24. Accelerators

24.1. If dose rates in excess of 2.5 mrems/hr may be incurred in accessible regions outside the shielding, or in the absence of shielding at any accessible location, such regions shall be clearly marked with signs that indicate the hazard.

24.2. Any accessible region inside or outside shielding where a dose rate in excess of 7.5 mrems/hr exists shall be segregated by marked barriers that impede unintentional access. Such areas shall be provided with clear visual indication whether the accelerator is on.

24.3. If the installation is provided with a shield segregating accessible locations, and if a dose rate in excess of 7.5 mrems/hr can be received inside the shield, additional provisions shall be made for audible indication that may be either intermittent or continuous, lasting for at least a 10-second period prior to turning on the beam.

24.4. If exit from the shield cannot be effected without motion of doors or other similar impediments, provisions shall be made that:

a. Such doors are interlocked with the accelerator controls in such a way that neutron production is impossible with the door open.

b. The doors can be opened from the inside of the enclosure.

c. At least one clearly marked crash button is provided inside the enclosure to suspend accelerator operation in such a way as to make neutron production impossible. It shall be possible to readily reach one such button from any point inside the enclosure within 5 seconds of onset of the audible warning signal.

24.5. At any installation where dose rates in excess of 100 mrems per work week may be received outside the shielding, it shall be the duty of the radiation protection officer to insure that a constant check is made that persons do not receive doses in excess of the permissible levels (see sections 21.1, 21.2). "Persons" includes operators, experimenters, visitors, and individuals employed in maintenance or other duties not directly associated with machine operation. Such checks shall include a daily assessment of the operations of the accelerator.

25. Reactors

25.1. On starting up a reactor for the first time the efficiency of the shielding shall be checked minutely. Every accessible region should be surveyed for radiation, and appropriate adjustments should be made in the shielding.

25.2. Because it is possible for radiation leaks to develop, surveys should be made at least annually and whenever changes have been made in the shield or its perforations.

25.3. Because on startup of a reactor radiation levels increase strongly, an audible warning system shall be installed to insure that all personnel in the vicinity of the reactor are made aware that startup is planned. Personnel must be able to communicate with the reactor operator within a period of time that is less than the warning period.

25.4. In the vicinity of the reactor, continuous visual indication shall be provided to inform personnel whether the reactor is in operation.

25.5. Rigid procedures shall govern the changes of shielding. These shall include consultation with the radiation protection officer.

25.6. Upon first removal of a shielding block, or first testing of a beam trap placed behind a movable shutter, monitoring shall be performed to assess the existing radiation hazard.

25.7. If dose rates in excess of 2.5 mrems/hr can be incurred in accessible locations, such regions shall be clearly marked with signs that indicate the hazard.

25.8. Any accessible region where a dose in excess of 7.5 mrems/hr exists shall be segregated by marked barriers that impede unintentional access.

25.9. Because in the event of an accident it may be necessary to quickly evacuate the building, and probably also the area outside, a plan shall be devised for this evacuation. Responsible persons shall be designated and available at all times for its execution. All personnel normally in the environs shall be made aware of the plan.

26. Surveys at Accelerators and Reactors

26.1. During tune-up and initial operations, surveys of both the gamma- and neutron-radiation dose rate at accessible locations outside the shielding shall be performed as soon as radiation intensities in excess of 2.5 mrems/hr are likely to be produced.

26.2. Prior to routine operation, every accelerator or reactor shall be surveyed. Such surveys shall be repeated whenever operating conditions are changed in such a way that the neutron or gamma hazard may change. In the absence of any such changes, surveys shall be made at

least once a year. In installations where liquid shields are employed, a survey shall be performed at least once every 6 months.

26.3. At installations where liquid shields are employed, special precautions shall be taken to insure that the liquid is maintained at the desired level.

26.4. When personnel are exposed to dose rates in excess of 80 mrems/hr, continuous personal monitoring shall be performed unless reliable estimates of the dose received can be obtained by other means.

27. Health

27.1. Prior to beginning of employment, each person working more than occasionally in areas where significant neutron doses may be received shall have a medical examination. It should be recognized that the examination is directed toward determining the normal or presumed "pre-irradiation" condition of the worker, and toward taking especial precautions if any abnormalities exist that might later be confused with radiation damage. The examination shall include a complete blood count, with determination of erythrocyte, leukocyte, and platelet levels, clotting time, and differential W.B.C. An eye examination shall include a determination of vision with and without glasses, dilation of pupils and examination of the lenses with the slit lamp (corneal microscope). Routine examinations as above shall be done once a year, unless more frequent examinations are indicated by readings of monitoring instruments.

28. Overexposure

28.1. Any exposure in excess of the permissible limit shall receive the immediate attention of the radiation protection officer and the management. Corrective measures shall be instituted to prevent reoccurrence and steps shall be taken to reduce individuals' average exposure.

28.2. An individual who has been exposed to more than 12.5 rems of ionizing radiations in a period of less than 1 month shall be deemed to have suffered an overexposure sufficiently grave to require execution of the steps set forth in 28.3.

28.3. Considerable effort shall be made to determine the dose received, the portion of the body exposed, and the character of the radiation. The legal representative of the institution shall be notified of the amount of exposure, the extent of the present injury, and the probable outcome. A competent panel, including a physician familiar with radiation risk and injury, a physicist, the radiation protection officer, and a responsible administrator, shall review the medical findings and decide on the advisability of renewed examinations. They shall investigate the reasons for the overexposure and undertake all reasonable efforts to prevent repetition of overexposures.

Appendix 1. Depth Dose

Experimental data on the neutron depth doses resulting from exposure to monoenergetic neutrons are not available. However, results have been reported from Po-Be and Po-B sources. In these two instances, the neutron dose I at a depth X, in terms of the dose I_0 at the surface of a phantom situated at a distance D from the source, can be expressed as

$$I = \frac{I_0}{(D+X)^2} b e^{-X/L}, \qquad (4)$$

where b and L are parameters characteristic of the source and the size of the tissue equivalent phantom, as follows:

Source	Phantom	b	L	Reference[a]
			cm	
Po-B	Small	1.20	10	(1)
Po-B	Semi-infinite	1.56	7.5	(2)
Po-Be	Small	1.38	11.39	(1)

[a] References: (1) W. A. Mills and G. S. Hurst, Nucleonics 12, 17 (April 1954); (2) T. A. Barr and G. S. Hurst, Nucleonics 15, 58 (August 1954).

Theoretical depth dose curves have been computed by Snyder for a variety of neutron energies. The curves shown in figures 2 to 14 apply to an infinite slab of tissue-equivalent material 30 cm thick, irradiated by a broad beam of monoenergetic neutrons with 1 neutron/cm² entering normal to one face of the slab.
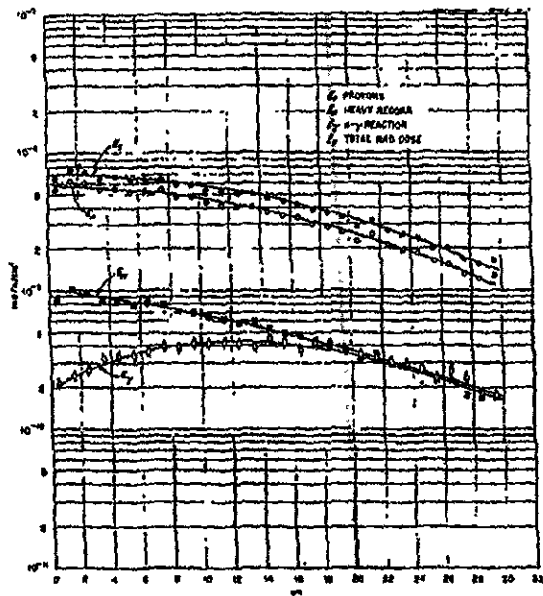
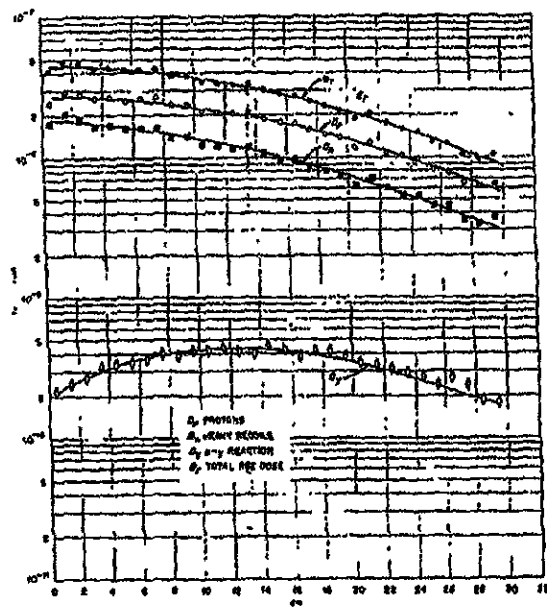FIGURE 2A. Absorbed dose (rads) from 10-Mev neutron beam.

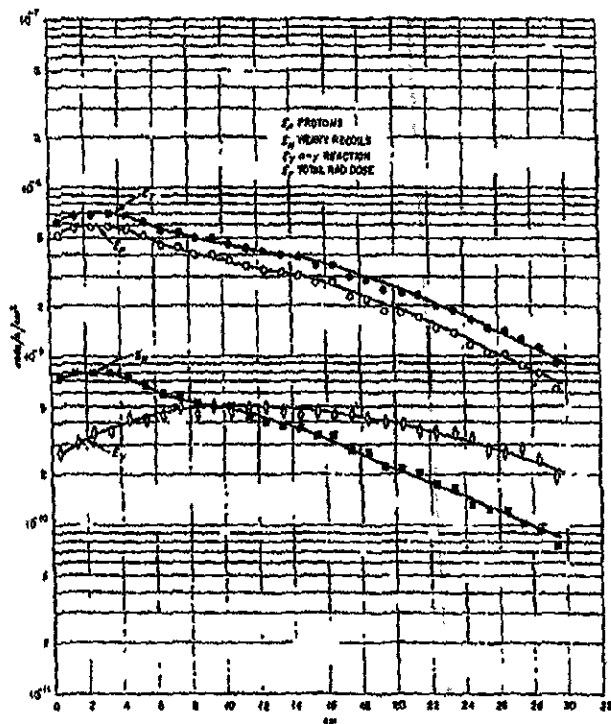FIGURE 2B. RBE dose (rems) from 10-Mev neutron beam.

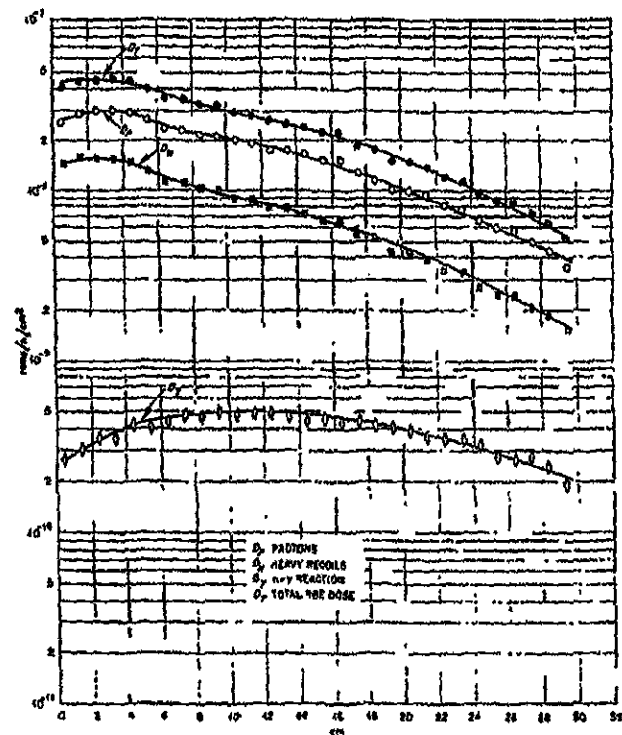FIGURE 3A. Absorbed dose (rads) from 7.5-Mev neutron beam.

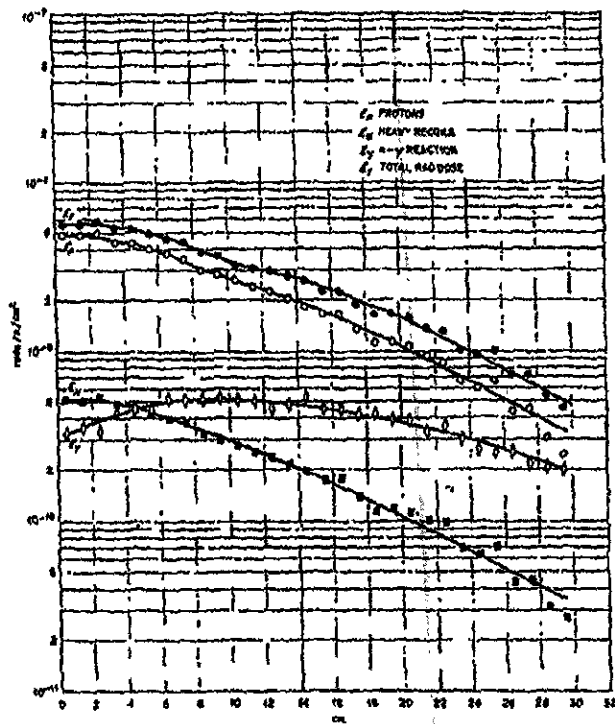FIGURE 3B. RBE dose (rems) from 7.5-Mev neutron beam.

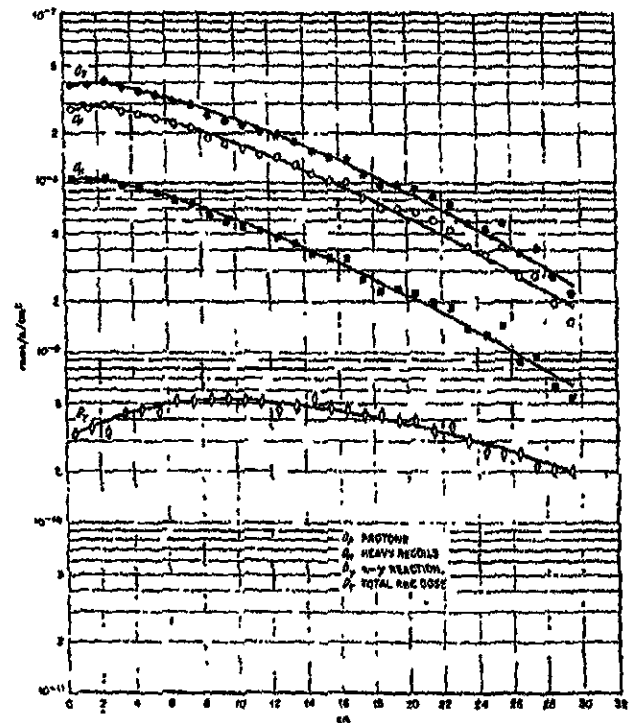FIGURE 4A. *Absorbed dose (rads) from 5-Mev neutron beam.*

FIGURE 4B. *RBE dose (rems) from 5-Mev neutron beam.*

FIGURE 5A. *Absorbed dose (rads) from 2.5-Mev neutron beam.*

FIGURE 5B. *RBE dose (rems) from 2.5-Mev neutron beam.*

FIGURE 6A. *Absorbed dose (rads) from 1-Mev neutron beam.*
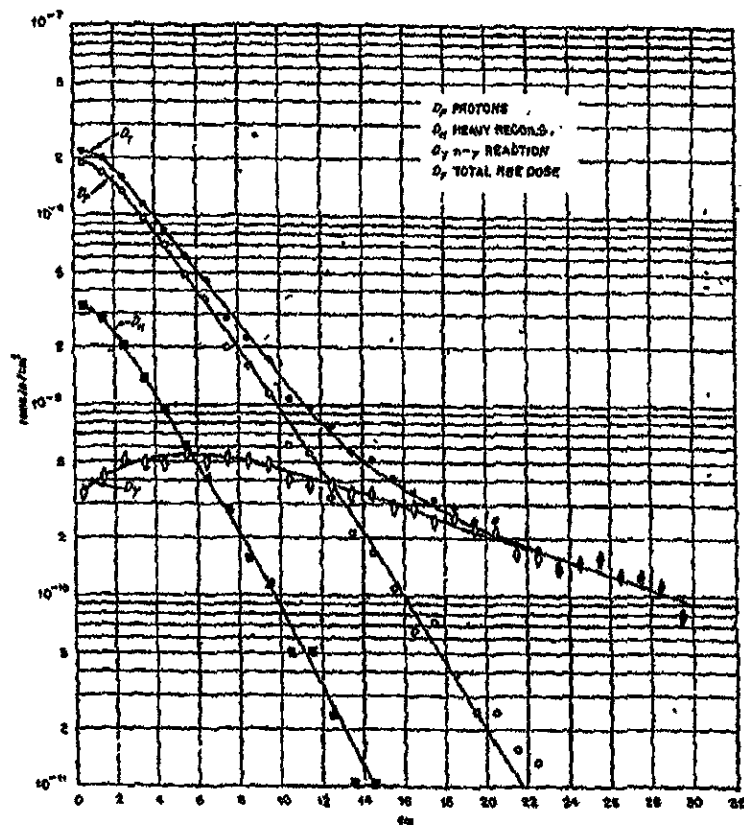
FIGURE 6B. *RBE dose (rems) from 1-Mev neutron beam.*
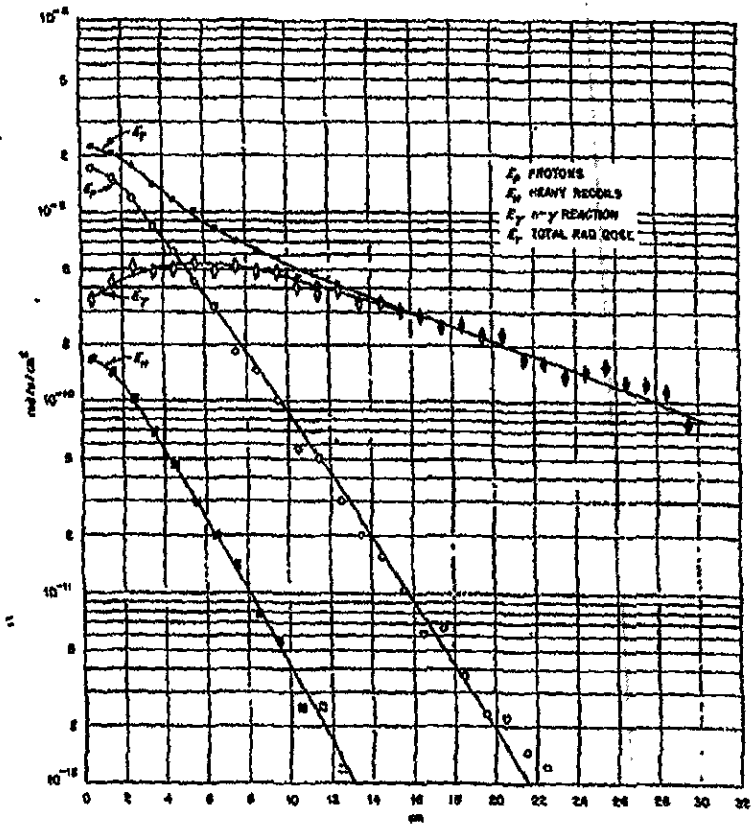
FIGURE 7A. *Absorbed dose (rads) from 0.5-Mev neutron beam.*

FIGURE 7B. *RBE dose (rems) from 0.5-Mev neutron beam.*

FIGURE 8A. *Absorbed dose (rads) from 0.1-Mev neutron beam.*

FIGURE 8B. *RBE dose (rems) from 0.1-Mev neutron beam.*

FIGURE 9A. *Absorbed dose (rads) from 0.02-Mev neutron beam.*

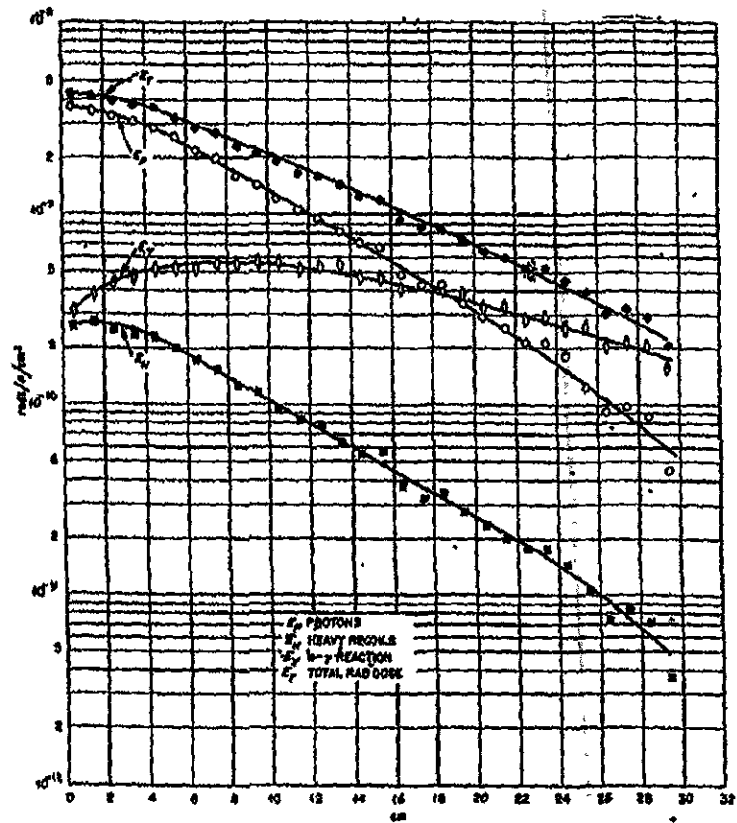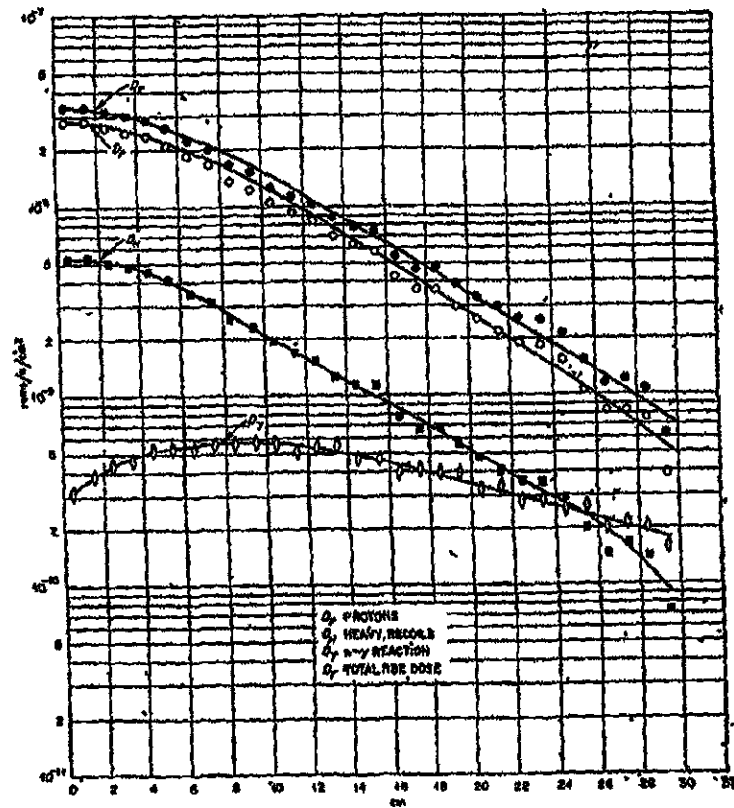FIGURE 9B. *RBE dose (rems) from 0.02-Mev neutron beam.*

FIGURE 10A. *Absorbed dose (rads) from 0.005-Mev neutron beam.*

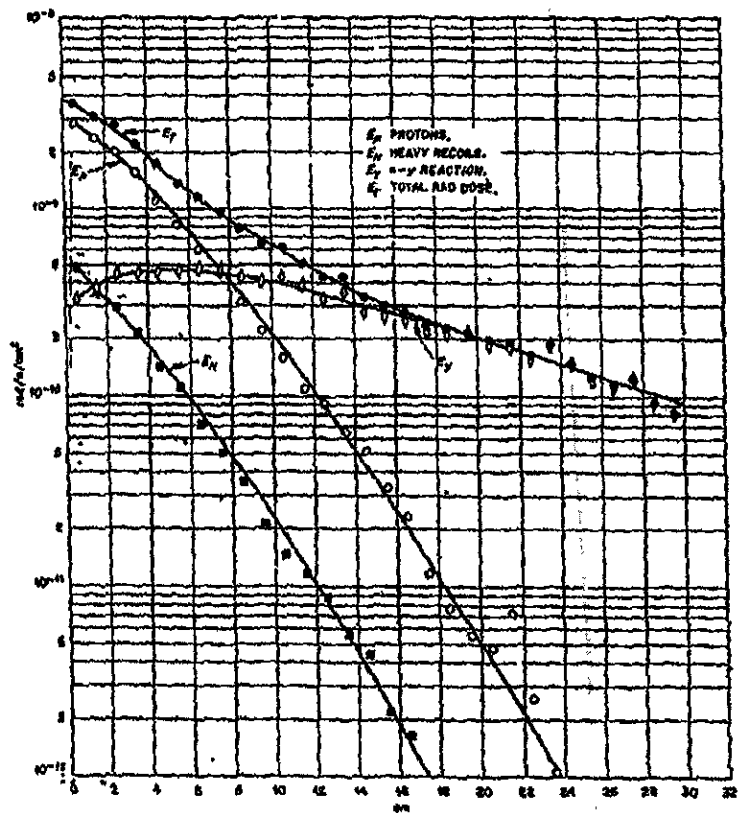FIGURE 10B. *RBE dose (rems) from 0.005-Mev neutron beam.*

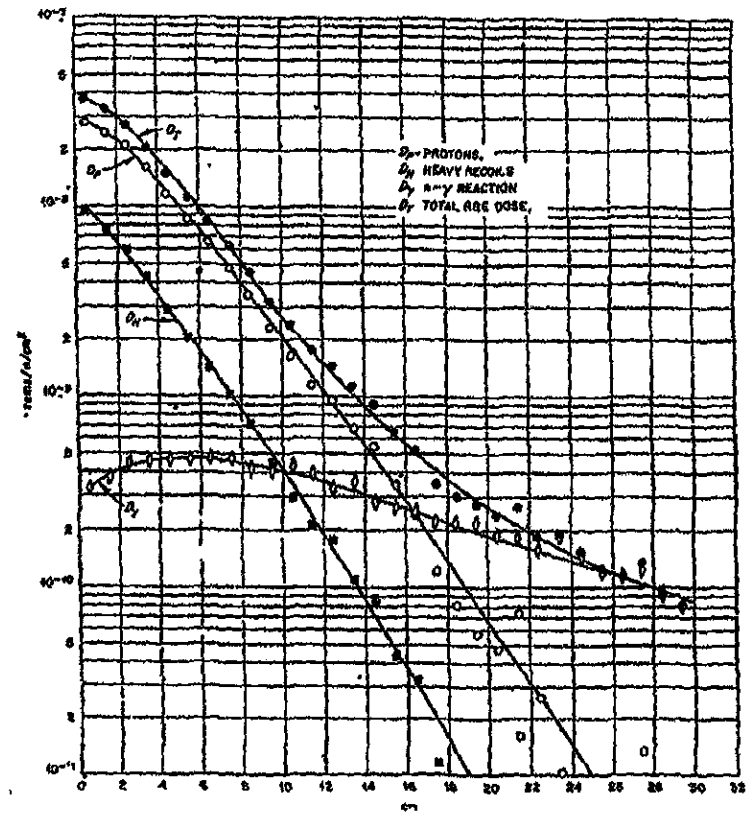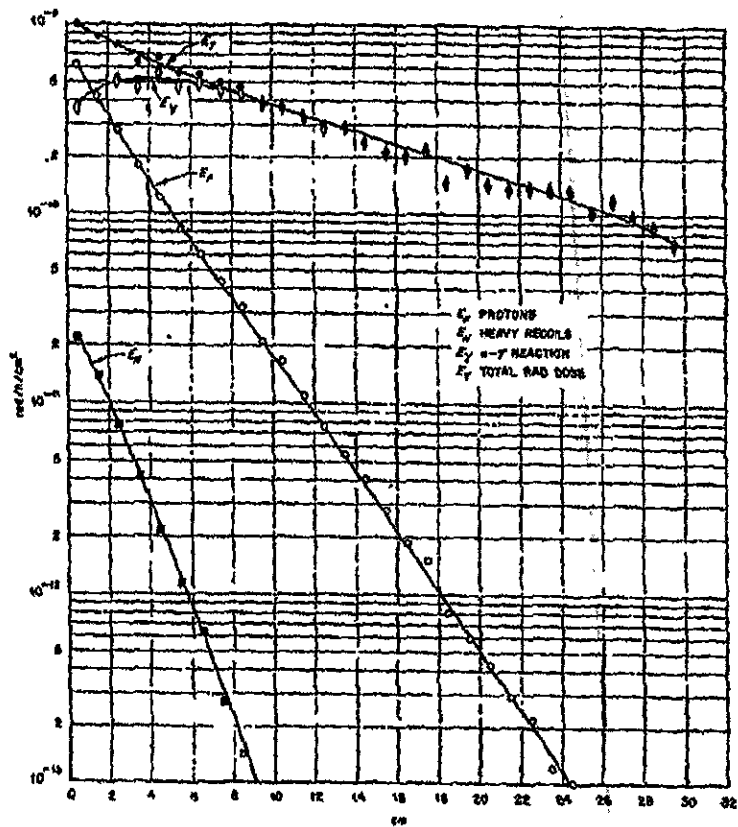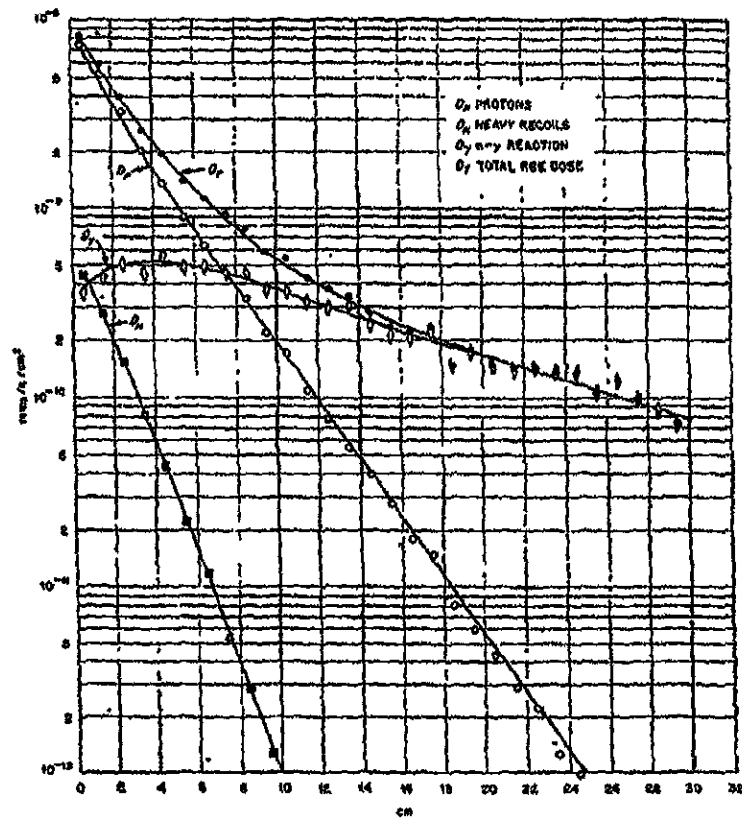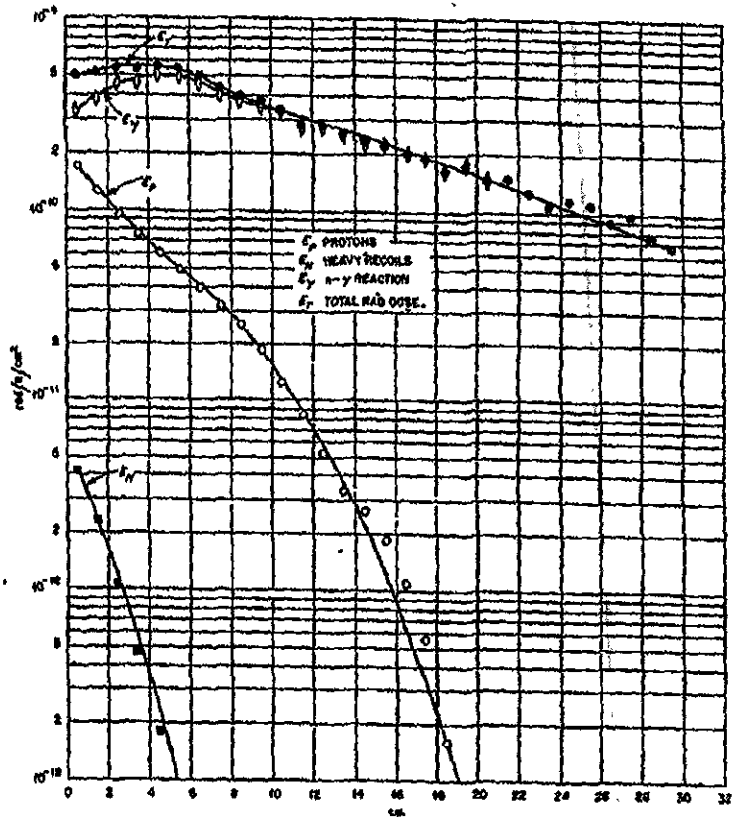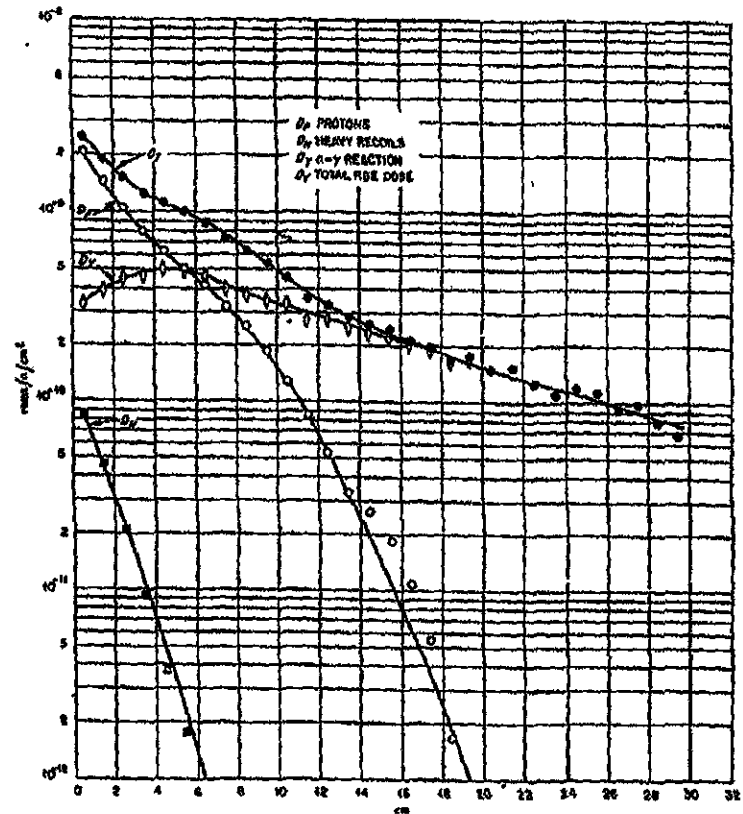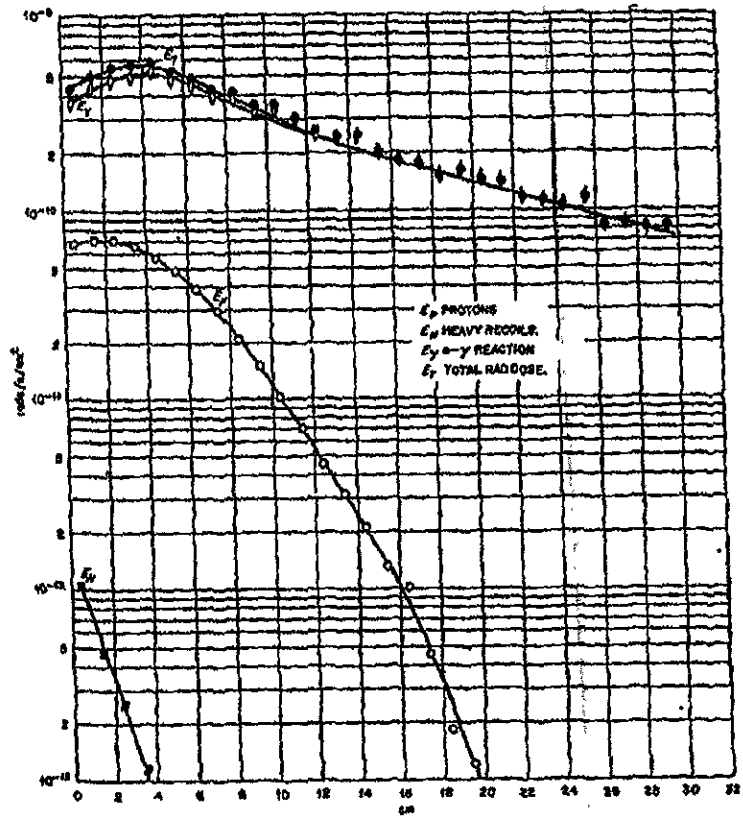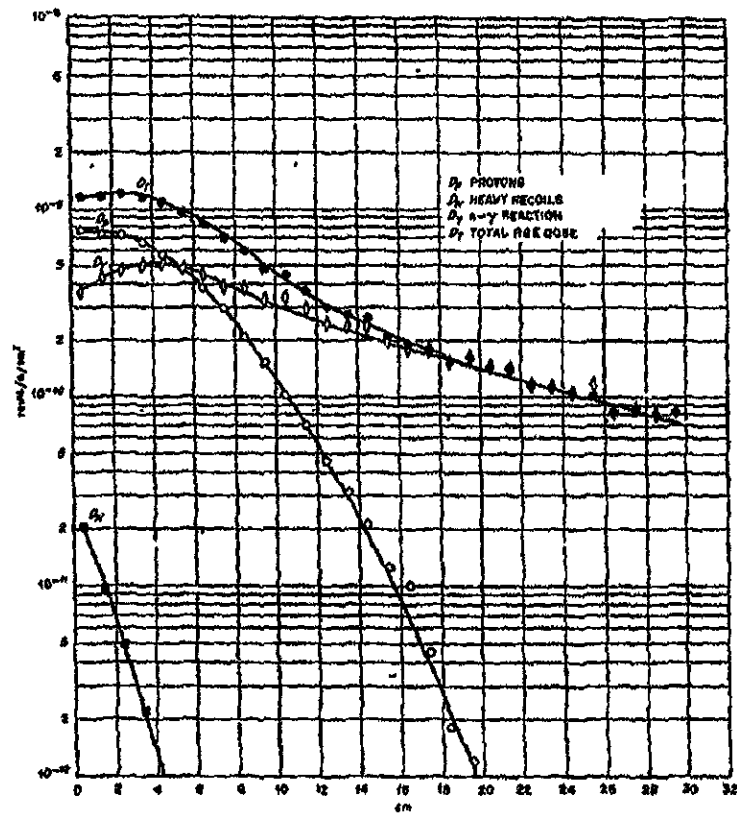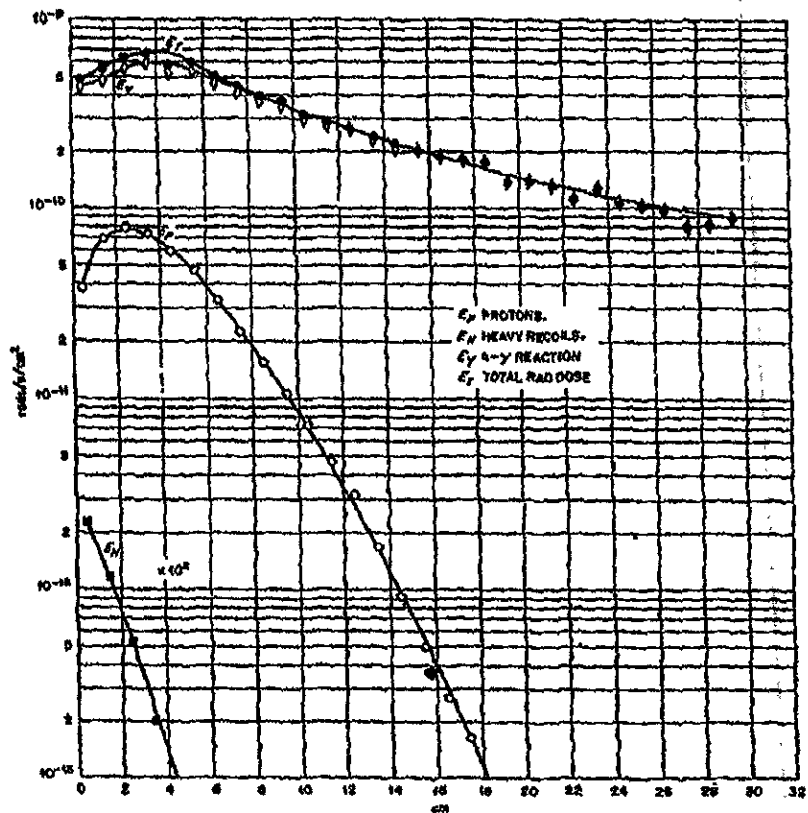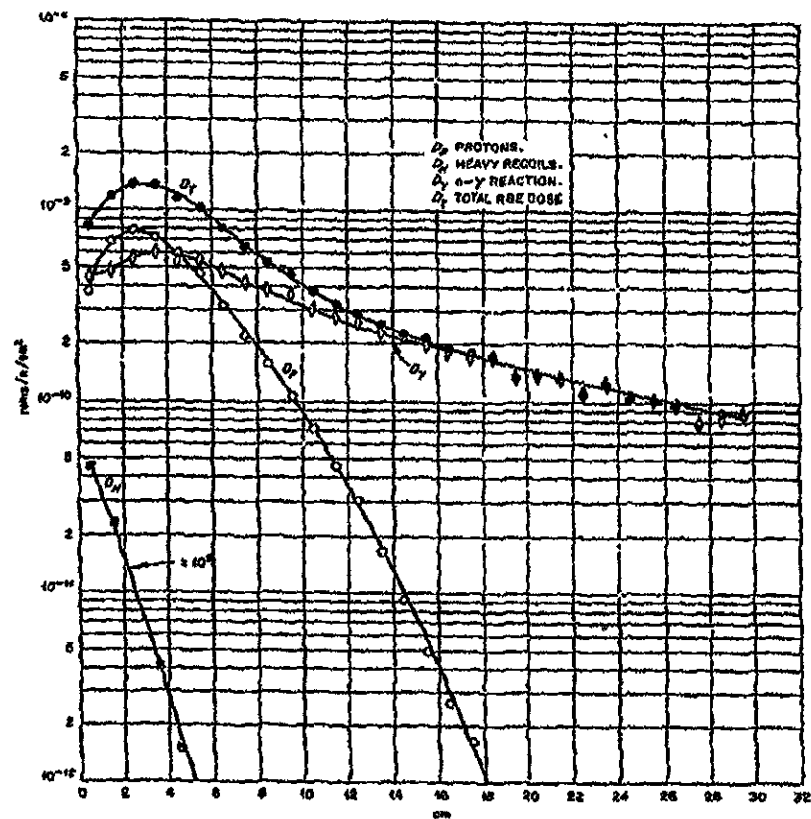FIGURE 11A. *Absorbed dose (rads) from 0.0001-Mev neutron beam.*

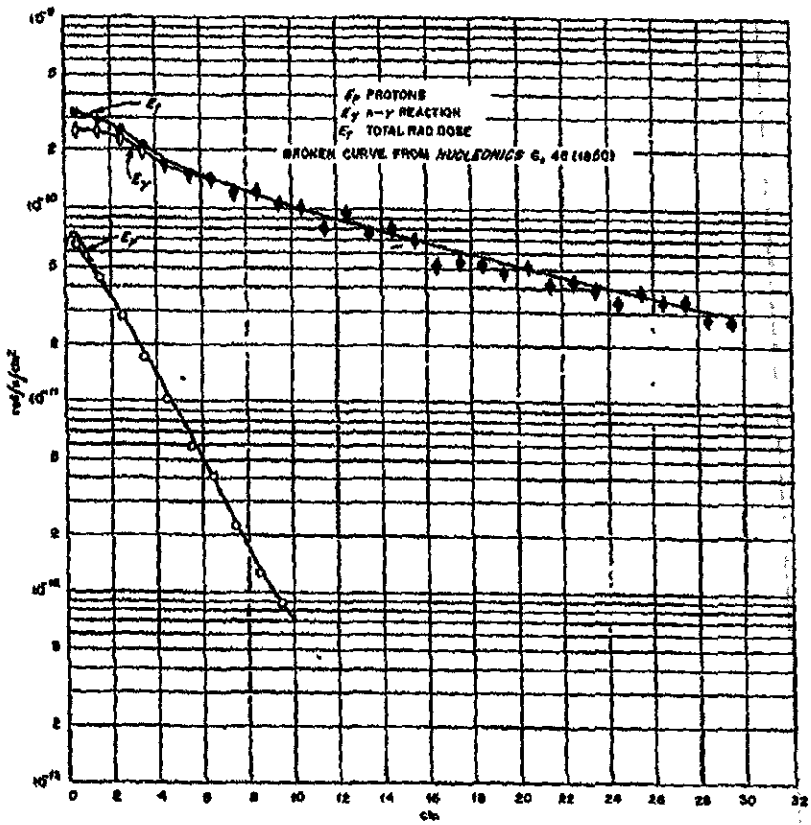FIGURE 11B. *RBE dose (rems) from 0.0001-Mev neutron beam.*

FIGURE 12A. *Absorbed dose (rads) from thermal neutron beam.*
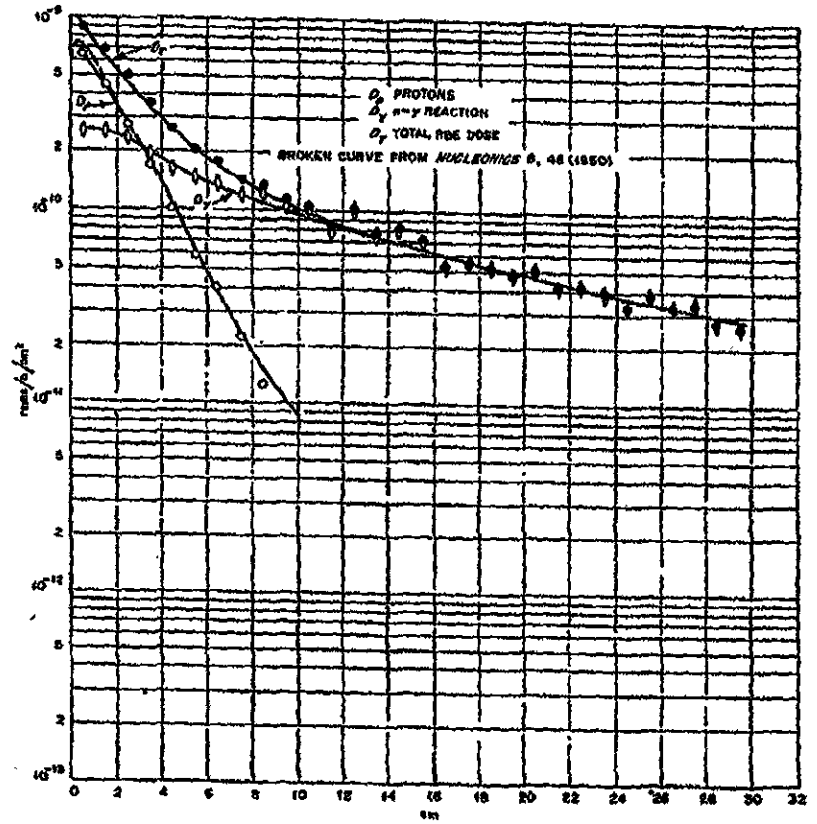
FIGURE 12B. *RBE dose (rems) from thermal neutron beam.*
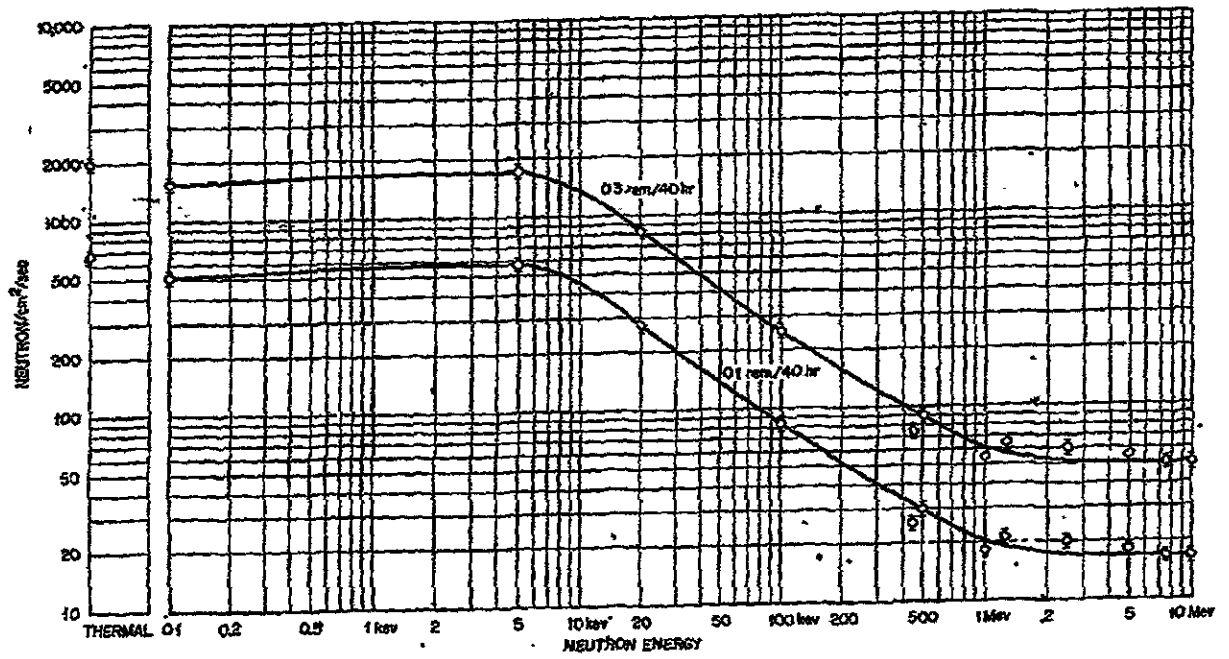
FIGURE 18. *Flux of neutrons to deliver the stated dose rate as a function of neutron energy.*
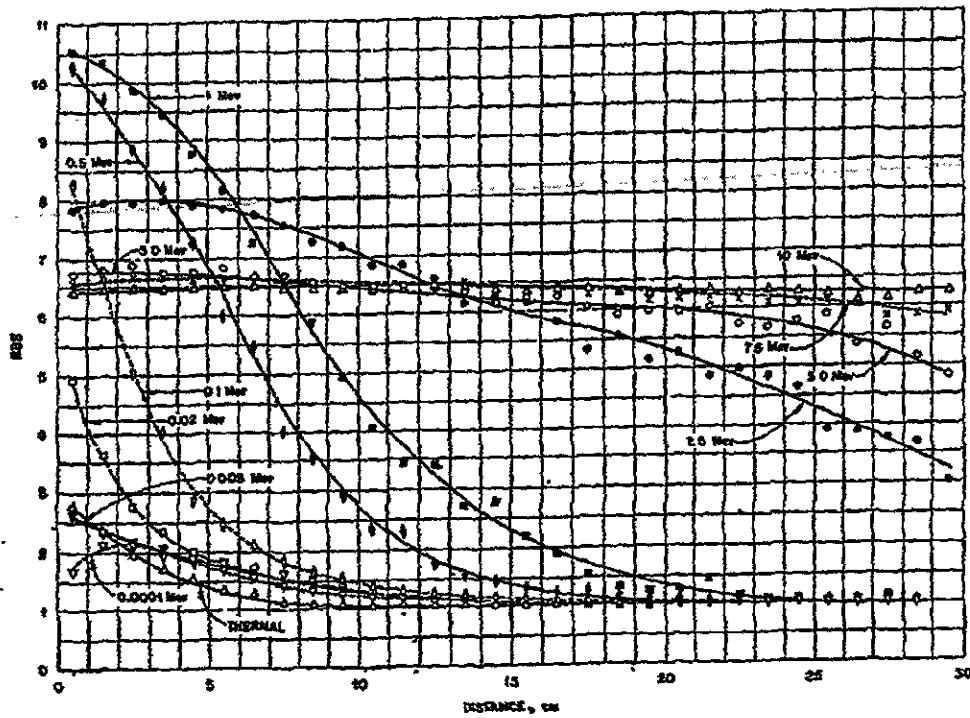
FIGURE 14. *RBE* ($=D_n/E_x$) *as a function of depth and neutron energy.*

FIGURE 15. Effect of buildup on neutron dose.

These curves were obtained by computing neutron histories in an unbiased manner and averaging the absorbed dose delivered from a sample of such histories. The minimum sample size for a given neutron energy was 4,000 such histories. In computing these histories, tissue was considered to consist of hydrogen and one heavier element. The percentage by weight of this heavier element was taken as the sum of the percentage weights of all body constituents other than hydrogen, and the cross section for this element was a composite of the cross section of these elements. All scattering was considered to be isotropic in the center of mass system of coordinates and the only reactions considered were elastic scattering, the (n,p) reaction with nitrogen, and the (n,γ) reaction with hydrogen. The energy of the recoil atoms and of the protons produced by the (n,p) reaction was considered as absorbed at the site and the absorbed energy was tabulated and averaged over each centimeter of depth to obtain the dose curves.

Handbook 59 of this series has prescribed the RBE as a function of specific ionization. The specific ionization as a function of particle energy has been given by Livingston and Bethe [8] and for the heavier elements of principal interest here has been approximated by Neufeld and Snyder [9]. The latter reference gives RBE as a function of the particle energy as well as the biological dose delivered by a particle of given energy. These curves were approximately by polynomials, and the biological data were computed for each particle produced by the neutrons in the course of their histories. The γ-rays produced were followed also using the Monte Carlo or sampling method, and their absorbed dose (—RBE dose) computed for each centimeter of depth.

The points given on the graphs are the computed points, and the smooth curve indicates the general trend. In some cases where the penetration was small, few or no events occurred in some intervals and the curves were not continued where the data did not seem significant. Figure 14 gives the RBE (—RBE dose/absorbed dose) as a function of energy and depth. The flux to deliver a dose of 0.8 rem in 40 hours or of 0.1 rem in 40 hours has been computed and is shown in figure 13. The ratios of maximum dose for the 30-cm slab to first collision dose for both absorbed dose and for RBE dose are shown in figure 15. The low values at 1 Mev and 0.44 Mev are not entirely due to statistical fluctuations but are largely due to the resonance peaks in the oxygen cross-section curve. As a check on
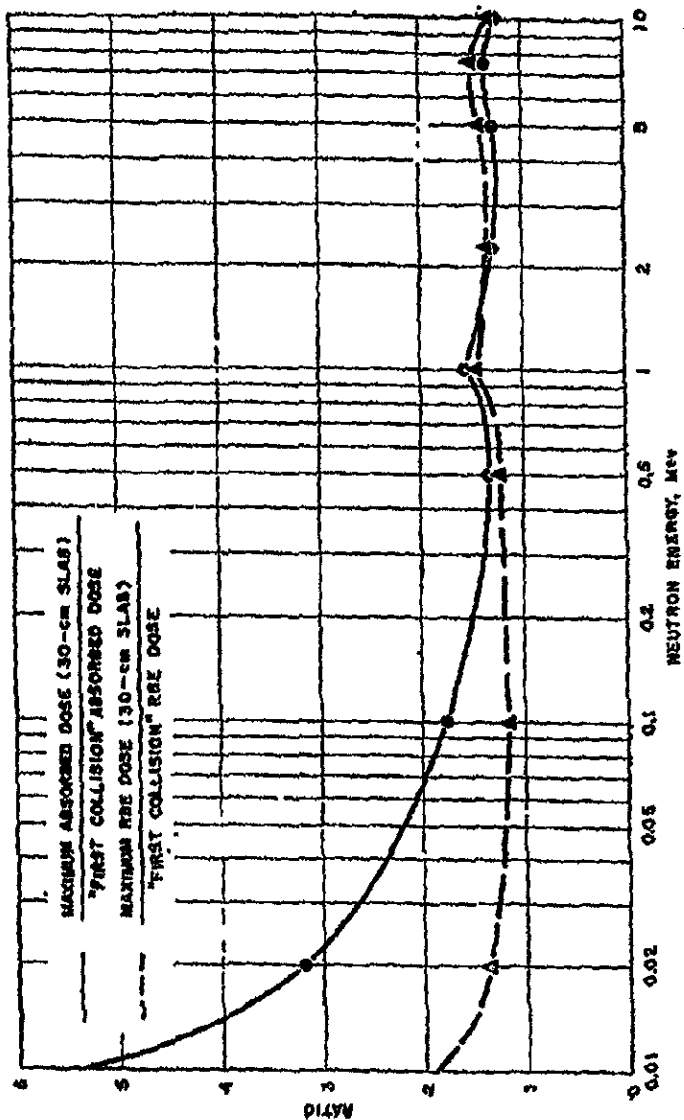
this, the points at 1.2 Mev and at 0.5 Mev were computed, and this confirmed that the actual curve will reflect much of the detailed structure of the cross-section curves. However, the accuracy of the present study would not warrant attempting to predict the amount involved in detail.

For thermal neutrons an earlier study by Snyder [10] indicated a sharp peak in the absorbed dose due to the (n,p) reaction. The present calculations averaged out this peak. Figure 12 shows the peak as well as the average computed in the present study. In computing the thermal value for figure 18 the peak value was used.

The reflection of thermal neutrons incident on a large slab of tissue is approximately 80 percent. Hence the reading obtained with pocket dosimeters worn for purposes of thermal-neutron dosimetry should be corrected accordingly (i.e., the reading should be divided by 1.8 before the hazard is evaluated according to the data presented here).

Appendix 2. Flux Detectors and Their Calibration

As the methods of flux measurement are different for the various energy ranges, even though the principles involved are about the same, they will be considered according to neutron energy. Flux measurement of thermal neutrons has received most effort because it is basic to most flux measurements in the other energy classifications.

A. Thermal Neutrons

One approach to the absolute measurement of thermal neutron flux is closely related to source calibration, because it is possible to produce a flux with a standard source that is known almost as accurately as the emission rate of the source. The thermal flux is produced in a "standard pile," which is merely a large block of graphite in which the source, usually a Ra-Be source, is placed. The fast neutrons emitted from the source are moderated in the graphite and produce a certain spatial distribution of slowing-down density, q. When thin foils of indium, wrapped in cadmium, are activated at various places in the standard pile, the activation is proportional to the neutron flux at the resonance energy of indium, 1.44 ev, hence to the q at 1.44 ev. The absolute q values are obtained in terms of the source strength Q because the integral of q over the

volume of the standard pile, assuming negligible capture and no escape of resonance neutrons from the pile, must equal Q. Once the slowing down density q is known in the standard pile it is a simple matter to compute the thermal flux from q, for it is necessary to know only the lifetime of the neutrons in graphite, easily obtained from the scattering constants of carbon.

The standard pile flux, while reasonably well known, is quite low, and activations intense enough for accurate foil counting can be obtained only with foils that are so thick that an appreciable change in the flux at a given point results from the neutron absorption of the foil. The flux perturbation does not affect the determination of an unknown flux relative to that in the standard pile if the unknown flux is also in graphite and if the foil geometry is identical, for in that case both fluxes will be equally perturbed. However, a correction for the perturbation must be made when fluxes in open beams are to be measured relative to the standard pile, for instance, and for this geometry the correction leaves an error of several percent.

A second method of absolute measurement of neutron flux is by means of the reaction rate of an element whose cross section is known, for

$$\text{reactions/sec} = nvN\sigma, \qquad (5)$$

where N is the number of atoms of reaction cross-section σ in the flux nv. The absorption cross section of boron, for example, which is essentially the (n,α) reaction, as the (n,p) and (n,γ) are negligible, has been accurately determined from the total cross section. The "best value" of the boron absorption cross section, from the AEC compilation, is 750 barns at 2,200 cm/sec, accurate to about 1 percent. The number of (n,α) reactions taking place in a certain amount of boron, located in a particular neutron flux, then gives the flux in terms of the reaction rate. Again, as for the standard pile, the rate of (n,α) reactions in boron depends on the neutron density, independent of the velocity, because boron is strictly $1/v$. There are several other elements for which the absorption cross section can be accurately determined from the total cross section by subtraction of scattering, for instance gold or indium. These elements differ from boron as flux standards because their absorption is a result of radioactive capture rather than the (n,α) process, and in addition they are not strictly $1/v$.

The reaction rate in a known amount of boron located in a neutron flux can be measured by a BF$_3$-filled proportional

counter or pulse-counting ion chamber, if one can be sure of a one-to-one correspondence between disintegrations and counts. The counting rate corresponding to complete detection of the disintegrations in ionization chambers has been obtained with an accuracy of about 2 percent. To this error must be added the uncertainty in the boron cross section, so the final accuracy is probably about the same as that of the standard pile flux. The absolute disintegration rate in boron has not been used in any determination of neutron flux to better than 5 percent, although the accuracy could definitely be increased.

The simplest method of *relative* flux measurement is by means of activation of a 1/v material in the form of a thin foil. Manganese, for instance, is a good material for relative determination of high flux because it has a convenient half life (2.6 hours), and a cross section (12 b) that is small enough so that excessively thin foils are not necessary. A thin foil of manganese activated in any thermal flux, and in a standard flux, will give the former flux (again, actual density times 2,200 cm/sec) correctly, regardless of the physical arrangement of the neutron flux, whether it be in graphite or some other moderator, in a parallel beam, or any arbitrary angular distribution. Because the foil is thin, there is no flux depression caused by its presence nor any self-protection of the foil itself. The activation of a thin foil does not depend on the direction of neutron travel through the foil. This is shown in the activation formula above, where flux is just the neutron density multiplied by the velocity, with no directional effects involved.

B. Intermediate Neutrons

There has been no extensive work on flux measurements for intermediate neutrons, primarily because there has been little need for accurate flux values in this energy range. For the purpose of neutron research, in which primarily total cross sections have been measured, only relative flux measurements, used in transmission measurements, are required. The intermediate neutrons in bulk matter, as in shields or in tissue, are soon slowed to thermal, especially in light elements, and spend most of their lifetime as thermal neutrons. Because of the rapid moderation, and the strong "self-protection" that reduces interaction in bulk material, intermediate neutrons are much less important than fast or thermal neutrons.

C. Fast and Relativistic Neutrons

Because fast neutrons represent such a wide energy range, over which detector response varies widely, accurate flux measurement is extremely difficult. For these neutrons the simplest method of flux measurement is by comparison with the flux emitted by a standard neutron source, usually a Ra-Be source.[7] For this comparison the standard source is not used in the standard pile but is placed in the center of a large room, and the flux of fast neutrons at a certain distance from the source is calculated from geometrical considerations. This flux is thus established with essentially the accuracy of the source calibration, about 5 percent usually. The unknown flux is then compared with this standard flux by some type of detector that will not be affected greatly by any energy difference between the unknown and the standard flux. Unfortunately, most fast neutron detectors are not energy-independent. The long counter is specifically designed for flux comparison, however, and can be used in the energy region 0.1 to 3 Mev, in which its sensitivity varies by less than 5 percent.

Fast neutrons, in contrast to those of lower energy, can be detected by means of the recoil protons they produce, and in principle these protons can be used for flux measurement. It is difficult, however, to measure the recoil protons quantitatively, for example, in an ionization chamber, and thus obtain the absolute flux incident on the recoil counter. In practice an accuracy of about 10 percent can be reached by this method. In a similar manner, an absolute flux can be measured by counting recoil protons in a photographic emulsion, but again measurements better than 10 percent are very difficult.

Activation of foils of known cross section, which is so useful for thermal flux measurement, is much more difficult to apply to fast neutrons, primarily because of the wide energy range of fast neutrons. In bulk material, moderation produces an energy spread, and a foil that has a cross section varying rapidly with energy does not give a readily interpretable result. Use of threshold reactions as (n,p), (n,α), (n,2n), and (n,f) can give approximate information on the flux value in different energy regions, depending on the reaction thresholds. Table 4 gives occasionally employed reactions and their thresholds.

[7] In computing the neutron flux from a standard source or in measuring its strength relative to another source by means of the long counter, it must be remembered that the emission of neutrons from sources is often anisotropic.

TABLE 4. *Threshold detectors*

Detector	Reaction	Product	aT½	bEt	cσ	Referencesd
				Mev	mb	
N_p^{14}	(n,f)	Many	Many	0.4	1500	(2)
Th^{232}	(n,f)	Many	Many	1.3	170	(2)
Pa^{231}	(n,f)	Many	Many	0.01	2000	(2)
U^{238}	(n,f)	Many	Many	1.3	550	(2)
P^{31}	(n,p)	Si^{31}	2.7 hr.	1.0	120	(1)
Ag^{107}	(n,p)	Pd^{107}	18 hr	0.4	2.0	(1)
S^{32}	(n,p)	P^{32}	14.3 days	1.0	300	(1)
Ti^{48}	(n,p)	Sc^{48}	44 hr.	1.3	23	(1)
Mg^{24}	(n,p)	Na^{24}	15 hr.	2.1	33	(1)
Fe^{54}	(n,p)	Mn^{54}	2.6 hr	2.6	13.5	(1)
Ag^{107}	(n,2n)	Ag^{106}	24.5 min	9.4	45	(1)
O^{16}	(n,2n)	O^{15}	20.5 min	20.2	25	(1)

aT½=half life of product nucleus.
bEt=threshold neutron energy.
cσ=approximate cross sections at energies well above threshold.
dReferences: (1) B. L. Cohen, Phys. Rev. 81, 184 (1951); Nucleonics 8, No. 2, 29 (1951); (2) BNL-325; Rev. Sci. Instr. 27, 155, (1956).

Appendix 3. Reaction Employed in Neutron Production

Table 5 summarizes pertinent data on reactions utilized in neutron production. Both cross sections and neutron yield are a function of bombarding voltage, which is given in Mev in parentheses; at other bombarding energies corresponding values may be quite different.

The information in the last two columns is qualitative only and is furnished as a rough guide.

For further information see references [11 to 16].

Appendix 4. Practical Use of Radiation Instruments

For health protection in the vicinity of a neutron source, measurements are needed of the radiation hazard from fast neutrons, intermediate neutrons, thermal neutrons, and gamma rays. A brief discussion of fundamentals will be given here. For further information the reader is referred to several review articles [17, 18, 19], and to the references below.

A. Instruments

1. *Fast neutrons and relativistic neutrons (10 kev to 30 Mev)*. As most tissue damage by fast neutrons is probably

due to ionization from recoil protons, the fast-neutron dose in rads is usually assessed with an ionization chamber or counter filled with a hydrogen-rich gas. Fast neutrons may be distinguished from gamma rays by (a) use of two ionization chambers, one sensitive to both neutrons and gamma rays, and the other sensitive to gamma rays alone or having a known residual neutron sensitivity (the difference in reading permits evaluation of the neutron dose), and (b) use of a proportional counter which is biased to discriminate against small pulses due to gamma rays. Examples of the ionization-chamber method are tissue-equivalent chambers [18] and CH_2-C and CH-C pairs of chambers [19]. Proportional counter instruments reading fast-neutron dose include the polyethylene-ethylene proportional counter [20] and the count-rate dosimeter [21].

The polyethylene-ethylene proportional counter may be used with a special circuit in which the pulses produced by heavy particles are weighted according to pulse height and a count results that is proportional to dose. This is sometimes called the pulse energy integrating dosimeter. This instrument is not directional, but requires somewhat complicated electronic circuitry. The count-rate dosimeter, however, is directional. Both dosimeters, being pulse-operated instruments, usually cannot be employed at sources that produce radiation in very short bursts for reasons that are given above (10.7). A scintillation-counter version of the count-rate dosimeter, in which an attempt has been made to eliminate directional response, has been reported [22].

Other fast-neutron survey instruments have been widely used, for example, the "Hornyak button" [23]. Caution: Such instruments do not in general read dose, and therefore knowledge of neutron spectrum is required to use them.

More detailed information about fast-neutron hazards may be obtained from instruments that measure the fast-neutron spectrum, and from instruments that measure the LET distribution of the ionization due to neutron recoils. Approximate measurements of neutron energy distributions may be made by activation of fast-neutron threshold detectors [24, 25, 26]. More accurate determination of neutron spectra may be made by measuring the proton recoil distribution in a cloud chamber or photographic emulsion [27], or in a proportional-counter spectrometer [28]. Information about the LET distribution and the dose is obtained from the tissue-equivalent wall, spherical pro-

TABLE 5. *Data on neutron production*

Target	Residual nucleus	Q	Reaction yields	
			Thick target yield or nominal cross section	Corresponding incident particle energy
(p,n) Reactions				
H³	He³	—0.76	9.50	3.0 (peak)
Li⁷	Be⁷	—1.65	0.56	2.2 (peak)
Be⁹	B⁹	—1.85		
(d,n) Reactions				
H²	He⁴	3.26	2.5×10⁴ n/μcoul 0.05 b	0.25 .25
H³	He⁴	17.6	~108 n/μcoul 5 b	.20 .10
Li⁶	Be⁷	3.39		.6
Li⁷	He³+He⁴	1.79	1.7×10⁷ n/μcoul	.2
Li⁷	He⁴+He⁴	15.1	0.05	.2
Be⁹	B¹⁰	4.35	4×10⁴ n/μcoul	
B¹⁰	O¹¹	6.47		
B¹¹	C¹²	13.7		
C¹²	N¹³	0.28		
C¹³	N¹⁴	5.3		
N¹⁴	O¹⁵	5.12		
F¹⁹	Ne²⁰	10.65		
(α,n) Reactions				
Li⁷	B¹⁰	—2.76	5×10⁴ n/sec/c	Po–Li
Be⁹	C¹²	5.7	2.3×10⁶ n/sec/c 1.7×10⁷ n/sec/c	Po–Be Ra–Be
B¹⁰	N¹³	1.07	6×10⁵ n/sec/c	Po–B
B¹¹	N¹⁴	0.15	7×10⁵ n/sec/c	Ra–B
F¹⁹	Na²²	—4.5	2×10⁵ n/sec/c	Po–F
Na²³	Al²⁶	—2.66	2.5×10⁶ n/sec/c 4×10⁴ n/sec/c	RaBeF₂ Po–Na
Mg²⁵	Si²⁸	2.67	2×10⁴ n/sec/c	Po–Mg
Mg²⁶	Si²⁹	0.04		
Al²⁷	P³⁰	—2.92	2×10⁴ n/sec/c	Po–Al

TABLE 5. *Data on neutron production*—Continued

Degree of gamma-ray contamination	Gamma-ray energies	Residual activity or comparative dosage
(p,n) Reactions		
	Mev	
Considerable ~10⁻⁹ γ/proton	20	None.
Considerable ~2×10⁻⁸/proton	17	Considerable (Be⁷).
Considerable 10⁻⁹ γ/proton		None.
(d,n) Reactions		
None	None	None.
Slight		None.
0.3 γ/n	0.4	Moderate (Be⁷)
	.4	None.
.08 γ/n	4.9.	None.
	3.6; 2.3; 1.7; 1.0; 0.7; 0.4.	Small (Be¹⁰).
	6.3; 6.7; 6.4; 4.5.	Small (O¹¹).
		None.
4.45		Slight (N¹³).
	6.1; 5.0; 3.3; 2.3; 1.6; 0.7.	None.
None		Slight (O¹⁵).
	10.6; 9.1; 3.4; 6.4; 5.2; 3.5; 2.3; 1.6; 0.5.	
(α,n) Reactions		
	0.3; 0.49	n/γ≈0.35.
	0.3; 4.45	n/γ≈4.0.
Many	0.3; 3.7; 2.4.	n/γ<0.01.
		n/γ≈5.0.
Many	0.3; 1.3; 1.5	n/γ<0.01.
Many		n/γ≈1.0.
	0.3; 0.4; 1.1; 1.3; 2.6.	n/γ<0.01. n/γ≈0.3.
	0.3; 1.5; 4.0.	n/γ≈0.15.
	0.3; 1.3; 2.3; 3.6.	n/γ≈0.1.

portional counter [29, 30], or by calculation from the measured neutron flux and energy distribution [31].

2. *Intermediate neutrons (0.5 ev to 10 kev).* Intermediate-energy neutron flux is usually not important for health protection because the biological effect per neutron is small compared to that for fast neutrons, and the intermediate neutrons are quickly moderated in the human body to become slow or thermal. The usual instrument for flux measurements is a boron counter in a moderator, such as the "long counter" [32, 33] or certain survey instruments [34]. The long counter is also useful for fast-neutron and thermal-neutron flux measurement.

3. *Thermal neutrons.* Thermal-neutron flux may be determined by absolute measurement of the induced radioactivity in any one of several elements [25, 35]. Boron-walled ionization chambers or counters may be calibrated by means of foil activation.

4. *Gamma rays.* Gamma-ray dose may be measured in carbon-walled chambers which often have only a slight response to neutrons (which must be corrected for). Because the interaction of neutrons with tissue can produce gamma rays, a precise determination of the dose requires appropriate phantoms. Commonly used gamma-ray survey meters and dosimeters cannot be used when large numbers of neutrons are present, as they possess considerable but unknown neutron sensitivity.

5. *Personnel monitoring.* Area monitoring of neutrons around permanent installations is especially important for fast neutrons, because often neither gamma-ray film badges nor pocket dosimeters give sufficient information to evaluate the dose. Fast-neutron nuclear-track film badges [36] are useful in case of accident and, although rather insensitive, are of some value in the permissible dose range. Slow neutrons may be adequately monitored by a cadmium-covered emulsion in a gamma-ray film badge or by boron-lined pocket chambers.

B. Calibration

With most neutron-monitoring instruments it is essential that the instrument be calibrated frequently in a known neutron flux. Extremely large errors are possible with many commercial instruments unless properly calibrated. A convenient method for calibration is the use of a portable radioactive neutron source. A primary standard source of a laboratory may be calibrated for total neutron emission rate by submission to the National Bureau of Standards.

Radioactive neutron sources may also be calibrated for dose rate at a fixed distance using a fast neutron dosimeter calibrated with an internal alpha source [20] or calibrated with monoenergetic neutrons. Known thermal fluxes for calibration may be obtained using the calibrated standard source in a graphite "standard pile" [37], or by use of a moderating geometry to produce a thermal-neutron flux, which may then be compared to a known thermal-neutron flux such as the National Bureau of Standards standard thermal-neutron density [38].

Appendix 5. Neutron-Capture Gamma Rays

Table 6 summarizes gamma-ray energies encountered in neutron capture.

Appendix 6. Shielding Calculations and Data

The same fundamental safety factors that were described [39] for gamma rays apply to neutrons: (a) Distance from source, (b) time of exposure, and (c) attenuation by shielding or other material. Because the several types of neutron sources differ considerably, they will be treated separately.

Reactors*

The neutrons from reactors come from (a) the fission process (prompt neutrons) and (b) radioactive nuclides produced in fission (delayed neutrons).

Prompt Neutrons from Fission

Prompt neutrons from fission are produced at all energies up to about 17 Mev, distributed as shown in figure 16. More accurate figures are given in table 7. The attenuation process consists primarily of collisions in which either large energy degradation is accomplished or the neutron is widely deflected so that its total escape path is significantly increased. Following degradation, many successive collisions take place in a relatively short further excursion before the neutrons are absorbed at low energy. Because in general cross sections decrease with increasing energy,

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*Note: The shielding of reactors is a complicated subject beyond the scope of this Handbook, but the treatment given here will give approximate answers to simple problems and will be adequate for estimating the effects of small changes in shields.

TABLE 6. *Capture gamma rays*[a]

Target	Thermal (n, γ) cross section	Highest energy gamma-ray	Average number of photons per capture
	Barns	Mev	
Aluminum	0.215	7.724	~2
Antimony	5.4	6.80	
Arsenic	4.1	7.30	
Barium	1.17	9.22	
Beryllium	0.009	6.814	2.7
Bismuth	.015	4.17	
Boron-10	3,990	0.478	
Cadmium	8,500	9.046	4.1
Calcium	0.406	7.83	1.8
Carbon-12	.0045	4.95	
Chlorine	32	8.56	2.1
Chromium	2.9	9.716	2
Cobalt	34.8	7.486	
Copper	3.59	7.914	2.6
Fluorine	0.009	6.68	
Gadolinium	36,300	7.78	3.9
Gold	94	6.494	3.5
Hydrogen-1	0.330	2.230	
Indium	190	5.86	3.8
Iron	2.43	10.16	1.7
Lead	0.17	7.38	
Lithium-6	910	0	
Magnesium	0.059	9.216	
Manganese	12.6	7.261	2.6
Mercury	380	5.446	3.3
Molybdenum	2.4	9.15	
Nickel	4.5	8.997	
Niobium	1.1	7.18	2.6
Nitrogen-14	0.1	10.8	
Phosphorous	0.192	7.94	
Platinum	8.1	7.920	
Potassium	1.89	9.28	
Praseodymium	11.2	5.83	
Rhodium	150	6.792	
Samarium	10,600	7.39	5.6
Scandium	22	8.85	
Selenium	11.8	10.488	
Silicon	0.160	10.55	
Silver	60	7.27	2.9
Sodium	0.47	6.41	
Strontium	1.16	9.22	
Sulfur	0.49	8.64	
Tantalum	21.3	6.07	
Thallium	3.3	8.54	
Tin	0.65	9.35	
Titanium	5.8	9.89	
Tungsten	19.2	7.42	
Vanadium-51	4.7	7.305	9.5
Zinc	1.06	9.51	
Zirconium	0.18	8.66	

[a] These data are taken from P. S. Mittelman and R. A. Liedtke, Nucleonics 13, No. 5, 50-51 (1955).
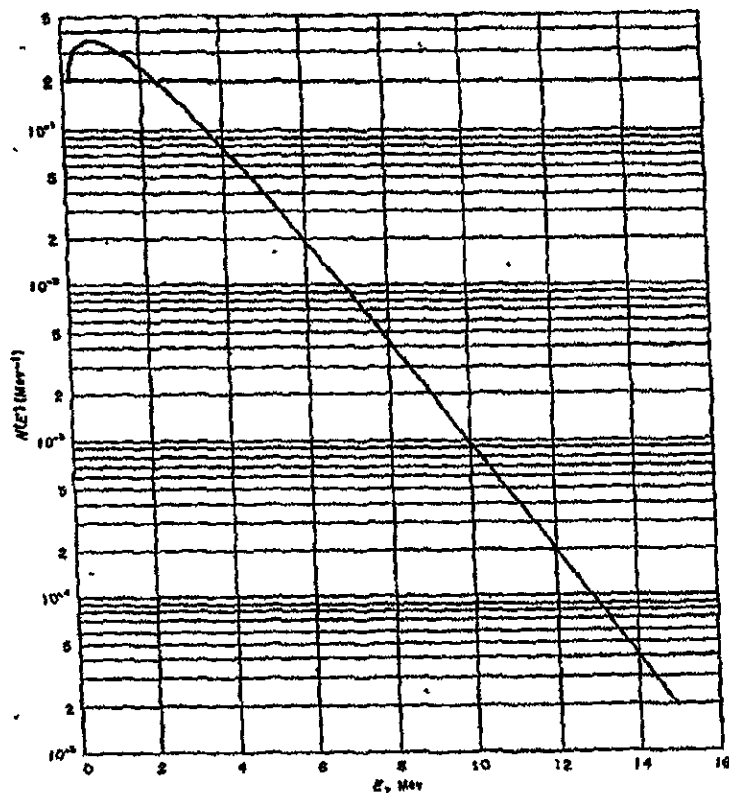
FIGURE 16. *Distribution of prompt fission neutrons.*
Number of neutrons per Mev energy interval versus neutron energy.

the neutrons produced at higher energy have the best chance of penetration. Balancing this is the original distribution in which the lower energies predominate. For most shields the neutrons produced at about 8 Mev are most likely to penetrate, although they may have had a collision near the outer shield surface and hence emerge at lower energy.

A simplified calculation of attenuation is made on the basis of the neutrons at about 8 Mev. Some allowance is necessary for the neutrons that have collided but still escape, a factor usually called the buildup factor. This is taken account of by a compensating factor obtained by

TABLE 7. *The U²³⁵ fission spectrum of prompt neutrons*[a]

$N(E)dE$ is the fraction of neutrons of energy E to $E+dE$ from thermal-neutron fission of U^{235}.

E	$N(E)$	$\int_0^E N(E)dE$	$\int_E^\infty N(E)dE$
Mev	Mev⁻¹		
0.1	0.2025	0.014	0.9860
.2	.2670	.377	.9628
.3	.3068	.663	.9387
.4	.3300	.987	.9018
.5	.3450	.1321	.8679
.6	.3528	.1669	.8331
.7	.3557	.2027	.7973
.8	.3542	.2385	.7615
.9	.3519	.2729	.7271
1.0	.3446	.3082	.6918
1.1	.3368	.3425	.6575
1.2	.3271	.3759	.6241
1.3	.3179	.4974	.5026
1.4	.3073	.4383	.5617
1.5	.2841	.4983	.5017
1.8	.2504	.5631	.4869
2.0	.2371	.6028	.3972
2.6	.1939	.7073	.2927
3.0	.1584	.7876	.2124
3.5	.1026	.8476	.1524
4.0	.07472	.8914	.1086
4.5	.06681	.9192	.0808
5.0	.03847	.9463	.0537
5.5	.02729	.9624	.0376
6.0	.01916	.9739	.0261
6.5	.01386	.9821	.0179
7.0	9.316×10⁻³	.9876	.0124
7.5	6.456	.9916	.0084
8.0	4.428	.9943	.0057
8.5	3.034		
9.0	2.076	.8960	.0040
9.5	1.418	.9973	.0027
10.0	9.58×10⁻⁴	.9982	.0018
11.0	4.384	.9986	1.211×10⁻³
12.0	1.994	.9904	5.501×10⁻⁴
		.9993	2.473
13.0	8.958×10⁻⁴	.9999	1.103
14.0	3.997	1.0	4.875×10⁻³
15.0	1.771	1.0	2.148

[a] Calculated from the formula of B. E. Watt, Phys. Rev. 87, 1037, (1952). See also J. E. Evans, Fast neutron spectra from the water boiler. Los Alamos Scientific Laboratory, LA-1395 (AECD 3073) (March, 1951).

counting all neutrons as if they were in the high-energy group. This simplifies the calculation, is reasonably accurate, and is conservative for shields that contain reasonable quantities of moderating material. Examples of such materials are the water-bearing concretes such as ordinary concrete, barytes concrete, magnesium oxychloride concrete, and water, masonite, moist earths, etc. Other materials
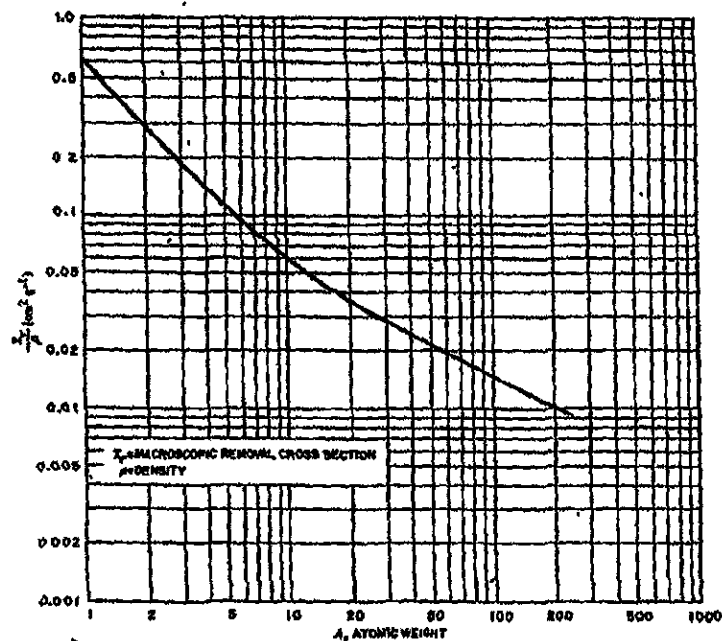
FIGURE 17. *Fission neutron attenuation coefficients.* (Macroscopic removal cross sections divided by density) versus (Atomic weight).

which are not moderators (large atomic weight, $A>16$), such as iron, lead, etc., can be counted as well, provided they are followed with moderating material on the outside.[e]

Removal Cross Sections: Removal cross sections are the cross sections to be used in calculating the attenuation of fast neutrons. The microscopic removal cross section (σ_r) is roughly three-quarters of the total cross section at 8 Mev, an exception being hydrogen, for which the fraction is somewhat higher.

The macroscopic removal cross section is given by

$$\Sigma_r = \frac{0.602\sigma_r\rho}{A} \quad (\text{cm}^{-1}), \quad (6)$$

[e] Caution: These nonmoderating materials are relatively transparent to low-energy (10 to 500 kev) neutrons, which will produce capture gamma radiation when reaching the moderating material; these hard gamma rays must be shielded out subsequently by other material.

where

σ_r=microscopic removal cross section (barns),
ρ=density (g/cm³),
A=atomic weight.

It will be noted that Σ_r/ρ is a quantity dependent only on the microscopic nuclear properties. This is a smoothly varying function of the atomic weight, which is given in figure 17. From this curve the removal cross section can be obtained for any element.

The macroscopic removal cross section for a material of several elements is obtained by simple summation over its constituents:

$$\Sigma_r, \text{compound}=\left(\frac{\Sigma_r}{\rho}\right)_1\rho_1+\left(\frac{\Sigma_r}{\rho}\right)_2\rho_2+\ldots, \qquad (7)$$

where

$\left(\dfrac{\Sigma_r}{\rho}\right)_1$=value from figure 17 for element 1 of compound, (cm²/g), etc.

ρ_1=density of element 1 (g/cm³), etc.

Example: Find Σ_r for CaCO₃; density = 2.711, molecular weight = 100.09.

Element	A	ρ	Σ_r/ρ	$(\Sigma_r\rho)_a \cdot \rho_a$
		g/cm³		
Ca	40.08	$\dfrac{40.08}{100.09}\times 2.711=1.087$	0.0242	0.0263
C	12.01	$\dfrac{12.01}{100.09}\times 2.711=0.325$	.0510	.0166
O	16	$\dfrac{3\times 16}{100.09}\times 2.711=1.299$	.0410	.0533
Total				0.0962

$$\Sigma_r, \text{CaCO}_3=0.0962 \text{ cm}^{-1}$$

Examples of removal cross-section calculations for common shield materials are given below. Table 8 shows the pertinent steps for barytes and ordinary concretes, starting with elemental assays shown in the second columns. In table 9 the results are summarized for several common materials.

The core of the reactor is defined to be the volume in which the fissions take place, and the shield is the material

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TABLE 8. Calculations of neutron removal cross sections (Σ_r)

Element	A, Atomic weight	ρ, Density in composition	Σ_r/ρ	Σ_r
		Barytes block concrete		
		g/cm³		
Fe	55.85	0.307	0.0200	0.0061
H	1.0	.015	.602	.0099
O	16.0	1.090	.041	.0447
Mg	24.32	0.013	.052	.0004
Ca	40.08	.159	.024	.0038
Ba	137.36	1.470	.124	.0182
Na	23.0	0.005	.058	.0002
Si	28.06	.061	.0205	.0012
Al	26.97	.020	.0301	.0006
Mn	54.93	.003	.0202	.0001
S	32.07	.348	.0275	.0096
Total		3.491		0.0945
		Ordinary concrete		
O	16.0	1.105	0.041	0.0452
Si	28.06	0.2315	.0205	.0083
Al	26.97	.0330	.0301	.0010
Fe	55.85	.0132	.0200	.0004
Ca	40.08	.7712	.024	.0185
C	12.01	.0761	.050	.0038
Na	23.00	.0116	.058	.0004
K	39.10	.0079	.0245	.0002
H	1.0	.0250	.602	.0150
Mg	24.32	.0426	.032	.0014
Total		2.2702		0.0942

TABLE 9. Macroscopic neutron removal cross sections (Σ_r) and attenuation lengths of some shielding and reactor materials

Material	Σ_r, Removal cross section	Attenuation length
	cm⁻¹	cm
Water	0.103	9.7
Iron	.1576	6.34
Ordinary concrete	.0942	10.6
Barytes block concrete	.0945	10.6
Graphite (ρ=1.54)	.785	12.7

external to it, including reflector, pressure shell, if any, and of course the shield proper.

The dose external to the shield is now calculated to be

$$D \text{ (rems/hr)}=5.4\times 10^6\frac{pF}{T\Sigma_{r0}\Sigma_r}e^{-x\Sigma_r}, \qquad (8)$$

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p—power density in core (watts/cm³).

Σ_{td}=macroscopic removal cross section of core material, calculated as above (cm⁻¹).

T—shield thickness (cm).

Σ_r—average macroscopic removal cross section for shield (cm⁻¹).

$$=(1/T)[(\Sigma_r x)_i+(\Sigma_r x)_j+\Sigma_r)_k+...].$$

$(\Sigma_r x)_i$=product of macroscopic removal cross section Σ_r and thickness x for the ith shield layer.

$5.4\times10^6=(7.5\times10^{10}$ fission neutrons per joule$)\times(\frac{1}{3}$, geometrical factor$)+(7,000$ neutrons/cm²/sec per rem/hr$)$.

F—geometric factor taking account of the core shape, as follows:

$=1$ for a large flat core surface adjacent to shield.

$=a/(T+a)$ for a spherical core of radius a.

$=[a/(T+a)]^{1/2}$ for a cylinder of radius a, measurement at midplane.

For small cores of whatever shape the formula for a sphere of equal volume may be used, provided the maximum diameter (e.g., corner to corner) d_m fulfills the following criterion:

$$d^2_m < \frac{8T}{\Sigma_r}. \qquad (9)$$

Delayed Neutrons

Delayed neutrons are produced from certain radioactive nuclides among the fission products. The half lives, energies, and yields of the dominant emitters (groups) are given in table 10. They are not important by comparison with the prompt fission neutrons, but for cases in which the fission products are removed from the reactor to regions outside its shield shortly after being produced they may be very important. Examples of such removal are experimental fuel loops and pneumatic sample handlers in which fission takes place in the sample.

For delayed neutrons the only really satisfactory shields are hydrogenous materials, and the effectiveness depends only on the hydrogen thickness of the shield. Because the neutrons are of too low energy for inelastic scattering, ele-

TABLE 10. *Delayed neutrons from fission products*

Group	Half life	λ, Decay constant	Energy	q, Dose per unit flux	v×10⁴ Absolute yields per 10⁴ fissions		Σ_rH, Removal cross section of hydrogen
					U²³⁵	Pu²³⁹	
	sec	sec⁻¹	kev	rems/hr per n/cm²/sec			Barns
1	55.6	0.0125	250	6.1×10⁻⁸	6.25	4.2	7.6
2	22.0	.0315	570	9.7	41.5	31.5	5.2
3	4.51	.154	412	8.1	53.2	37.3	6.0
4	1.52	.455	670	10.5	60.3		4.7
5	0.43	1.61	400	8.0	21.2	35.7	6.1
Total					182.35	109.2	

ments other than hydrogen are relatively ineffective and are ignored. For convenience the removal cross section of hydrogen for each energy is also given in table 10.

The dose due to the delayed neutrons is just the sum of the contributions from several emitters:

$$D_{total}=D_1+D_2+D_3+D_4+..., \qquad (10)$$

where 1, 2, 3, etc., refer to the different groups of delayed neutrons. The contribution from one emitter is calculated as follows:

$$D_1 \text{ (rems/hr)} = \frac{S_1 q_1 e^{-\Sigma_{rH}T}}{4\pi R^2}, \qquad (11)$$

where

R=distance from center of delayed neutron source to place where D_1 is to be determined.

T—shield thickness (cm).

$\bar{\Sigma}_{rH}$=average macroscopic removal cross section (cm⁻¹)

$$=1/T[(\Sigma_{rH}x)_i+(\Sigma_{rH}x)_j+....].$$

$(\Sigma_{rH}x)_i$=product of hydrogen removal cross section and thickness for the ith shield layer.

q_1=dose, rems/hr, per unit neutron flux (neutrons/cm²/sec) for the delayed neutrons of group 1, table 9.

S_1=source strength, neutrons/sec, of neutrons from group 1.

For the common case of a uranium-bearing liquid (or wire) loop circulating into and out of a reactor, the source,

S_1, is just the rate of neutron release outside the reactor shield (but of course inside the special loop shield). Then, after many cycles, for each group there is an expression of the following form:

$$S_1 = \frac{fy_1 \; (1-e^{-\lambda_1 t_i}) \; (e^{-\lambda_1 t_2}) \; (1-e^{-\lambda_1 t_i})}{\lambda_1 t_1 (1-e^{-\lambda_1 T})}, \qquad (12)$$

where

f—fission rate, fission/sec (total).

y_1—yield, neutrons produced in group 1 per fission.

λ_1—decay constant, sec^{-1}, for group 1.

t_i—time each loop particle spends in the neutron field per cycle.

t_s—time each loop particle spends on the way out through the shield.

t_r—time each loop particle spends in the exterior part of the loop, from which it radiates.

T—cycle time.

Radioactive Neutron Sources

The dose rate from a radioactive neutron source is best calculated somewhat differently from that from other sources. Owing to the small intensities available, relatively thin shields are needed and hence the scattered neutrons offer a very significant contribution to the total dose. To take account of this, a buildup factor is used. For Po-Be and Po-B sources with water or paraffin shields at least 20 cm thick, the value of this buildup factor is approximately 5. For these common situations, the following formula gives the dose rate:

$$D \; (\text{rems/hr}) = \frac{SBqe^{-\Sigma_r T}}{4\pi R^2}, \qquad (13)$$

where

S—source strength (neutrons/sec).

R—distance from source to point for which the dose rate is to be calculated.

B—buildup factor, ~5.

q—dose per unit flux for neutrons of the source energy (rems/hr per neutron/cm^2/sec).

Σ_r—macroscopic removal cross section for the given source and shield, obtained from table 8 where possible, or calculated using nine-tenths of the

hydrogen cross section for shields and sources not included thereon.

T—shield thickness (cm).

Accelerators

Accelerators which yield neutrons can be shielded according to formulas which have been developed for reactors. For neutron energies up to 30 Mev the removal cross sections can be taken to be about three-quarters of the total cross section.

Appendix 7. Neutron Protection Near High-Energy Electron Accelerators

Because there are many types of high-energy electron accelerators and because installations vary widely, it is not practicable to give general rules for neutron protection which would be valid for all cases. However, two chief factors may be pointed out that govern the neutron shielding problem. These factors are (1) the energy of the photons and (2) the type of shielding material used. The primary sources of neutrons near a high-energy electron accelerator are the machine itself and the point where the beam strikes the wall [40].

Neutron production. To liberate a neutron from the nucleus requires that the absorbed photon energy exceed the binding energy of the particle in the nucleus (of the order of 6 to 18 Mev, mostly about 10 Mev). Therefore electron accelerators of energy below 10 Mev will usually have very small or no neutron backgrounds. Common photoneutron threshold energies and information on neutron yields for various elements are given in Handbook 55 of this series.

For electron accelerators of energies much larger than photoneutron threshold energies, the yield of neutrons by photons incident on lead may be approximated by [41, 42, 43] $Y_{\text{Pb}} \simeq 0.4 W_0$ where Y— the number of neutrons produced and W_0 is the total incident photon energy (i.e., number of photons times photon energies) in Bev. The yield Y varies approximately as the square root of the atomic number. The above relation may therefore be rewritten $Y_z \simeq 0.044\sqrt{Z} W_0$. The photons have energies of the order of 20 Mev. The neutrons are produced with a wide distribution in energy, the average for lead being about 2 to 3 Mev, and for carbon about 5 Mev [40]. As photon

energy increases, the neutron protection problem becomes more serious, presumably because of a high-energy tail to the neutron distribution. As the mean free path for fast neutrons becomes larger (approximately as the energy) as the neutron energy increases, the choice of shielding may, for very high energy electron accelerators, actually be dictated by the fast-neutron hazard.

Shielding materials. Concrete or other hydrogen-containing materials are desirable for neutron shielding. However, many electron accelerators are shielded with lead, frequently as an integral part of the machine itself. The lead acts both as a strong source of photoneutrons and as a poor neutron shield.

After the neutron production near the high-energy electron accelerator has been established (remembering that neutrons may be produced in the shield itself), the neutron shielding requirements may be determined according to the methods outlined in appendix 6.

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WASHINGTON, March 29, 1957.

Scientists Find Device to Protect Radium Workers From Death-

Remote Laboratory Is Used for Research Activities With Valuable Metal.

By James N. Miller.

IN A remote Washington laboratory the other day two scientists, a man and a woman, unwrapped an enormous crate. Armed with hammers, they worked with feverish haste, so that within half an hour their job was done.

Whereupon the gentleman turned the lady remai : "Nice ing, Miss Torrey, Looks as if we ad something worth while here."

And so they had. The crate, which they now examined with unconcealed pride, contained their own "brain child," a truly marvelous new device, the only one of its kind in the world, which enables research workers to test radium, the most dangerous metal known, without the slightest danger of their being unduly exposed to the death-dealing gamma rays.

The two scientists, Dr. L. F. Curtiss and Constance L. Torrey of the United States Bureau of Standards, after a year's research, developed the idea for the revolutionary machine toward the close of 1935 and detailed a New York manufacturer to build the actual apparatus.

Most of the commercial radium preparations now being tested for "purity," with the aid of the new device, and others also developed within recent months, are the so-called "radium needles" and tubes, widely used in therapeutic work by the medical profession. They may contain as little as one, or as high as 100, milligrams of radium, and often are only an eigth of an inch long and one-thirty-second of an inch diameter.

RFQUE medical agency will buy a d r so of these tiny "needles." They may have about the sams size and appearance, and it is necessary to have an identifying number cut on each individual needle, so they can be told apart. In such case the medical agency will send the needles to the Bureau of Standards, whereupon this procedure will be followed:
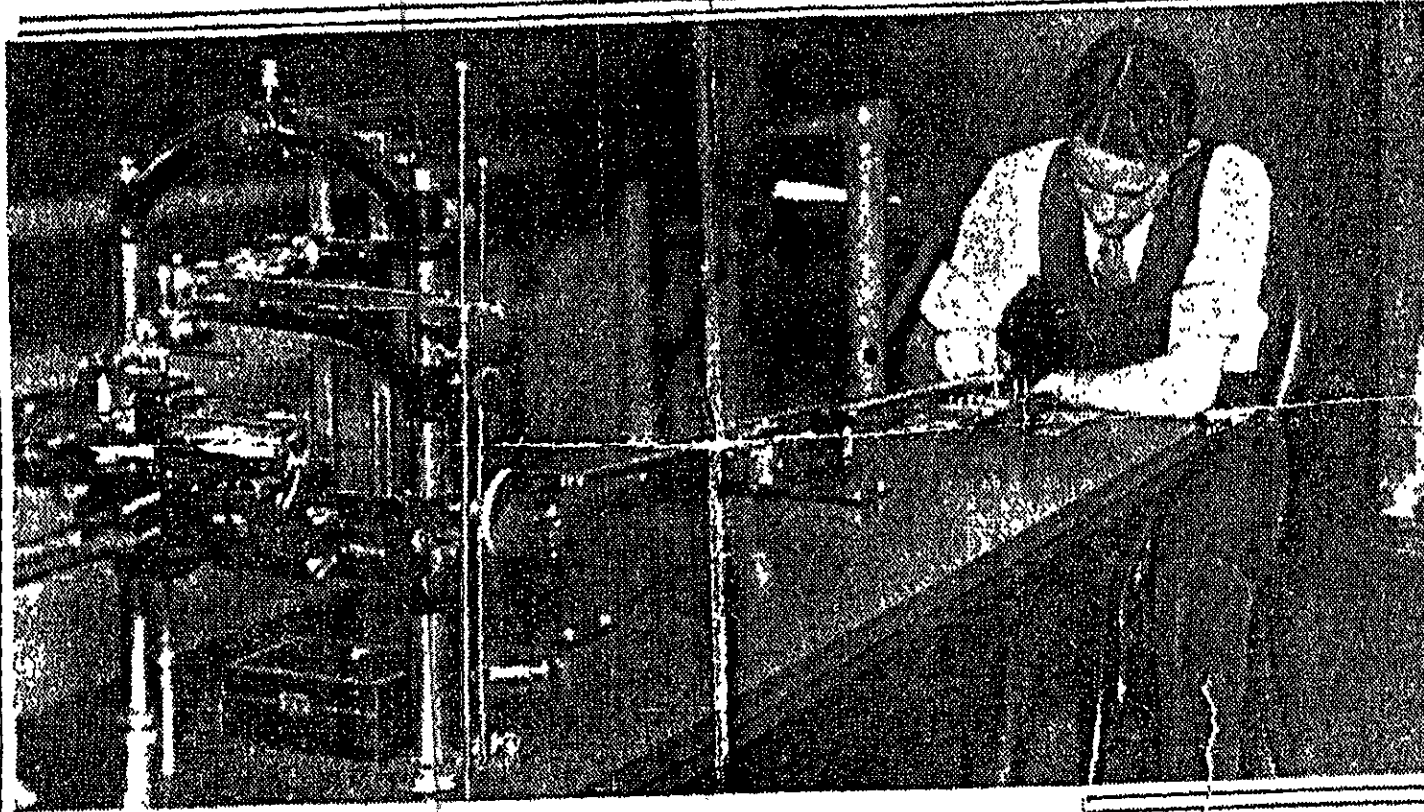
5 feet away from the "needle," manipulates a "long-distance" stylus, which carves the number. If the stylus moves a half inch, the number will be only one-one-hundredth of an inch in height.

The bureau scientists have also perfected a projection machine for reading these numbers from a safe distance of 5 feet away from the test specimen. The average therapeutic worker can use a magnifying glass for this purpose without personal danger, because he might have occasion to identify the numbers on his radium needles only two or three times a year, whereas many times in a single day the Federal experts must do so.

Since always, by means of these protective devices and numerous others, the bureau scientists are able to maintain a distance of 10 inches to 5 feet away from the dangerous element they are studying, there is very little

Engraving microscopic numbers on a radium preparation for identification by medical agencies. The operator (B. W. Brown) sitting in perfect safety, 5 feet from the radium, manipulates a stylus, the only machine of its kind in the world, which has just been developed by Bureau of Standards scientists after years of research.

cally the scientists are given a thorough physical examination.

SHALL we now make an imaginary visit to the bureau's radium testing laboratories and examine for ourselves some of the most interesting of the new safety gadgets? All right, let's go.

The laboratories are on the third floor of the east building in the heart of the exclusive residence section of upper Connecticut avenue. You enter room 317, a small office, and are privileged to meet the internationally

has handled about 10 times as much radium as did the late Mme. Curie, who discovered the precious element.

First of all, Dr. Curtiss explains certain elementary but highly interesting facts about the metal, which is about 50,000 times worth its weight in gold and sells on the commercial market for $50,000 a gram.

"How big is a gram? About the size of a tiny tube less than an eighth of an inch square.

Radium is a fiercely active substance, which emits three kinds of rays, alpha rays, beta rays and gamma

which carry a positive charge. They are expelled from the radium atoms at speeds of the order of 10,000 miles a second.

"Beta rays are negative electrons, expelled at much higher speeds closely approaching the velocity of light—186,000 miles a second. Because they are lighter than alpha particles, they can penetrate material through which alpha rays cannot pass.

"Gamma rays are similar in nature to ordinary light, but have a wave length very much shorter and consequently are more penetrating. As a striking example from our own test program, one or two milligrams of fairly pure radium, inclosed in a leaden tube, with walls 1/4-inch thick, are able to discharge rapidly a gold leaf electroscope held anywhere near it.

"Radium preparations are also self-luminous and possess an extraordinary power of bringing about chemical

Dealing Rays

Ingenious Method Employed to Keep Element Beyond Possible Danger.

extremely marked physiological effects, destroying tissues when the radiations act upon them too long or too intensively. Fortunately for medical science, the radiations also destroy disease cells of certain types as well as normal tissue. This is notably the case in cancers, other malignan' tumors and in certain non-malig tumors."

WHAT is purity in radium, and how does it concern the bureau scientists? Bear in mind, first of all (Dr. Curtiss explains), that all commercial radium is in the form of a radium salt and a radium salt, in turn, is in the form of a radium bromide, chloride or sulphate. Strictly speaking, there is no such thing on the market as "pure radium." However, most of the radium salt sold in the United States is accompanied by a certificate, issued by the Bureau of Standards, stating the precise amount of radiation it gives off. Oddly enough, the certificate never states whether or not there actually is any radium present!

Practically all the radium used in this country and elsewhere in the world is prepared by the Radium Belge Co. in Brussels. There radium is purified in the final stages by a process known as "fractional distillation," which removes all foreign matter other than the bromide or choride occurring with the radium in nature.

The official judgment of the bureau scientists, after almost countless tests of sample radium in their laboratories, is this: Ordinarily, if a radium salt is 99 per cent radium bromide or chloride, it is considered of a fair degree of purity. It is a safe generalization. Dr. Curtiss says, that practically all the so-called radium on the American commercial market is around 99 per cent radium salt. Bear in mind, however, that the radioactive properties of a piece of radium, that is to say, the strength of its gamma rays, are entirely independent of the particular chemical combination of radium with other elements.

Strictly speaking, then, what the Bureau of Standards experts are called upon to do daily, as their most im-

Radium

portant job, is not to determine the "purity" of a sample piece of radium salt, but to discover exactly what its gamma ray strength is. This is measured with the new "gamma ray electroscope."

First, the test radium is placed on a piece of gold leaf inside the device and then the scientist, peering in perfect safety at the sample from a distance of about 10 feet, snaps a switch that turns on an electric current which charges the gold leaf, and as the radiation, produced by the gamma rays of the radium, goes through the instrument in which the gold leaf is hung, it makes the air a conductor and slowly discharges the electricity on the gold leaf.

Since the leaf was held in an upright position by the electric charge, it will begin to fall as the charge leaks off. The rate of fall is proportionate to the strength of the gamma radiation. Thus, as we hinted earlier, is the bureau's major test for radium "purity."

ANOTHER novel device, used for a somewhat similar purpose, is the "emanation electroscope." Actually it is so sensitive that it enables the scientist to measure one-thousand-millionth of a gram of radium gas!

This apparatus is occasionally used to do special jobs for medical agencies. For instance, to measure the exact amount of "radon," or radium. Their ... is so microscopic that it

is entirely invisible, ... naked eye, but under the ordinary microscope.

When using the emanation electroscope, first the specialist forces a tiny amount of the radium gas into the device by means of a vacuum pump. Whereupon the observer, armed with a stop-watch, times the motion of a piece of gold leaf inside the apparatus in similar fashion as he does when measuring a piece of radium salt in the bureau's gamma-ray electroscope.

HAVING shown you all these devices, Dr. Curtiss now takes you to the basement of the east building for a look at America's queerest safe. Never does it house money, gold or precious documents, but instead is used exclusively to hold a quantity of radium barely the size of a tiny cube less than three-eighths of an inch square.

Yet the door alone weighs 2,700 pounds, whereas the rest of the safe is heavy as lead, in fact, is lead-lined 6 inches throughout its structure. Why? Because lead shields the research worker from the deadly radiations of gamma rays.

Dr. Curtiss opens up the 10-ton radium safe. He says it is the only one of the kind in the entire world and is another "brain child" of the Bureau of Standards technicians. If necessary it would house a tenth of America's entire radium supply, which at a rough guess totals about 50 grams and is worth around $22,000,000.

You note with keenest interest th compartments used for storage of th test radium. These are ingeniousl arranged so that the scientist is ex posed to the rays from one compart ment at a time only, and even the well within the safety range.

In front of each slide is a lead bloc 2 inches thick. When all the slide are closed the scientist has 2 inchi of protective lead between himse and all the radium in the safe, eve after he opens the outside door.

The radium itself is kept in sms wooden cases lined throughout wit half an inch of lead. The wood used because it protects the han against the dangerous secondary "beta" rays.

This safe, by the way, is not the or one used by the specialists. It is t largest of three, and if necessary cou house 50 grams of radium. Howev the maximum amount ever kept i side was 7 grams.

The smallest safe is in the cent room, or main office, of the bure technicians. It is about 3 feet by ? feet by 2. Ordinarily it contains abc 200 milligrams, or two-tenths of gram. This safe is not lead lin but contains lead plates for screeni off the dangerous rays.

The third, or medium-sized safe, kept in an adjoining office. It lined throughout with an inch lead and at present contains no dium.

WHEN transporting a sample radium from one room to other the Federal research wor simply takes it out of the safe v a pair of giant tweezers so built t always the hands are kept 10 inc away from the dangerous elem Next he places it inside a queer-k ing wooden container which at glance resembles an average violin. The lid is easily snapped and the radium can be carried al in perfect safety until ready for t ing.

The length and diameter of a c mercial radium preparation are m

WHEN transporting a sample of radium from one room to another the Federal research worker simply takes it out of the safe with a pair of giant tweezers so built that always the hands are kept 10 inches away from the dangerous element. Next he places it inside a queer-looking wooden container which at first glance resembles an average size violin. The lid is easily snapped on and the radium can be carried about in perfect safety until ready for testing.

The length and diameter of a commercial radium preparation are measured as follows: First the bureau scientist, with the aid of his "safety tweezers," drops the test metal into a tiny, notch-shaped trough. Whereupon he goes into the next room, on the opposite side of the wall, and

THE SUNDAY STAR, WASHINGTON, D. C., MARCH

kes his reading by peering at the cimen through an enormous "protor." As the name suggests, it jects an enlarged image on a graded screen and in such fashion t the length and diameter can be i immediately. For always the aratus is kept in perfect focus. ever do the Federal specialists e to purchase special glass vials ther containers in which to keep r test radium and for this rea- They keep most of it in the ainer in which it arrives and in y instance the container has ed satisfactory in ability to resist powerful gamma radiations. Most e original containers, by the way, not made of glass, as seems to e general notion. Instead, they're latinum, chiefly because it's a

trunks of trees, which in some remote geological age soaked up the original water containing the radium—maybe millions of years ago. These tree trunks petrified, that is, turned into rock, thereby permanently inclosing their radium content.

Many beautiful rock samples of these ancient tree trunks are to be found in Dr. Curtiss' office. The Bureau of Standards is particularly proud of owning several such, brownish red in shade, which strikingly show the grain of the original wood in a beautiful, petrified form.

Dr. Curtiss, for purposes of comparison, showed samples of carnotite and pitchblende. The latter is a lovely jet black, which no doubt accounts for its name.

"Lift them both," he urged, "and I'll show you something interesting."

native Indians are employed along with other workmen. One day not long ago a certain redskin walked up to his boss foreman and announced he was certain there was another large pitchblende deposit on the shore of the lake about a mile and a half from the main quarry then being worked.

"What makes you so sure?" asked the boss.

"Because," said the Indian, "it smells the same over there as it does here."

Naturally the boss didn't take this very seriously. However, a couple of days later he casually walked over the ice to the place described by the redskin. scraped off a layer of snow, and, to his surprise and delight, found that the nose test was correct. Sure enough, there was an additional deposit of

in 1921 those 35 grams were about $4,000,000.

A possible reason for the radium "boom" might have bee the working of the carnotite p in Colorado had hit its peak Moreover, there was a wider market for the metal, for up to time there had been very little refined in the entire world. T stopped within three or fou because it was found that could never really rank seri competition with pitchblend source of radium.

Recently the bureau exper what is believed to be the lar mercial radium preparation e This was a half gram of r

THE WAS

Hunting Priceless Radium


Underwood & Underwood

No, it is not a new vacuum cleaner, but a recently developed radium detector at the Bureau of Standards. The device already has proved its efficiency in locating the tiny hollow needles used in applying radium. The needles frequently become lost and heretofore a long and tedious search has been necessary to find them.

Burrell Brown
age 24

Giant Balloon Tests Weather Device
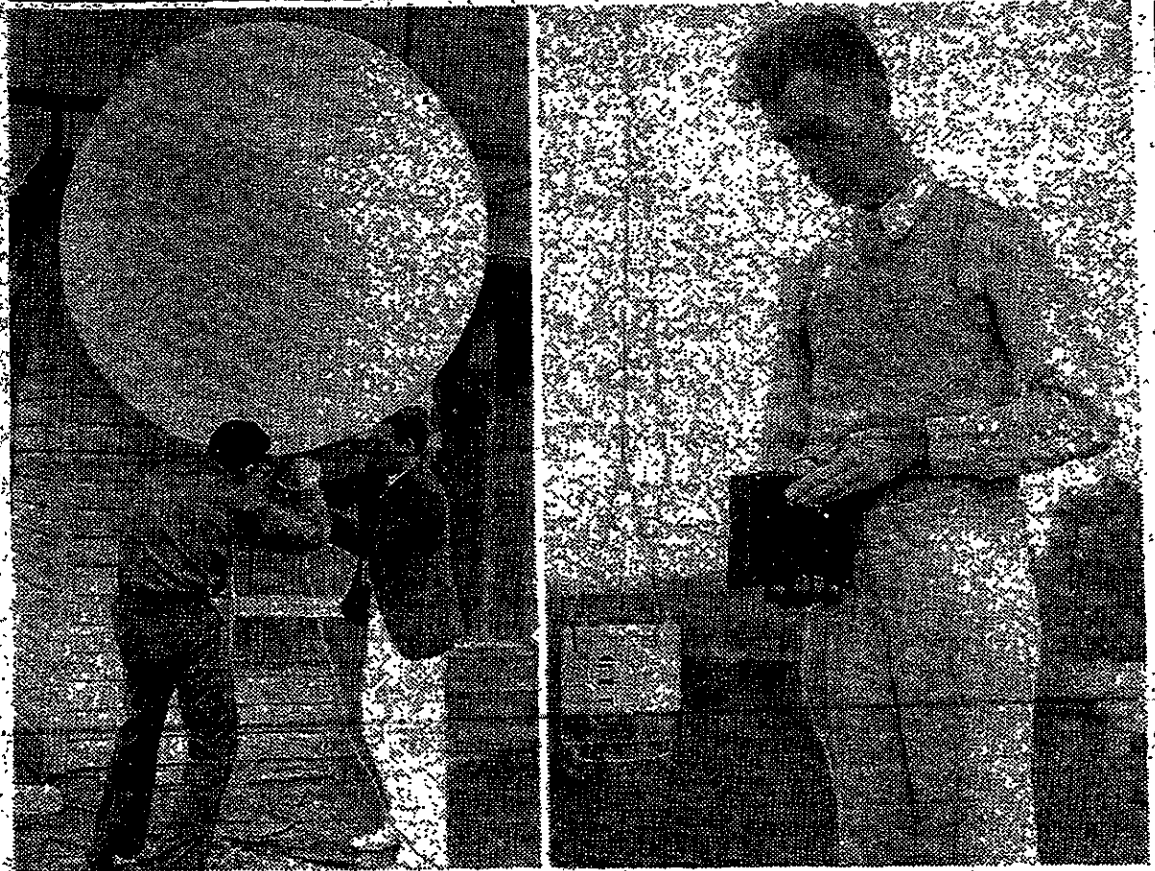
Two Bureau of Standards scientists are making ready here for the test of another radio-meteorograph, a device attached to parachute and balloon and sent into the skies to radio back signals giving the pressure, temperature and humidity of the upper air. Dr. L. V. Astin of the bureau's electrical division (left) and Dr. L. F. Curtiss, head of the radium division, are tying the mouth of a giant balloon, preparatory to beginning the test. The balloon carries the instrument as high as 50,000 or 60,000 feet—9 to 11 miles high. The test was conducted for the Weather Bureau.

L. L. Stockmann, who works with Dr. Curtiss, is testing the radiometeorograph here with a wave meter before attaching it to the parachute and the balloon. The inflated bag usually breaks several hours after it soars away from the roof of the West Building on the Bureau of Standards grounds. The instrument is then parachuted to the ground. The bureau often gets them back. The radiometeorograph contains two radio tubes, a miniature dry cell battery and devices which record and send the weather signals which, in turn, are recorded on a chronograph in Dr. Astin's roof radio room.
—Star Staff Photos.

UNCLE SAM'S RADIUM GIRL

Miss Constance Torrey of the U.S. Bureau of Standards, has one of the most unusual jobs in the world—measuring and testing radium. She is shown here holding about $2,000 worth of this precious element in the little platinum tube at the end of the tweezers. In the small trough below the tweezers, more than a million dollars worth of radium has been placed at one time or another. The pieces range in size from less than one milligram to more than 200 milligrams. About 20 milligrams are in the platinum tube. Virtually all radium is sent to the Bureau of Standards for measuring before being used.

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This woman, Miss Torry is also mentioned in the article, "Scientists Find Device to Protect Radium Workers from Death." She lost the tips of her fingers from handling radium. Miss Torry was one of my dad's co-workers in the Radioactivity Lab at the National Bureau of Standards.

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UNITED STATES GOVERNMENT

Memorandum

U.S. DEPARTMENT OF COMMERCE
NATIONAL BUREAU OF STANDARDS

TO   :   Mr. Robert S. Walleigh
           Associated Director Administration
           Through: Dr. Wayne W. Meinke
                      Chairman, Radiation Safety Committee

DATE: March 22, 1968

In reply refer to: 500.02

FROM   :   Abraham Schwebel
           Chief, Health Physics Section

SUBJECT: Release of buildings at Van Ness Street after decontamination

From the early 1920's till 1952, the Radioactivity Laboratory was located in the East Building (Building #2). All radium samples used in this country for medical purposes during this period were measured in this building. Since the technology of sealing radioactive sources was then in the process of development, there were numerous leaking sources received for analysis. As a result, a number of rooms in the East Building became contaminated with radium-226. When the radium laboratory was moved to its new quarters in a specially constructed laboratory in 1952, all the wood benches, floors and other equipment in these rooms were removed as radioactive waste and new floors and baseboards were installed in a number of rooms. Since 1952 these rooms were used as non-radioactive laboratories and offices. At this time, it should be pointed out, there were no health physicists at the Bureau and this work of decontamination was supervised by the physicists in charge of the radium laboratory.

When the move was made to Gaithersburg, it was recalled that the attic in the East Building had been used as a laboratory and for storage areas. Room 507 had been cleaned, painted and new floors put down sometime after 1952. The walls, though they showed no removable contamination, were found to be radioactive in spots to the extent of ≈1 mr/hr. There were a number of spots on the floor which gave γ-ray readings of <1 mr/hr. The doors to this room were sealed and "Caution – Radioactive Material" (CRM) signs placed on the door. In rooms 518 and 519 numerous samples of radioluminescent paint were found as was a large amount of contaminated equipment which had been stored there years ago. All this material was removed as radioactive waste. A number of dry smears were made throughout the attic but no removable contamination was found. No attempt was made to decontaminate rooms 518, 519 or 520 but the doors were sealed with CRM tape and CRM signs were posted.

Since none of the areas in the East Building were controlled areas and since they had been used for other purposes since 1952, no further surveys were made in the East Building. The buildings were subsequently turned over to GSA who in turn leased them to the District of Columbia. At this point a D.C. radiological monitoring team found the CRM signs and made a survey. They found that the attic had a large number of spots which indicated α contamination to the extent of 100,000-1,000,000 counts/minute. These were fixed and could not be removed by dry smears. They also found a few small areas in the hallways that contained over 100,000 α counts/minute. The question of who was to clean this mess arose. Mr. Morton Kelly very kindly supplied three laborers for a period of two weeks and a clean up was begun.

The first order of business was to determine the ^{222}Rn content of the air. If there were excessive amounts of ^{226}Ra in the buildings then there should be an elevation of the ^{222}Rn and its daughter products. Air samples were taken outside the building and in the various rooms found contaminated. The ^{222}Rn and daughter products were found to be approximately 2×10^{-10} μCi/ml of air in the outside air and from $2 - 8 \times 10^{-10}$ μCi/ml inside the building. The International Commission on Radiological Protection has recommended a maximum permissible concentration for ^{222}Rn and its daughter products of 3×10^{-9} μCi/ml for the population at large. The ICRP recommends a level of no more than 1×10^{-8} μCi/ml of air for people who live in the environs of a plant which gives rise to these radioactive materials, and a level of 3×10^{-8} μCi/ml of air for workers in a plant where such materials are generated. On the basis of the ^{222}Rn the building air was certainly safe from + radiological contamination. In no case was any long lived α activity found in the air samples which were taken by means of Staplex air samplers and analyzed according to the method of Dr. John Harley of the Health and Safety Lab, New York Operations Office of the Atomic Energy Commission.

All floors were scrubbed with liquid detergents using wire brushes. Where necessary, paint remover was used. The levels were brought down to below 5,000 α counts/minute. At this point the floors were painted with a lead pigment paint. The lead pigment was used to dilute the isotopic ^{210}Pb. After painting, no α contamination could be detected except in one or two small areas. These areas were repainted. All the walls were vacuum cleaned and all contaminated equipment was removed as radioactive waste.

It was then decided to check all the rooms in which radium had at one time been handled. Hot spots reading as high as 20 mr/hr gamma radiation were found in rooms 418, 316A, 316 and 317. Further surveys revealed local contamination in other rooms, not used for radium work, as well as extensive contamination of the fourth floor hallway and, even more surprising, extensive contamination of the hallway in front of the fourth floor lecture room. No contamination was found in the lecture room itself. Investigation revealed that one of the physicists measuring radium had periodically vented the radium ampoules through the window of room 418. The windows next to the lecture hall were usually kept open for ventilation in the spring and summer, so that the air exhausted out one window swept back into the hallway through others.

To remove the contamination in room 418, the floor, floor moldings, exhaust dust cover, windows, doorway frames and doors had to be removed. The air duct was vacuumed and painted. On the third floor the floors in rooms 316A, 316 and 317 and a partition between 316A and 316 had to be removed. A 3-inch thick concrete floor had to be removed from all three rooms since it was highly contaminated. Fortunately, the permanent concrete sub-floor was not contaminated so that no structural demolition was necessary. The floor moldings, some of the window framing and doors had to be removed.

Since the hallway floors were contaminated, an outside contractor was hired to drill up parts of the floor with a jack hammer. Some of the floor on the fourth floor, especially the one in front of the lecture hall, had to be removed. Approximately 3 inches of concrete were removed. In the attic 3 sections of floor were removed with a jack hammer and replaced with Sakrete mix.

After all the rough work had been done, and all rooms and hallways vacuumed with special absolute filtered vacuum cleaners, a 10-man janitorial force came in to wash down all the floors and hallways in the building.

After the clean up, a recheck was made of the ^{222}Rn and daughter products. The radon content in air outside the building was 8.3×10^{-11} µCi/ml of air while it varied from $1.5 - 4 \times 10^{-10}$ µCi/ml of air in the building. This is well below the maximum permissible concentration in an unrestricted environment. In no case was any long lived radioactive contamination found. All smears were down to background after 72 hours indicating that all the α activity was due to radon and thoron and their daughter products. Approximately 200 dry smears were

taken over 100 cm^2 areas. All smears read less than 10 α counts/
minute which is the level permitted at NBS.

Tables and charts summarizing our findings are attached.

To the best of our knowledge, all the buildings in the downtown complex
are now free of radioactive contamination. This is not to state that
there may not be a small spot here and there that has escaped our
detection, but we can unequivocally state that there is absolutely no
radiation hazard of any sort.

Four weeks were spent in decontaminating the building. This amounted
to approximately 1000 man hours, 50% contributed by the Health Physics
Section, 25% by Plant Division courtesy of Mr. Morton Kelly, and 25%
by GSA. Approximately 100 55-gallon drums were filled with radioactive
waste. Approximately $1500 was spent for supplies such as drums which
cost $750. In addition, a large number of respirators, coveralls,
shoes, paint, towels, etc., were consumed. The Health Physics Section
has absorbed the cost of the decontamination as far as supplies and
equipment are concerned.

Attachments

Air Samples for ^{222}Rn

	Date Taken	Before Clean-Up	During Clean-Up	After Clean-Up
1. Outdoor - East Bldg.	2-23-68	1.2×10^{-10}		
	2-28-68 AM	1.0×10^{-10}		
	2-28-68 PM	1.3×10^{-10}		
	3-19-68	8.3×10^{-11}		8.3×10^{-11}
2. Room 518 and 519	2-16-68	2.0×10^{-10}		
	2-27-68			1.5×10^{-10}
3. Room 522	2-16-68	2.3×10^{-10}		
	2-27-68			
	3-19-68			1.5×10^{-10}
				1.6×10^{-10}
4. Room 507	2-16-68	3.5×10^{-10}		
	2-19-68	3.2×10^{-10}		
	2-23-68			
	2-23-68			2.2×10^{-10}
	2-23-68			1.5×10^{-10}
	2-27-68		4.3×10^{-10}	
	3-19-68	Room sealed overnight - no ventilation		6.2×10^{-10}
				4.1×10^{-10}
5. Room 418	2-23-68	3.8×10^{-10}		
	2-23-68		7.4×10^{-10}	
	2-27-68			
	3-19-68			1.6×10^{-10}
				1.5×10^{-10}
6. Room 403	2-28-68	2.8×10^{-10}		
	3-19-68			2.4×10^{-10}
7. Room 316A	2-27-68	3.6×10^{-10}		
	3-19-68			
8. Room 316	2-27-68	3.3×10^{-10}		2.5×10^{-10}
	3-19-68			2.5×10^{-10}
9. Room 317	2-28-68	3.6×10^{-10}		

1. No contamination >2500 c/m α was detected in Rooms 500, 501, 505, 506, 508, 509, 510, 512, 516, 517, and the Filter Room.

2. Room 502

   ≤30,000 c/m on wood shelving
   ≤ 2,000 c/m ledge over door
   ≤ 5,000 c/m concrete floor

3. Room 503

   ≤12,000 c/m wood shelving
   ≤ 1,500 c/m ledge over door
   ≤ 1,800 c/m wood frame for room partition
   ≤ 2,500 c/m floor

4. Room 504

   ≤ 4,000 c/m spots on wood shelving
      1,500 c/m floor

5. Room 507

   ≤ 20,000 c/m steel beams
   ≤250,000 c/m concrete ceiling by skylight
      5,000-150,000 c/m brick section of left wall
   >500,000 c/m spots on lower section left brick wall
         5-20 mr/hr contact spots on concrete ceiling
   ≤   40 mr/hr contact 4 spots on floor in room

6. Room 518

   ≤ 36,000 c/m wood shelving
   ≤   500 c/m walls
   ≤ 1,500 c/m ledge over door
   ≤ 7,500 c/m floor

7. Room 519

   ≤ 22,000 c/m wood shelving
   ≤   500 c/m walls
   ≤ 3,000 c/m floor

8. Room 520

   ≤ 48,000 c/m wood shelving
   ≤ 28,000 c/m ledge over door
   ≤ 8,000 c/m floor
   ≤ 7,500 c/m wood frame between rooms

9. Room 521

    ≤ 3,500 c/m spots on wood shelving
    ≤ 12,000 c/m floor

10. Room 522

    ≤120,000 c/m inside ventilation duct
    ≤200,000 c/m floor around air duct
    ≤ 40,000 c/m other floor area
    ≤ 35,000 c/m ledge over door
    ≤ 60,000 c/m section wall behind air duct
    ≤ 75,000 c/m wood shelving
    ≤ 12,000 wood frame for room partition

11. Steps leading to roof

    ≤ 70,000 c/m

12. Roof

    ≤ 10,000 c/m gravel on roof near skylight
    ≤350,000 c/m fan vent
    ≤150,000 c/m skylight
    >500,000 c/m fan
    ≤   5 mr/hr contact fan vent and skylight
    ≤   15 mr/hr contact fan

Before Decontamination

Fourth Floor

1. Room 403

   $\leq$ 20,000 c/m air duct
   $\leq$ 30,000 c/m and 1.0 mr/hr contact windows and frames
   $\leq$ 5,000 c/m and 1.5 mr/hr contact wall molding

2. Room 418

   $\leq$ 60,000 c/m and 10 mr/hr contact windows and frames
   .2-6 mr/hr contact on wall molding
   $\leq$ 150,000 c/m air duct

3. Room 410

   $\leq$ 10,000 c/m to left door 6 ft^2 area
   2 mr/hr contact spot floor by door to Room 411

4. Room 411

   0.8 mr/hr contact floor area by room entrance $\approx$ 2 ft^2

5. Auditorium Lobby Area

   $\leq$ 30,000 c/m and 0.5-8 mr/hr windows and frames
   $\leq$ 12,000 c/m marble room molding
   $\leq$ 15,000 c/m rug
   $\leq$ 60,000 c/m major portion of terrazzo floor

Before Decontamination

Third Floor

1. Room 315A

   $\leq$ 30,000 c/m baseboard
   $\leq$ 20,000 c/m wood floor

2. Room 315

   $\leq$ 15,000 c/m eleven spots on floor

3. Room 316.1

   $\leq$ 75,000 c/m and $\leq$20 mr/hr contact floor
   $\leq$ 20,000 c/m and $\leq$20 mr/hr window and frame
   $\leq$ 75,000 c/m baseboard
   $\leq$ 5 mr/hr contact all wallboards and door frame

4. Room 316

   $\leq$ 8,000 c/m and $\leq$10 mr/hr window and frame
   $\leq$ 5 mr/hr contact all wallboards and door frame
   $\leq$ 80,000 c/m baseboard
   $\leq$ 5 mr/hr contact floor

5. Room 317

   $\leq$ 6,500 c/m and $\leq$5 mr/hr window and frame
   $\leq$ 3 mr/hr contact all wallboards and door frame
   $\leq$ 32,000 c/m baseboard
   $\leq$ 3 mr/hr contact floor

After decontamination no gamma-ray readings could be detected anywhere with the exception of Room 507. In Room 507 there was a hot area on one wall reading 1.5 mr/hr and over an area of approximately 5 feet readings of approximately 0.2-0.5 mr/hr can be found. Since this is on bearing walls no attempt was made to decontaminate by removal of brick. The areas were painted with 2 coats of lead paint. On the opening for the skylight the concrete reads up to 0.2 mr/hr in several small areas (approximately 10 in^2). The overall readings in the room are approximately equal to normal background of about 0.01 mr/hr. Two hundred dry smears taken in the various rooms gave counts of less than 10 α counts/minute.

Since the acceptable limits set were 2500 fixed α counts maximum and 1000 average and 0.2 mr/hr maximum gamma and 0.05 mr/hr average gamma readings, we have more than met the standards we set.

Copies to: W. W. Meinke
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Public Health

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Radiological Health Division

April 18, 1969

Final Report of Radiation Protection Evaluation of Washington
Technical Institute Buildings, Formerly Occupied by NBS.

In the absence of Mr. John V. Brink, Chief, Bureau of Public Health
Engineering of this Department, I was asked to report to you our findings
and recommendations with respect to the radioactive contamination of the
subject buildings.

There is attached a copy of a letter from Mr. Brink, (Attachment 1)
which briefly outlines the problem initially discussed with you in February.
Following the first discussions of this problem with Dr. Schwabel, Chief,
Health Physics Section of the National Bureau of Standards, our staff made
an exhaustive survey of all buildings formerly occupied by NBS and scheduled
for release to your Department. Our findings were reported to Dr. Schwabel
who arranged for decontamination of the facilities.

There is enclosed a copy of the final report of Dr. Schwabel following
the completion of the decontamination. (Attachment 2). Following these
cleaning operations by Dr. Schwabel's staff, members of our Division made
a final reevaluation of all areas. We are in substantial agreement with the
conclusions of Dr. Schwabel, however there remain three areas of concern,
all located in Building #2:

1. Room 507

Although we do not feel that a radiation hazard exists in this
room at the present time, radiation measurements show that an unknown quantity
of radium is still present in cracks, pits, holes, and painted cinderblock.
This material is largely fixed contamination and is no hazard as long as it
remains fixed. Unfortunately, however, the normal aging processes coupled
with the possibility of unusual use or wear and tear may uncover this radium
creating a hazardous, or at least, an alarming condition. For this reason
and the fact that this space does not appear to be very useful, we recommend
that the doorway be bricked over and the skylight sealed to permanently eli-
minate the availability of this room.