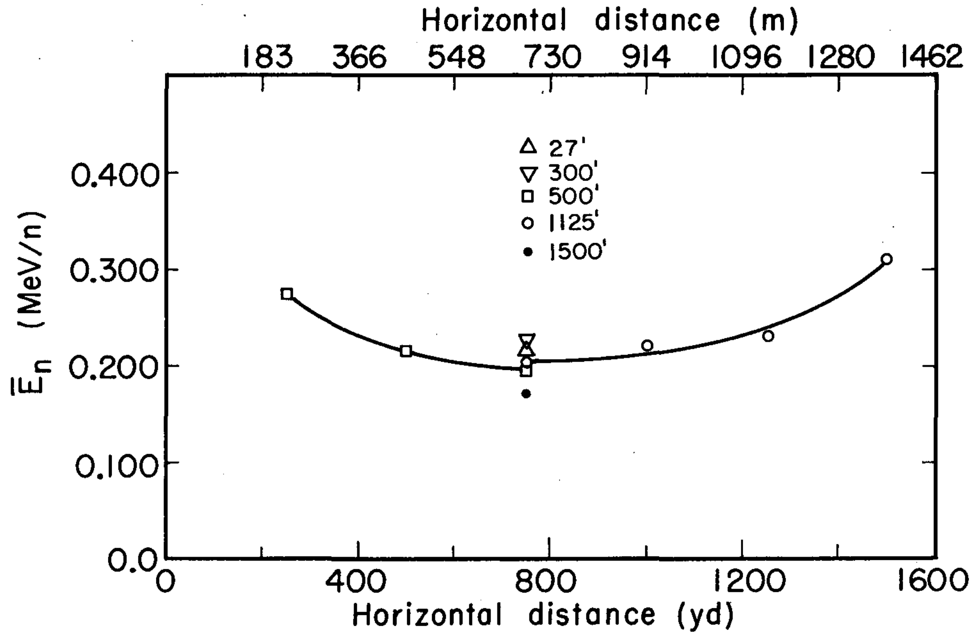


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Fig. 9