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PERSONNEL ALBEDO NEUTRON DOSIMETER WITH THERMOLUMINESCENT ^{6}LiF AND ^{7}LiF

J. E. HOY

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Savannah River Laboratory

Aiken, South Carolina

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PERSONNEL ALBEDO NEUTRON DOSIMETER WITH THERMOLUMINESCENT ^{6}LiF AND ^{7}LiF

by

J. E. Hoy

Approved by

E. L. Albenesius, Research Manager
Radiological Sciences Division

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E. I. DU PONT DE NEMOURS & COMPANY
SAVANNAH RIVER LABORATORY
AIKEN, S. C. 29801

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ABSTRACT

A belt-mounted thermoluminescent neutron dosimeter (TLND) that measures body-reflected neutrons and relates them to dose equivalent was designed at Savannah River. The dosimeter contains thermoluminescent LiF chips (enriched in ^{6}Li or ^{7}Li), which are more easily read and accurately interpreted than nuclear emulsions previously used. The dosimeter responds to a much wider energy range of neutrons than nuclear emulsions and is as accurate as most neutron survey instruments for measuring continuous spectra from scattered neutrons. The directional sensitivity of the TLND is approximately the same as that of nuclear emulsions.

When the TLND is exposed to monoenergetic neutrons above 1 Mev or to a high-energy neutron source in air at <200 cm, the response of the TLND is about one-third the estimated dose equivalent at the surface of the body. With low-energy spectra, the TLND overresponds with a relative response <2. The sensitivity range of the TLND is from 10 mrem to 50,000 rem with no observed rate dependence up to 7×10^7 rem/sec. The TLND result is not affected by exposure to gamma radiation (with gamma-to-neutron dose equivalent ratios of 100:1), except at low neutron exposures.

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PERSONNEL ALBEDO NEUTRON DOSIMETER WITH THERMOLUMINESCENT ^{6}LiF AND ^{7}LiF

INTRODUCTION

A method for determining a radiation worker's neutron dose equivalent is required by national and international regulations for the purpose of controlling and limiting his exposure. In the past, health physicists have relied on nuclear track emulsions for deriving cumulative neutron exposures. Personnel monitoring using nuclear emulsions is adequate for neutron energies greater than 0.5 Mev but fails at neutron energies below 0.5 Mev or when the associated gamma exposure exceeds 0.5 R. Because shielding in production facilities degrades the neutron energy so that a large fraction of the exposure falls below the film response threshold, a personnel neutron dosimeter is needed to detect neutrons in this energy range.

Since the discovery and development of thermoluminescent dosimetry, particularly the LiF dosimeters, considerable effort has been directed toward the personnel neutron dosimetry problem. Because LiF enriched in ^{7}Li responds primarily to gamma radiation, and LiF enriched in ^{6}Li responds to both gamma radiation and neutrons of thermal and resonance energies, the radiation characteristics of these materials have been extensively investigated. Preston[1] used packets of ^{6}LiF and ^{7}LiF powder to detect albedo neutrons (neutrons, primarily of thermal energy, reflected by a human body) as a means of dosimetry. A similar dosimeter for use around nuclear power stations gave promising results.[2,3] Hankins[4] studied the energy dependence of thermal neutron detectors placed at various positions around the head and found a large energy dependence to some types of spectra. Several other prototype albedo neutron dosimeters have been evaluated.[5,6]

A prototype thermoluminescent neutron dosimeter was evaluated by Korba and Hoy[7] in 1969. This albedo-type design responded satisfactorily in limited laboratory studies, but was unsatisfactory in field tests because of its large size and because the mounting clip did not maintain a constant distance between the body and the dosimeter. An air gap of 1/4 in. between the badge and the body reduced the response by 50%. A new, belt-mounted dosimeter was designed, and its performance is described in this report.

THE TLND BADGE

The TLND badge (Figure 1) was designed to respond to albedo neutrons, which can be related to the total neutron exposure under most conditions. Albedo neutrons and associated gamma activity are detected by a pair of small thermoluminescent dosimeter chips placed in a moderating hemisphere of polyethylene that is shielded from incident thermal neutrons by a cadmium dome. Another pair of dosimeter chips in a small compartment in the dome of the hemisphere detects a portion of the incident activity and thereby provides a correction for overresponse to low energy spectra. The badge components are enclosed in a protective stainless steel case in the shape of a 2-in.-diameter hemisphere. The case is mounted on a belt that is worn around the waist. The badge becomes a neutron dosimeter when it is mounted against a human body or other suitable reflecting material.

Components

The components of the TLND are thermoluminescent dosimeter chips, polyethylene moderator, cadmium shield, stainless steel case, and belt.

Thermoluminescent Dosimeter Chips

Two pairs of thermoluminescent dosimeter chips are used in the badge. Each pair consists of one chip of LiF enriched in ^{6}Li (TLD-600 from Harshaw Chemical Company) and one chip of LiF enriched in ^{7}Li (TLD-700 from Harshaw). The TLD-600 chips are rectangular (0.10 in. x 0.15 in. by 0.035 in. thick), and the TLD-700 chips are square (0.125 in. x 0.125 in. by 0.035 in. thick).*

The TLD-700 chips respond primarily to gamma radiation. The TLD-600 chips respond to gamma radiation, thermal neutrons, and neutrons of certain resonance energies.

Batches of chips of each type are selected so that the gamma response of each chip is within ±10% of the batch mean; 94% of the chips were found to be within ±5% of the batch mean at 1 R.

* Before the rectangular TLD-600 chips became available, a 0.025-in.-diameter hole was drilled in the center of square TLD-600 chips to distinguish them from the TLD-700 chips. The data in this report were obtained with square TLD-600 chips.

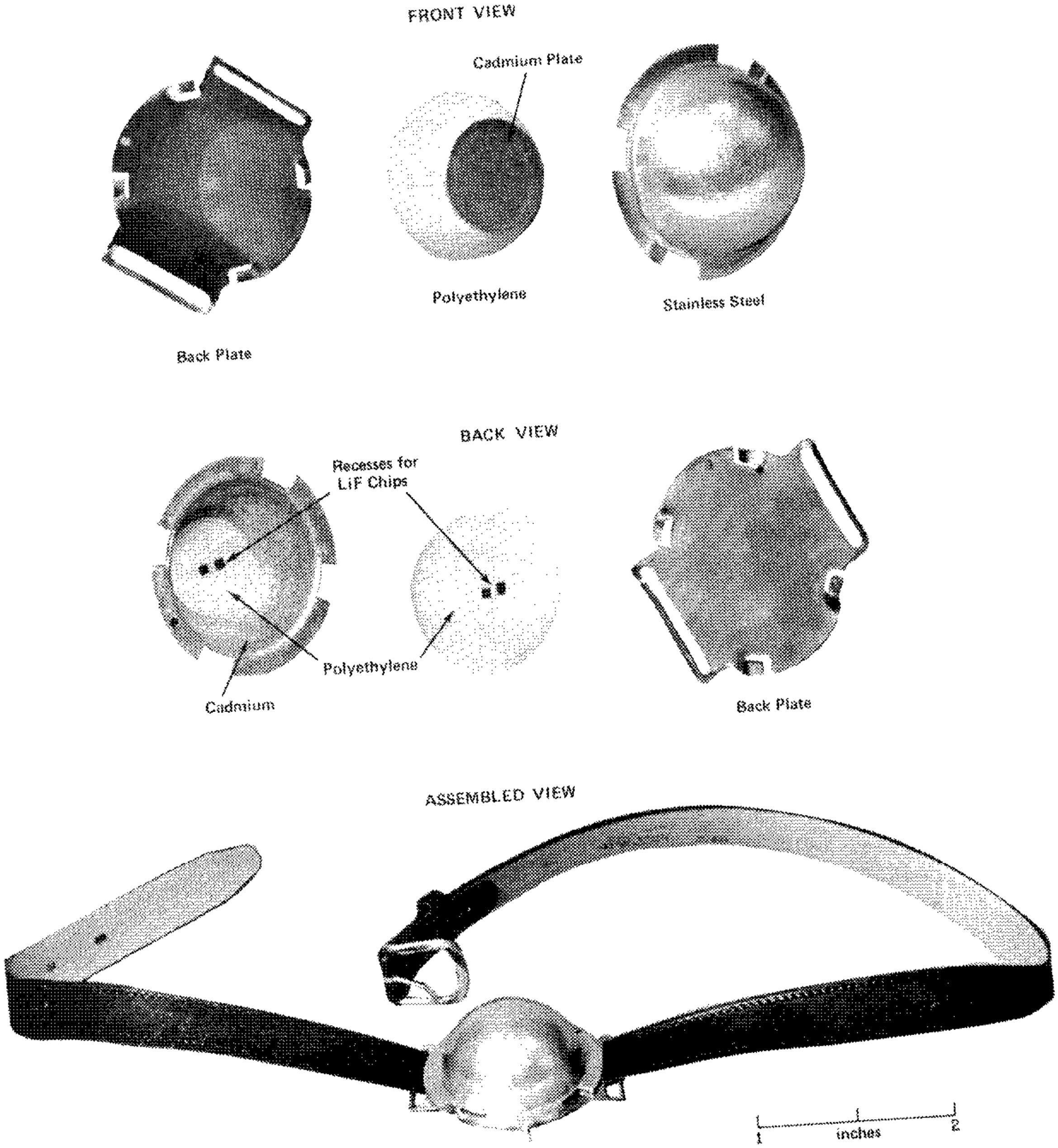

FIG. 1 THERMOLUMINESCENT NEUTRON DOSIMETER BADGE

Variation in the thermal neutron response of the TLD-600 chips was the largest contributor to variation in the dosimeter response. The batch variation was ±8.7% at one standard error. This large batch variation is attributed to slight differences in the ^{6}Li content of the TLD-600 chips and to a non-isotropic thermal neutron fluence during test exposures.

Before the chips are loaded in the badge, they are annealed for one hour at 400°C, cooled rapidly and reproducibly, and held at 100°C for two hours. The annealing pans are made of 30-gauge nickel sheets, and each pan holds approximately 1,000 chips.

Polyethylene Moderator

The polyethylene moderator is a 2-in.-diameter hemisphere divided into two sections. The smaller section, a spherical segment, is 1/4 in. thick. The two sections are separated by a cadmium shield plate. Two recesses for dosimeter chips are bored near the center of the larger face of each section.

Cadmium Shield

A dome-shaped cadmium shield covers the polyethylene hemisphere, and a cadmium plate separates the two sections of the hemisphere. The cadmium plate and dome are 1/32 in. thick except a 1/2-in.-radius area in the center of the dome is only 0.003 in. thick. This thin area permits a small fraction of the incident thermal neutrons to reach the dosimeter chips in the smaller section of polyethylene. The cadmium dome is laminated to the stainless steel dome of the case.

Case and Belt

A protective stainless steel case surrounds the other components. The back plate of the case has slots for attaching the badge to a belt. A 1/8-in.-diameter hole in the rim of the case can be used to lock it closed.

The belt maintains the required close fit between the TLND badge and the body. All calibration studies have included the thickness of the belt (1/8-in. web) in the derivation of badge response.

Assembly of Badge Components

One pair of dosimeter chips (one TLD-600 and one TLD-700) is placed in the larger section of polyethylene moderator. This pair is shielded from most incident thermal neutrons but is exposed to albedo neutrons.

The other pair of dosimeter chips is placed in the smaller section (spherical segment) of polyethylene, which is shielded from most incident thermal neutrons by the cadmium dome and from albedo neutrons by the cadmium plate between the polyethylene sections. This pair responds to some incident radiation that passes through the thin area in the cadmium dome. This response is used to correct for the overresponse of the other pair of dosimeter chips to low energy neutrons.

DETERMINING DOSE EQUIVALENT WITH THE TLND

Reading the TLND

To read the neutron exposure accumulated on the TLND badge, the badge is disassembled, and the thermoluminescence of each dosimeter chip is read with an Eberline TLR-5 reader set for a 6-second preheat cycle at 125°C** and a 12-second read cycle at 285°C.** The nitrogen purge is set at 7 ft^3/hr during reading. Sensitivity is adjusted to give 1,000 counts on TLD-700 chips exposed to 1,000 mR of radium or ^{60}Co gamma radiation. The error in exposing and reading TLD-600 and TLD-700 chips at 1 R is ±4% at the 95% confidence level.

About 90 badges per day can be unloaded, read, and reloaded by one technician using one reader.

Calibrating the TLND

The TLND is calibrated by exposing it to a standard neutron source by the procedure given in Appendix B.

Calculating the Dose Equivalent

The dose equivalent* is calculated from the thermoluminescent readings on the dosimeter chips by the following equation, which was developed for use with a variety of neutron spectra:

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* The *dose equivalent* is discussed in Appendix A.

** Indicated pan temperature

$$DE = [(B_6R - B_7) - (S_6R - S_7)] \times CF \times \frac{\ln T}{-1.52} \qquad (1)$$

where

DE = the dose equivalent in mrem

B_6 = meter reading of TLD-600 chip in larger section of polyethylene, which is nearer the body

B_7 = meter reading of TLD-700 chip nearer the body

S_6 = meter reading of TLD-600 chip in smaller section of polyethylene, which is nearer the source

S_7 = meter reading of TLD-700 chip nearer the source

R = gamma response ratio of TLD-700/TLD-600

CF = calibration factor in mrem/count. The calibration factor is derived by the procedure given in Appendix B.

$\frac{\ln T}{-1.52}$ = an energy correction which is limited to positive values between 0.05 and 1.0. If the correction is negative or outside these limits, this quantity is set = 1.

$T = \frac{(S_6R - S_7)}{(B_6R - B_7)}$ and is the gamma-corrected ratio of the two TLD-600 chips.

Both CF and R will change with the sensitivity of the TLD material used. For the initial batch of material, $R = 1.44$ and $CF = 0.162$.

EVALUATION OF THE TLND

The performance of the TLND was evaluated by measuring the response to various continuous neutron spectra, the directional sensitivity, the effect of body size on the response, the upper and lower limits of sensitivity, the response to monoenergetic neutrons, and the response to scattered radiations from high-energy accelerators. The results of these evaluations are described in the following sections, and the TLND is compared to other monitoring devices in the final section. Previous experience with thermoluminescent dosimeters used for beta-gamma monitoring showed that fading, reproducibility, handling, and annealing characteristics of LiF dosimeters were all satisfactory.

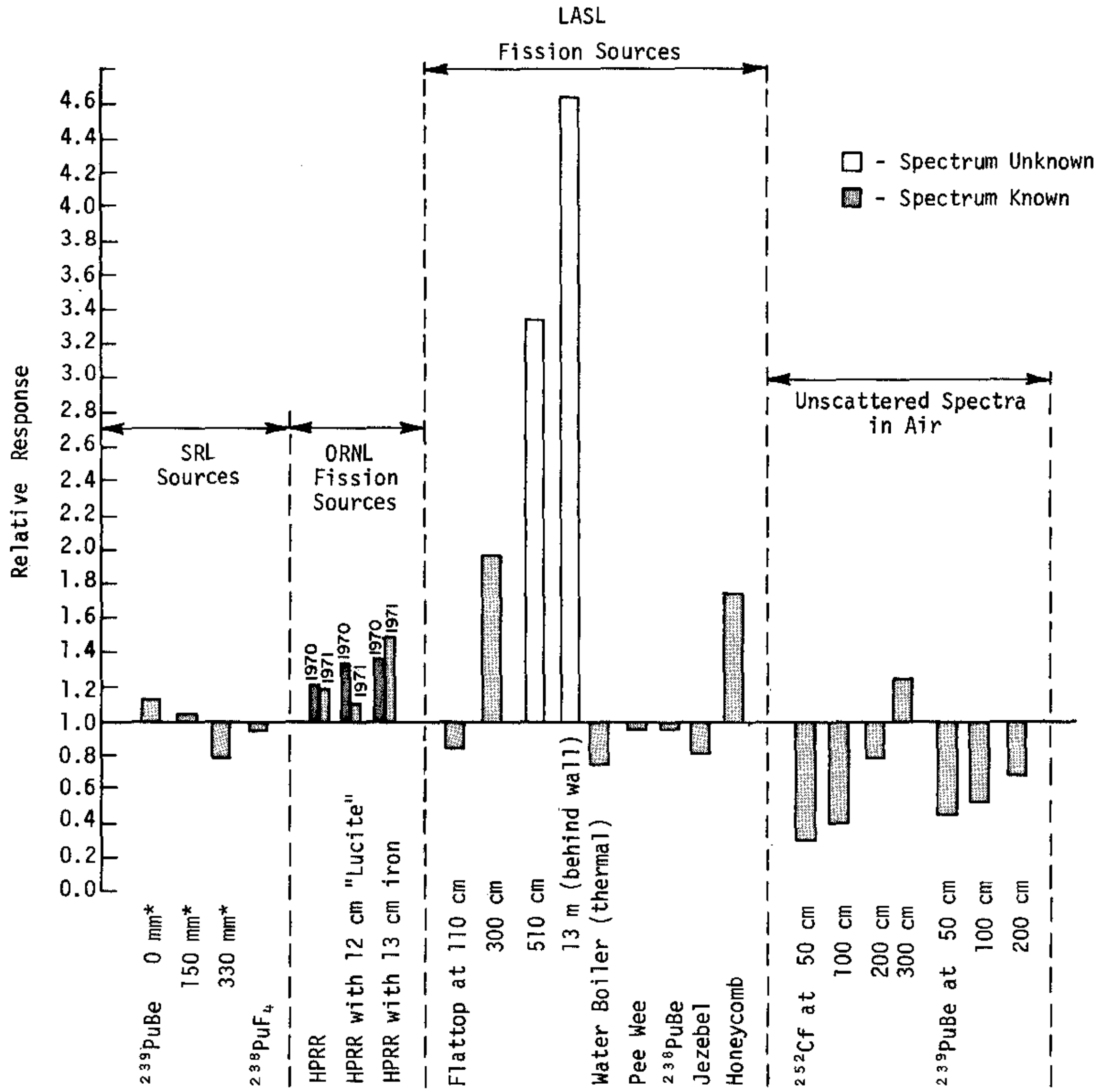
Response to Continuous Neutron Spectra

Exposure of TLND badges to neutron sources of various continuous spectra showed the TLND response is only slightly influenced by the neutron energy distribution of the source, but is strongly influenced by the amount of scattering materials near the source. For scattered spectra the tendency is to overrespond, particularly at large source-to-detector distances.

In these tests, a TLND badge was strapped to a body phantom (a 6-gal polyethylene jug filled with water) and exposed from the front to a neutron source with continuous energy spectrum. The spectra of the sources are described in Appendix C. The relative response to each spectrum is given in Figure 2 (the relative response is defined as the dose equivalent measured by the TLND divided by the calculated dose equivalent for the exposure).

With the exception of two spectra from the Flattop Assembly (at 510 cm and a second point 13 m behind a wall), the results of scattered spectra were between -20 and +100%. The estimated dose equivalent from the Flattop Assembly at 510 cm was based on an inverse square reduction of dose equivalent calculated from 110 cm. Because the inverse square method excludes scatter, the estimated dose equivalent is probably low, and the observed relative response, high. In the other case (Flattop at 13 m), the phantom was located behind a concrete wall where the spectrum was unknown and completely scattered. Data from Threshold Detector Units (TDU) exposed at this location were incomplete in the energy region below 750 kev.[8] The dose equivalent for this exposure was estimated to be 84 rem from multisphere[9] and available TDU data. The dose equivalent was 391 rem with the TLND, 11.7 rem with the proportional chamber, 121 rem with a 10-in.-dia. sphere, and 22 rem with nuclear emulsion film normalized to this exposure. The TLND response is high, but probably by a lower value than indicated in Figure 2.

The response of the TLND to unscattered spectra is low and represents a limitation of the badge that cannot be resolved



* Distance inside heavy water tank.

FIG. 2 RELATIVE RESPONSE OF TLND FOR ALL TEST SPECTRA without knowledge of the exposure conditions. Attempts to differentiate between scattered and unscattered conditions by using the readings from the two pairs of LiF chips included in the dosimeter have been unsuccessful.

The movement of an unscattered source 1 m off the floor from the center of a large room toward a concrete wall will increase the relative response of the neutron dosimeter from a low of 0.52 (unscattered) to 0.80 when the source is 100 cm from the wall. When the source is in contact with the wall, the relative response increases to 1.10 of its actual dose rate from direct and scattered neutrons (Figure 3). The critical factor appears to be the nearness of the source to a material that will produce scatter.

At Savannah River and other nuclear production plants, personnel are unlikely to be exposed to neutron sources in air or other scatter-free media for periods of time that would produce significant exposures. Shielding and containment of neutron sources preclude direct personnel exposure. The most probable exposure would, therefore, be from scattered neutrons, and calibration on this type of exposure is recommended with the understanding that if a scatter-free exposure occurs, a special interpretation of the badge will be required.

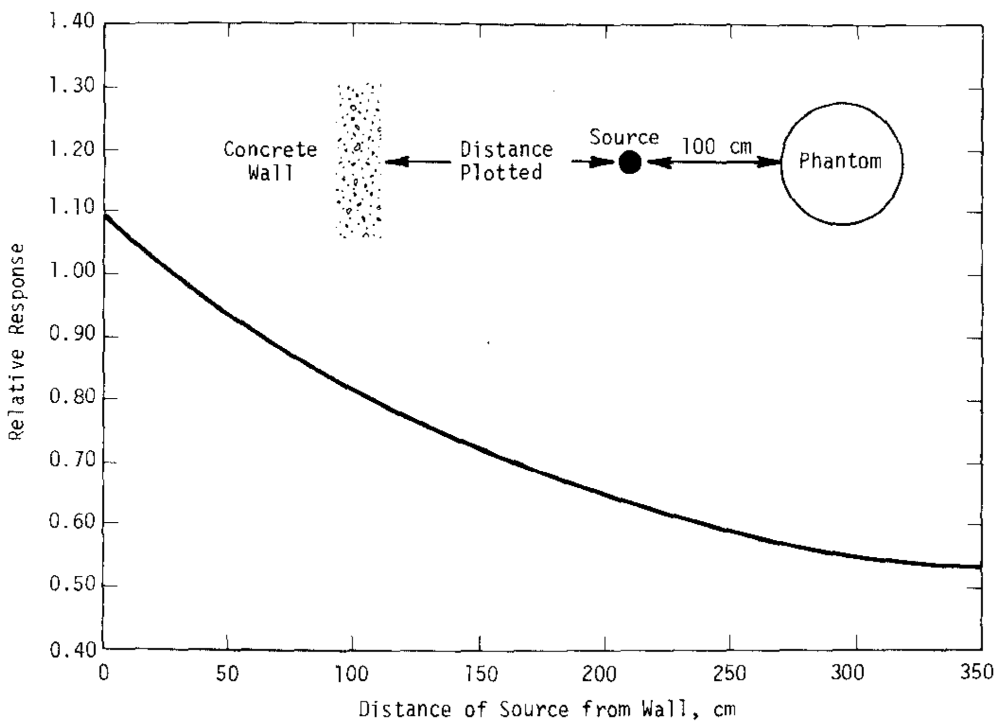


FIG. 3 EFFECT OF SCATTER ON RESPONSE OF TLND

Directional Effects on TLND Response

Directional sensitivity with the TLND is not significantly different from that with nuclear emulsions or other types of detectors worn at one position on the body. If the dosimeter and body phantom are continuously rotated during the exposure, the relative response to a $^{238}PuF_4$ spectrum is 0.65 (compared to 1.00 for a stationary front exposure). The response of NTA film rotated during calibration is 0.68 of the stationary frontal response.[10]

When TLND badges were exposed from the front, side, and rear to various spectra, the badge response depended somewhat on the spectra, but the average response from each direction showed the general trend given in Table I.

TABLE I

Relative Response of TLND and Nuclear Emulsion Film

	Average Relative Response of TLND	Relative Response of NTA Film[11,12]
Front	1.02	1.00
Side	0.65	0.51
Rear	0.16	0.10

Effect of Body Size on TLND Response

The response of the TLND, when worn around the waist, will not be affected by variations in body size. Tests with polyethylene as a backing material for the badge and belt showed that at least 3 in. of hydrogenous material is needed behind the badge for proper response. This material must also extend at least 2 in. to all sides of the badge. At a moderator thickness of 2 in., the response is low by 20%; at a thickness of 1 in., the response is low by 30%.

Sensitivity Range

The lower limit of sensitivity of the TLND depends on uniformity of response of the thermoluminescent material and the level of associated gamma exposure. In one experiment, a neutron dose equivalent of 10 mrem was added to a gamma exposure of 1,000 mR. The dosimeter recorded an average neutron dose equivalent of 5.3 mrem at a standard error of ±160%. In the absence of additional gamma exposure, a 5 mrem exposure resulted in an average dose equivalent of 5.1 mrem at a standard error of ±4%.

The response of the TLND at various dose equivalent rates is shown in Table II.

The highest dose equivalent measured under controlled conditions was 46,000 rem at the Flattop Assembly[8] at Los Alamos Scientific Laboratory. The TLND result was lower than the estimated dose equivalent for this exposure by 10%. The dose equivalent rate during the exposure was 7,700 rem/sec.

The highest dose equivalent rate was obtained at ORNL during the Seventh Nuclear Accident Dosimetry Intercomparison Study in July 1970. An estimated dose equivalent rate of 7×10^7 rem/sec was generated during the first test burst of HPRR. The TLND response was 21% higher than the estimated dose equivalent for this exposure.*

TABLE II

TLND Response to Various Spectra

Source[a]	Spectrum Figure No.	Dose Equivalent Rate, mrem/hr Calculated	TLND
$^{238}PuF_4$	C-1	13.3	12.5
^{239}PuBe at 0 mm[b]	C-2	34.2	39.6
^{239}PuBe at 150 mm[b]	C-3	12.3	13.4
^{239}PuBe at 330 mm[b]	C-4	2.1	1.7
^{252}Cf[c]	C-5	4.7	1.9
^{239}PuBe[c]	C-6	19.2	9.0

a. Sources and their spectra are described in Appendix C.

b. Distance inside heavy water tank. TLND 81 cm from face of tank.

c. Unscattered spectra.

* HPRR exposure data are usually interpreted in units of dose (rads). Conversion to dose equivalent was based on measured fluence and the spectral distribution published by Auxier.[13,14]

Response to Monoenergetic Neutrons

In a single series of exposures to monoenergetic neutrons, the TLND overresponded between 140 and 450 kev with a peak at 250 kev of +70% (Figure 4). Above 600 kev, the response dropped rapidly to ∿0.25 and was relatively constant at this value between 2 and 9 Mev. These data agree well with results from ^{239}PuBe unscattered spectra at 50 cm, which show a 0.42 response for the 4.3 Mev average energy spectrum. Energies below 120 kev, except for a thermal exposure point, were not available.

In these tests, the TLND was exposed to neutrons from proton-target reactions at Battelle-Northwest Laboratories and Wright-Patterson Air Force Base in a scatter-free environment. Neutrons were normally incident to the front of the phantom. Results were normalized to a $^{238}PuF_4$ calibration and represent the relative response of the TLND to monoenergetic neutrons. The monoenergetic response curve (Figure 4) shows how scattering of high-energy neutrons could increase the response of the dosimeter by adding more neutrons in the lower energy regions where the TLND response is high.

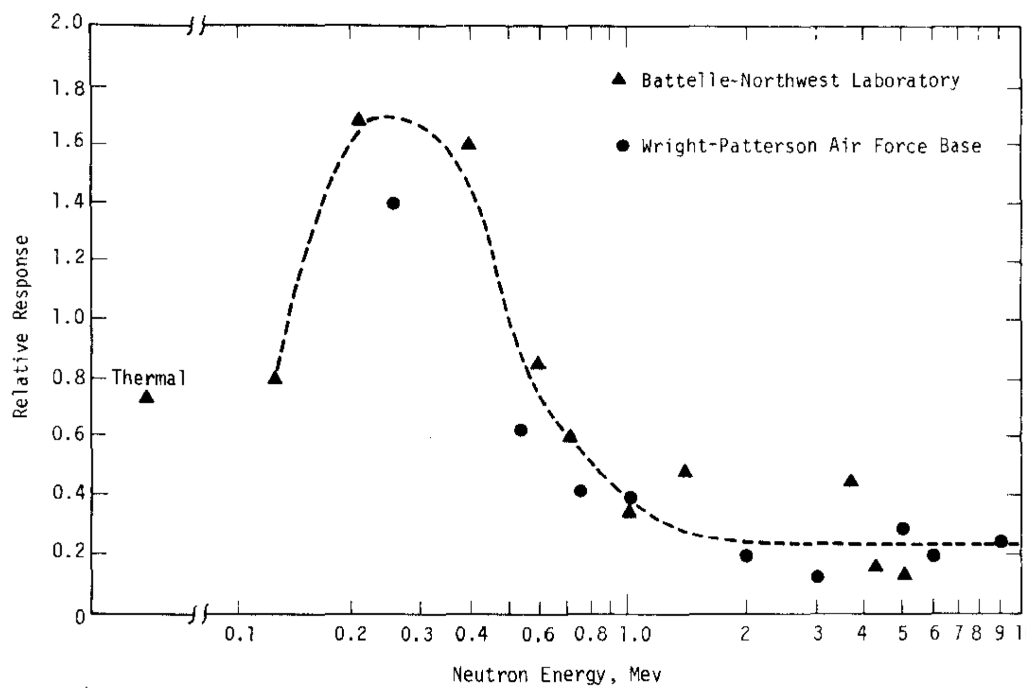


FIG. 4 UNSCATTERED MONOENERGETIC NEUTRON RESPONSE

Response to Scattered Radiation from High-Energy Accelerators*

The TLND is acceptable for determining dose equivalent from sources with high-energy scattered neutrons. Personnel neutron dosimetry is needed around occupied areas adjacent to accelerator beams, but film dosimetry cannot be used for energies >30 Mev.

In tests with high-energy scattered neutron sources, the TLND was mounted on a 6-gal polyethylene jug of water placed in the area to be evaluated. NTA films, a 10-in.-dia. Bonner sphere, and a set of spheres of various diameters (Bonner Spectrometer) were exposed at the same time. The dose equivalent for each exposure was based on multisphere spectrometer data normalized with a ^{239}PuBe spectrum. Although there is some uncertainty in average energy, the dose equivalent is reliable because the fluence-to-dose equivalent conversion factors change gradually with energy. Results from the TLND and the 10-in.-dia. Bonner sphere are compared to results from the multisphere spectrometer in Figure 5.

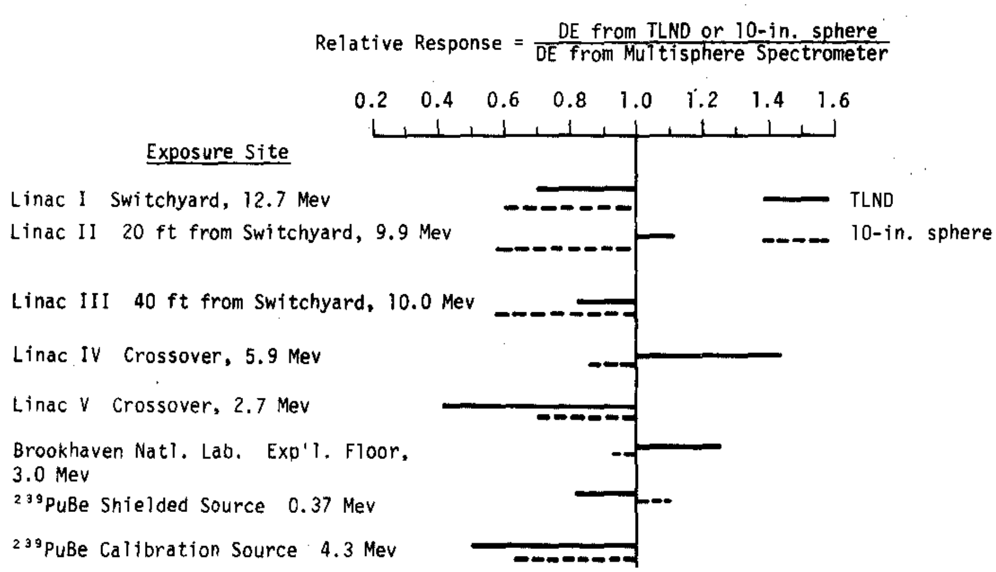


FIG. 5 RELATIVE RESPONSE OF TLND AND 10-in. SPHERE FOR SPECTRA AT NATIONAL ACCELERATOR LABORATORY

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* Exposure and interpretation by Radiation Physics Section of National Accelerator Laboratory, Batavia, Ill.

Comparison of TLND with Nuclear Track Emulsion Film and Neutron Survey Instruments

A completely energy-independent neutron dosimeter or neutron survey instrument does not exist. All have limitations in the low energy or high energy sections of the spectrum. Nuclear track emulsion is widely used as a neutron dosimeter for personnel monitoring in industry today. It is generally satisfactory if the fluence is primarily due to high energy neutrons above approximately 0.5 Mev, or if a special calibration is made to the exposing spectrum.

The responses of some neutron monitoring devices are compared to that of the TLND in Figure 6. With the exception of NTA film, all neutron survey devices were generally within ±40% of the calculated dose equivalent.

The large variation in the energy response of NTA film when exposed to these selected spectra demonstrates one of the major problems of calibration and use of film for personnel dosimetry. If NTA film is calibrated to a fast spectrum, the low energy spectra will be underestimated by approximately 100%. If NTA is calibrated to a $^{238}PuF_4$ spectrum (at SRL this spectrum is representative of most production material), the film will overestimate the dose equivalent from the higher energy. The TLND is much less energy dependent than NTA film for the spectra tested. In addition, the TLND may be used over an extended time cycle without fading and is less sensitive to gamma interference.

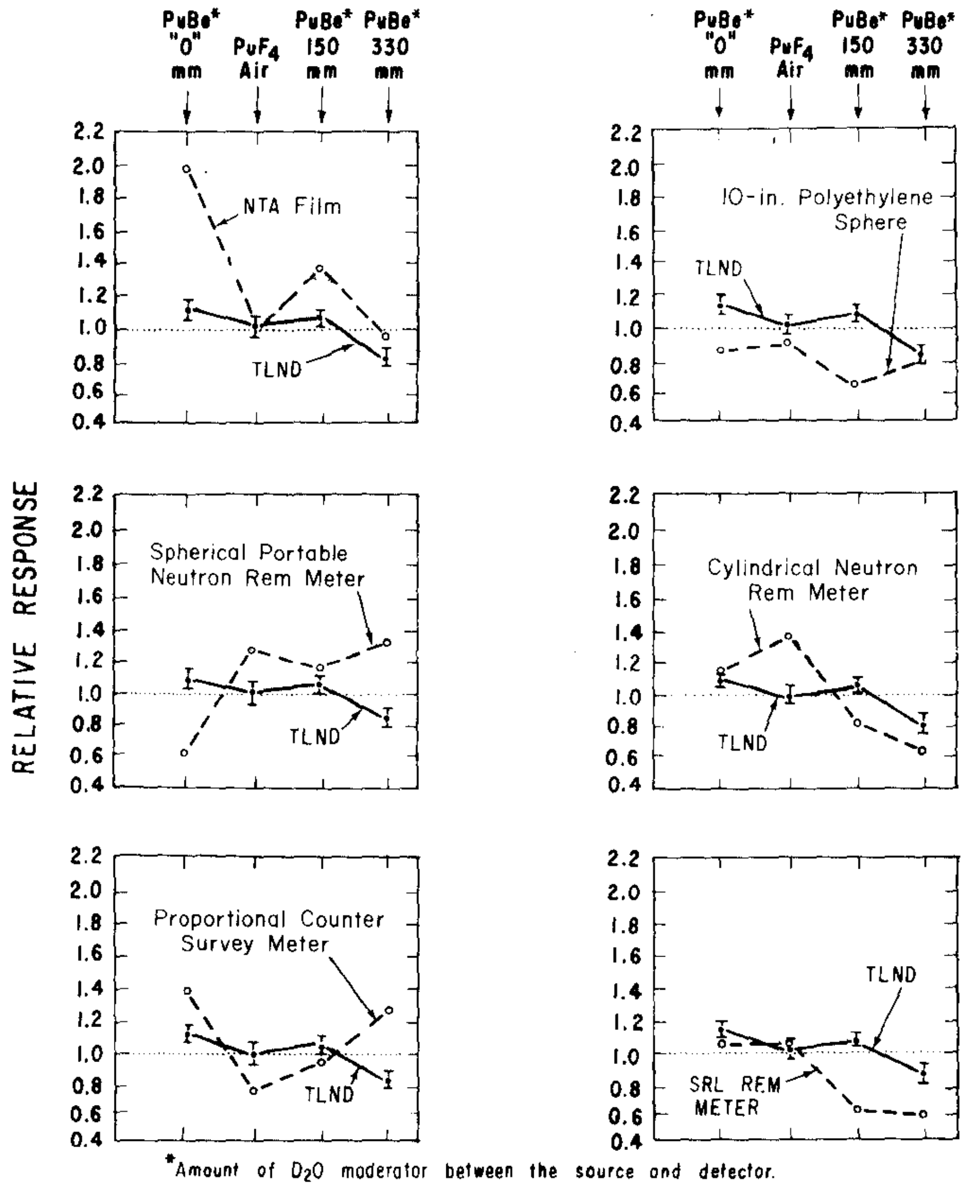

FIG. 6 TLND RESPONSE COMPARED TO NUCLEAR TRACK EMULSION FILM AND NEUTRON SURVEY INSTRUMENTS

ACKNOWLEDGMENTS

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APPENDIX A

INTERPRETING THE DOSE EQUIVALENT

The *dose equivalent* is the risk-related quantity recommended by national[15] and international[16] regulatory bodies for control of personnel exposures to neutron radiation. The unit of dose equivalent (DE) is the *rem*, which is defined as the product of absorbed dose (D) in *rads* times a quality factor (QF), a distribution factor (DF), and other modifying factors. The QF is unity for X-rays and gamma radiation but is related to the linear energy transfer of the more densely ionizing particles. QF is intended to be a factor, over and above consideration of absorbed energy alone, to account for observed tissue damage. For the present, the DF and other modifying factors are assumed to be unity.

It is obvious from the above definition that DE cannot be directly measured and must be derived from the neutron fluence, energy distribution, type of tissue, depth in tissue, tissue sensitivity, and uniformity of exposure. Neufeld[17] has critically reviewed radiation quantities and units used in protection measurements by health physicists, and the reader interested in a fuller discussion of existing problems is referred to his review.

Fortunately, fluence-to-DE conversion factors at specific energies and depths in soft tissue have been published.[18,19] These permit the DE to be derived for neutron spectra of known energy distribution, but the result depends on the method of interpretation.

The two most common differences in interpreting DE from published data depend on whether an isolated unit mass of tissue or a large slab of tissue is selected as the basis for deposition of energy. The former excludes n,γ capture in surrounding tissue, while the latter includes these interactions. Differences between the two are small for neutron spectra having most of their fluence greater than 0.2 Mev but are large for the lower energy components. Figure A-1 shows the differences in D and DE derived by these two interpretations.

This report is based on dose equivalent at the surface of an infinite slab of tissue 30 cm thick. In this interpretation, the DE is slightly higher than that published by Auxier[19] for the front segment (3 cm thick) of a 60-cm-high right circular cylinder that is 30 cm in diameter. The surface value was chosen to approximate the DE received by the lens of the eye.

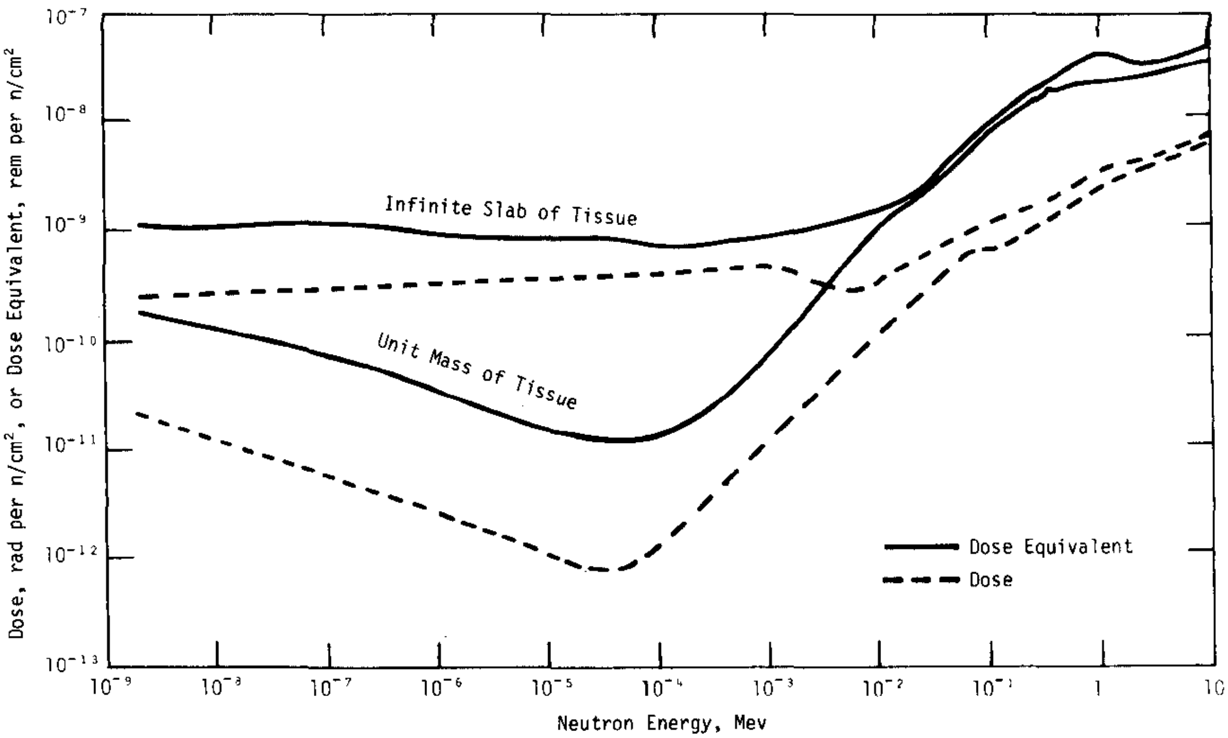

FIG. A-1 DOSE AND DOSE EQUIVALENT FOR VARIOUS NEUTRON ENERGIES

APPENDIX B

CALIBRATING THE TLND

There are no nationally accepted standards for calibrating neutron dosimeters in terms of exposure dose equivalent (DE). Each investigator must calibrate the dosimeter with the sources he has available and choose the unit of absorbed dose (D) and quality factors (QF) for each spectrum to derive the corresponding DE.

At Savannah River, the TLND is calibrated with a unique $^{238}PuF_4$ neutron source because the neutron spectrum from this source is representative of personnel exposures there. However, the TLND may also be calibrated with ^{239}PuBe or ^{252}Cf standard sources.

Calibration Procedure

1. Determine R (the gamma response ratio of TLD-700/TLD-600) as follows:

   a. Expose 25 TLD-600 chips and 25 TLD-700 chips to 1 R of gamma radiation from a ^{60}Co, radium, or ^{137}Cs source. (The chips do not have to be in badges, but should be enclosed in sufficient material to assure electron equilibrium.)

   b. Read the resulting thermoluminescence of each chip with the reader adjusted to respond one unit per mR of exposure on the TLD-700 chips.

   c. Calculate the average thermoluminescence of each type of chip.

   d. Calculate R from these average readings.

2. Expose at least three TLND badges (one at a time) to the neutron source as follows:

   a. Strap the badge to the center of a body phantom (a 6-gallon polyethylene jug filled with water to simulate backscatter from the body).

   b. Position the phantom so that the badge faces the neutron source and is in the same horizontal plane as the source. The front of the phantom (rear of the badge) should be 81 cm from the center of a $^{238}PuF_4$ source in a scattered geometry* or 100 cm from the center of a $^{239}PuBe$ or ^{252}Cf source in a scatter-free geometry.

   c. Expose the badge to the source for a suitable period of time (one or more hours with the $^{238}PuF_4$ source) and record the exposure time.

   d. Calculate the known dose equivalent from the exposure time.

3. Calculate CF for each badge as follows, and then calculate an average CF. For calibrations with the $^{238}PuF_4$ source, use the following equation:

$$CF = \frac{DE}{[(B_6R - B_7) - (S_6R - S_7)] \ln T/-1.52} \qquad (B-1)$$

   For calibrations with $^{239}PuBe$ or ^{252}Cf sources at a distance of 100 cm:

   - Use Equation B-1 if the badge will be used to measure personnel exposure from sources suspended in air away from scattering materials or if over-estimation of dose equivalent from scattered neutrons can be tolerated. Because these are high-energy neutron sources, the response of the dosimeter to unscattered neutrons will be low, and therefore the CF will be high.

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* Scattered geometry is provided by positioning the $^{238}PuF_4$ source 1 ft above a 55-gal paraffin-filled storage drum.

- Use the following equation if personnel are normally exposed to scattered neutrons, such as those in production facilities where shielding and other scattering materials are present:

$$CF = \frac{0.5\ DE}{[(B_6R - B_7) - (S_6R - S_7)]\ \ln T/-1.52} \qquad \text{(B-2)}$$

Calibration Check Procedure

The calibration of the badge may be checked periodically by exposing the badge to a standard source under the same conditions as in the "Calibration Procedure" above. If the dose equivalent determined by this exposure is not within ±10% of the known dose equivalent for the exposure, R and CF must be reevaluated.

APPENDIX C

NEUTRON SOURCES

The TLND was evaluated by exposures to a variety of neutron spectra from sources at the Savannah River Laboratory, Los Alamos Scientific Laboratory, Oak Ridge National Laboratory, Battelle-Northwest Laboratory, Wright-Patterson Air Force Base, and the National Accelerator Laboratory. The dose equivalent for each exposure was calculated from the neutron spectrum if it was available. The spectra for a few of the sources were not known; for these, the dose equivalent was estimated from measurements with survey instruments. The neutron sources and their spectra, the methods for determining the spectra, and the calculation of the dose equivalent received in each exposure are described in this appendix.

Neutron Sources at Savannah River

The TLND was exposed to six neutron spectra of widely different energy distribution at Savannah River. The sources of these spectra are listed in Table C-1.

Three of the spectra are produced by a 10-Ci ^{239}PuBe source in a tank (2 ft diameter by 4 ft long) of heavy water. The source can be positioned at different distances in the tank by a horizontal mechanical drive. At the front of the tank (designated the 0 mm position), moderation is minimum, but back-scatter is significant because of the heavy water behind the source. Other calibrated positions are at 150 mm and 330 mm in the heavy water. The dose equivalent derived for this source includes the effects of room and tank scattering.

The $^{238}PuF_4$ source has a scattered spectrum because the source is suspended 1 ft above a 55-gal paraffin-filled storage drum. The neutron spectra from the ^{252}Cf source and the ^{239}PuBe source in air are unscattered because these sources are suspended in air in a large room.

For each source, the neutron spectrum above 0.2 Mev was determined with a ^{6}Li and ^{3}He commercial neutron spectrometer. (Spectrometers do not respond over the entire energy range, but have a low energy cutoff between 0.2 and 0.5 Mev.) For some of the sources, the neutron fluence below 0.2 Mev was determined by measuring the fluence in several energy intervals as follows:

Energy Interval	Method of Measurement
Thermal - 10 Mev	A. Long Counter[20,21] calibrated with a 10-Ci ^{239}PuBe source in air.
0.4 ev - 10 Mev	B. Long Counter with cadmium shield between source and counter.
0.2 Mev - 10 Mev	C. Proportional Counter calibrated with 10-Ci ^{239}PuBe source in air. The Proportional Counter detects proton recoil and with gamma discrimination will not respond to neutrons below $\sim$0.2 Mev.
Thermal - 0.4 ev	Difference between A and B
0.4 ev - 0.2 Mev	Difference between A and C

In addition, the room scatter was estimated by placing 6 in. of polyethylene and cadmium in front of the source and measuring the fluence at various angles on either side of the normal direction of measurement with the Long Counter. This scattered fluence was converted to dose equivalent by assuming a direct proportion of the scattered fluence to total fluence.

The dose equivalent from each source was derived from the source spectrum as follows:

- The area under the curve was measured for each energy interval of the spectrum, and the fraction of the total area was calculated for each energy interval.
- This fraction was multiplied by the total flux to obtain the flux in each energy interval.
- The conversion factor for each energy interval was read from the top curve in Figure A-1.
- This factor was multiplied by the corresponding flux to obtain the dose equivalent rate for each energy interval.
- These rates were added to yield the total dose equivalent rate.

The derivations of spectra for sources at Savannah River are given in Figures C-1 through C-6.

Other Neutron Sources

The TLND was exposed to the neutron sources at Los Alamos Scientific Laboratory and Oak Ridge National Laboratory listed in Tables C-2 and C-3. The neutron spectra of most of these sources and the derivations of the resulting dose equivalents are given in Figures C-7 through C-12.

TABLE C-1

Exposures to Continuous Spectra at Savannah River Laboratory

Neutron Source	Distance from Source	Spectrum Type[a]	Spectrum Figure No.	TLND Readings, counts B_7	B_6	S_7	S_6	Dose Equivalent, rem TLND	Calc[b]	Relative Response
$^{238}PuF_4$	81 cm	S	C-1	34	634	41	86	0.208	0.203	0.98
^{239}PuBe (10 Ci) at 0 mm[c]	81 cm[d]	S	C-2	21	190	22	45	0.041	0.037	1.10
^{239}PuBe (10 Ci) at 150 mm[c]	81 cm[d]	S	C-3	20	312	25	103	0.042	0.034	1.24
^{239}PuBe (10 Ci) at 330 mm[c]	81 cm[d]	S	C-4	29	382	43	178	0.030	0.036	0.84
^{252}Cf (3 μg)	50 cm	U	C-5	29	306	33	47	0.103	0.306	0.34
	100 cm	U	C-5	22	116	24	27	0.032	0.076	0.42
	200 cm	U	C-5	17	48	17	14	0.014	0.018	0.80
	300 cm	U	C-5	26	319	33	69	0.059	0.048	1.23
^{239}PuBe (10 Ci) in Air	50 cm	U	C-6	24	200	27	33	0.068	0.145	0.47
	100 cm	U	C-6	30	457	38	64	0.155	0.319	0.49
	200 cm	U	C-6	43	588	48	86	0.190	0.290	0.66

a. S = scattered; U = unscattered.
b. Calculated from neutron spectrum.
c. Distance inside heavy water tank.
d. Distance from face of tank.

TABLE C-2

Exposures to Continuous Spectra at Los Alamos Scientific Laboratory

Neutron Source	Distance from Source	Spectrum Type[a]	Spectrum Figure No.	Reported Dose[b] Fast, rad	Reported Dose[b] Thermal, rem	TLND Readings, counts B_7	B_6	S_7	S_6	Dose Equivalent, rem TLND	Dose Equivalent, rem Calc	Relative Response
Flattop Assembly	110 cm	S	C-7	2,930		2.36×10^6	1.63×10^8	2.89×10^6	2.65×10^7	41,438	45,975	0.90
	300 cm	S	*c*			8.0×10^3	7.9×10^5	1.00×10^4	8.5×10^4	262	130	2.01
	510 cm	S	*d*	142	25	2.56×10^5	2.64×10^7	3.55×10^4	3.27×10^6	7,724	2,220	3.48[c]
	13 m[e]	S	*f*	6.8	5.9	2.90×10^4	1.74×10^6	4.40×10^4	3.20×10^5	409	84	4.87[c]
Water Boiler Reactor	60 cm	S	*g*			510	2,063	732	1,536	0.054	0.072	0.75
PeeWee Assembly	95.5 cm	S	C-8[h]	1.84	1.91	14,162	174,400	23,200	21,890	39	36.5	1.06
^{239}PuBe	50 cm	U	C-6			235	5,506	297	547	2.20	5.22	0.42
		U	C-6			282	5,842	330	600	2.29	5.5	0.42
Honeycomb Assembly	3 m	S	C-8[h]			2.3×10^4	1.88×10^4	3.0×10^4	2.21×10^5	596	329	1.81
Jezebel Assembly	3 m	S	C-9			8.0×10^3	5.11×10^5	1.0×10^4	5.9×10^4	165	225	0.74
^{238}PuBe	30 cm	U	*i*			1.64×10^5	1.06×10^6	2.01×10^5	1.95×10^5	393	411	0.96
$^{238}PuF_4$ Moderated by 4 in. Polyethylene	30 cm	S	–			826	9,283	1,237	2,958	1.50	1.25	1.20
Control		–	–			94	80	97	74	8×10^{-4}	0.0	–

a. S = scattered; U = unscattered.
b. First-collision absorbed dose.
c. Spectrum assumed the same as at 110 cm.
d. DE estimated by inverse square from 110 cm exposure position (no correction for floor scatter).
e. Source behind wall.
f. DE estimated from multisphere measurement and threshold detector unit data.
g. Thermal neutrons.
h. Based on Parka spectrum.
i. 50 kev.

TABLE C-3

Exposures to Continuous Spectra at Oak Ridge National Laboratory[a]

Description of Neutron Source	Spectrum Type[b]	Spectrum Figure No.	Reported Dose, rad	TLND Readings, counts B_7	B_6	S_7	S_6	Dose Equivalent, rem TLND	Dose Equivalent, rem Calc	Relative Response
HPRR, unmoderated										
1970	S	C-10	331	1.42×10^5	1.36×10^7	1.64×10^5	1.12×10^6	5,210	4,110	1.27
1971	S	C-10	331	1.89×10^5	1.69×10^7	2.30×10^5	1.49×10^6	5,976	4,800	1.24
HPRR with 12 cm "Lucite"[d]										
1970	S	C-11	34.5	7.6×10^4	2.4×10^6	1.21×10^5	5.46×10^5	491	415	1.18
1971	S	C-11	34.5	9.8×10^4	2.9×10^6	1.47×10^5	6.63×10^5	588	443	1.33
HPRR with 13 cm iron										
1970	S	C-12	135	6.4×10^5	7.3×10^6	7.5×10^4	5.9×10^5	2,592	1,820	1.42
1971	S	C-12	135	8.6×10^4	1.0×10^7	9.6×10^4	9.3×10^5	3,421	2,520	1.36

a. Results are from TLND on front of phantom (Bomab) positioned 3 m from HPRR.
b. S = scattered; U = unscattered.
c. First-collision absorbed dose.
d. Trademark of DuPont.

Derivation of Dose Equivalent Rate from Neutron Spectrum

Energy Interval	Fraction of Total Area	Flux per Energy Interval, n/cm^2-hr	Conversion Factor, (mrem/hr)/(n/cm^2-hr)	Dose Equivalent Rate, mrem/hr
Thermal	Not shown	8.14	0.00375	0.03
0.4 eV to 2 Mev	Not shown	43.2	0.0324	1.40
0.2 to 0.5 Mev	0.078	7.18	0.065	0.47
0.5 to 0.7 Mev	0.076	7.18	0.10	0.72
0.7 to 0.85 Mev	0.090	8.28	0.12	0.99
0.85 to 1.0 Mev	0.091	8.35	0.13	1.08
1.0 to 3 Mev	0.668	61.32	0.14	8.58
Total	1.001	143.5		13.3

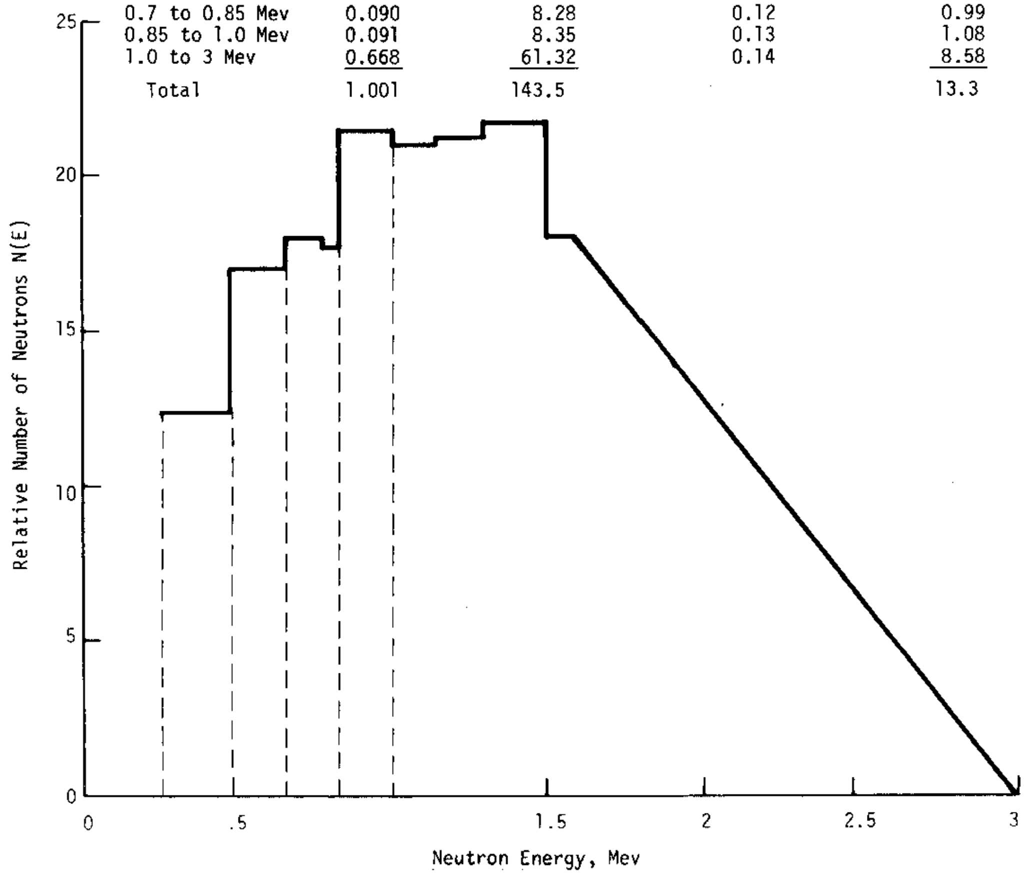


FIG. C-1 PuF_4 SOURCE IN AIR: NEUTRON SPECTRUM AND DOSE EQUIVALENT RATE 81 cm FROM SOURCE

Derivation of Dose Equivalent Rate from Neutron Spectrum

Energy Interval	Fraction of Total Area	Flux per Energy Interval, n/cm^2-hr	Conversion Factor, (mrem/hr)/(n/cm^2-hr)	Dose Equivalent Rate, mrem/hr
Thermal	Not shown	2.0	0.00375	0.12
0.4 eV to 0.2 Mev	Not shown	304	0.0324	9.85
0.2 to 0.5 Mev	0.030	5.52	0.068	0.38
0.5 to 1 Mev	0.164	30.17	0.122	3.68
1.0 to 2 Mev	0.250	46.00	0.137	6.30
2.0 to 5 Mev	0.373	68.63	0.129	8.85
>5 Mev	0.183	33.67	0.148	5.98
Total	1.004	521		34.2

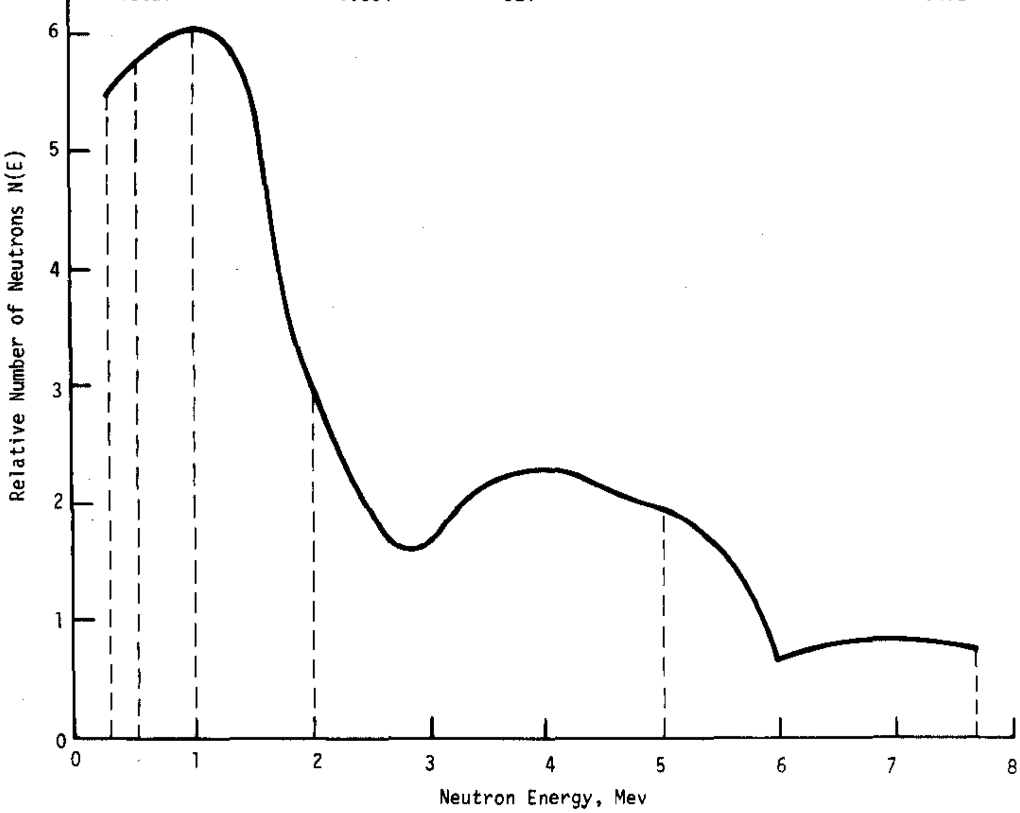


FIG. C-2 10-Ci ^{239}PuBe SOURCE 0 mm INSIDE HEAVY WATER TANK: NEUTRON SPECTRUM AND DOSE EQUIVALENT RATE 81 cm FROM TANK

Derivation of Dose Equivalent Rate from Neutron Spectrum

Energy Interval	Fraction of Total Area	Flux per Energy Interval, n/cm²-hr	Conversion Factor, (mrem/hr)/(n/cm²-hr)	Dose Equivalent Rate, mrem/hr
Thermal	Not shown	52.0	0.00375	0.2
0.4 eV to 0.2 Mev	Not shown	101.0	0.0324	3.3
0.25 to 1 Mev	0.210	10.5	0.122	1.3
1 to 2 Mev	0.358	17.9	0.137	2.5
2 to 5 Mev	0.285	14.2	0.129	1.8
5 to 7 Mev	0.147	7.4	0.148	1.1
Total	1.000	203		10.2
			Room Scatter at 81 cm	2.1
			Total	12.3

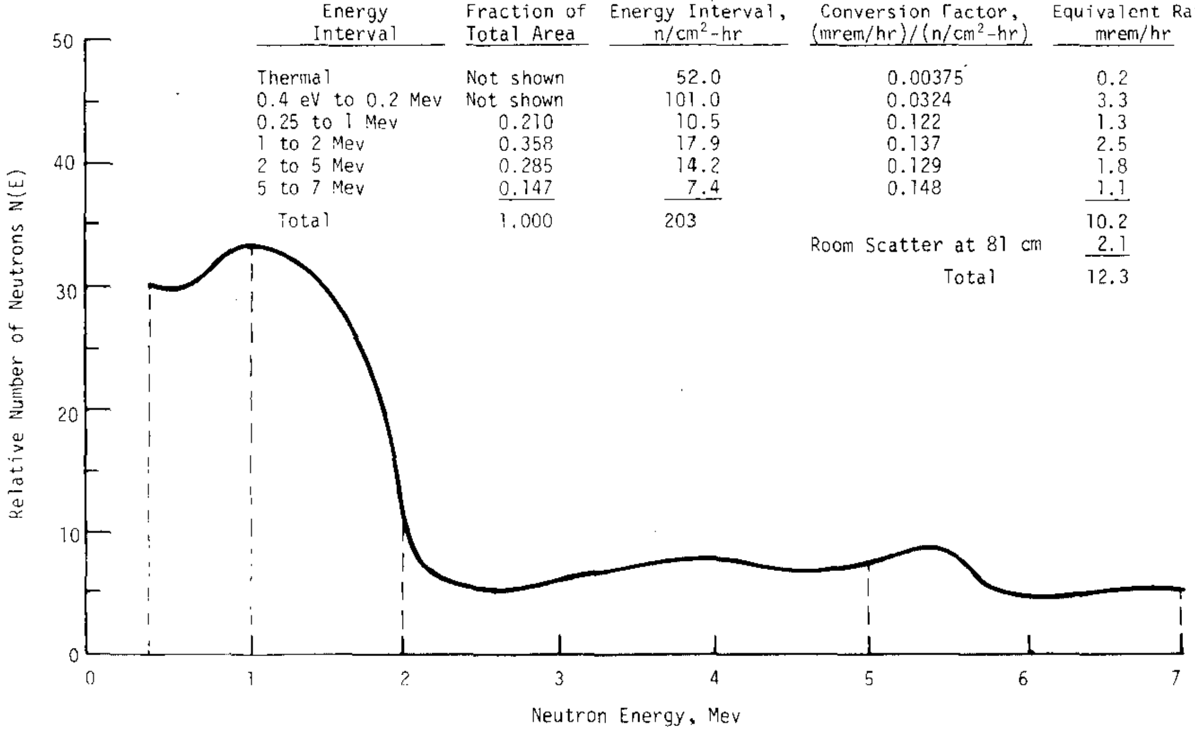


FIG. C-3 10-Ci ^{239}PuBe SOURCE 150 mm INSIDE HEAVY WATER TANK: NEUTRON SPECTRUM AND DOSE EQUIVALENT RATE 81 cm FROM TANK

Derivation of Dose Equivalent Rate from Neutron Spectrum

Energy Interval	Fraction of Total Area	Flux per Energy Interval, n/cm²-hr	Conversion Factor, (mrem/hr)/(n/cm²-hr)	Dose Equivalent Rate, mrem/hr
Thermal	Not shown	22.2	0.00375	0.08
0.4 eV to 0.2 Mev	Not shown	14.3	0.0324	0.46
0.5 eV to 1 Mev	0.110	0.748	0.122	0.09
1 to 2 Mev	0.150	1.02	0.137	0.14
2 to 5 Mev	0.553	3.76	0.129	0.49
5 to 7 Mev	0.186	1.26	0.148	0.19
Total	0.999	43.3		1.45
			Room Scatter at 81 cm	.69
				2.14

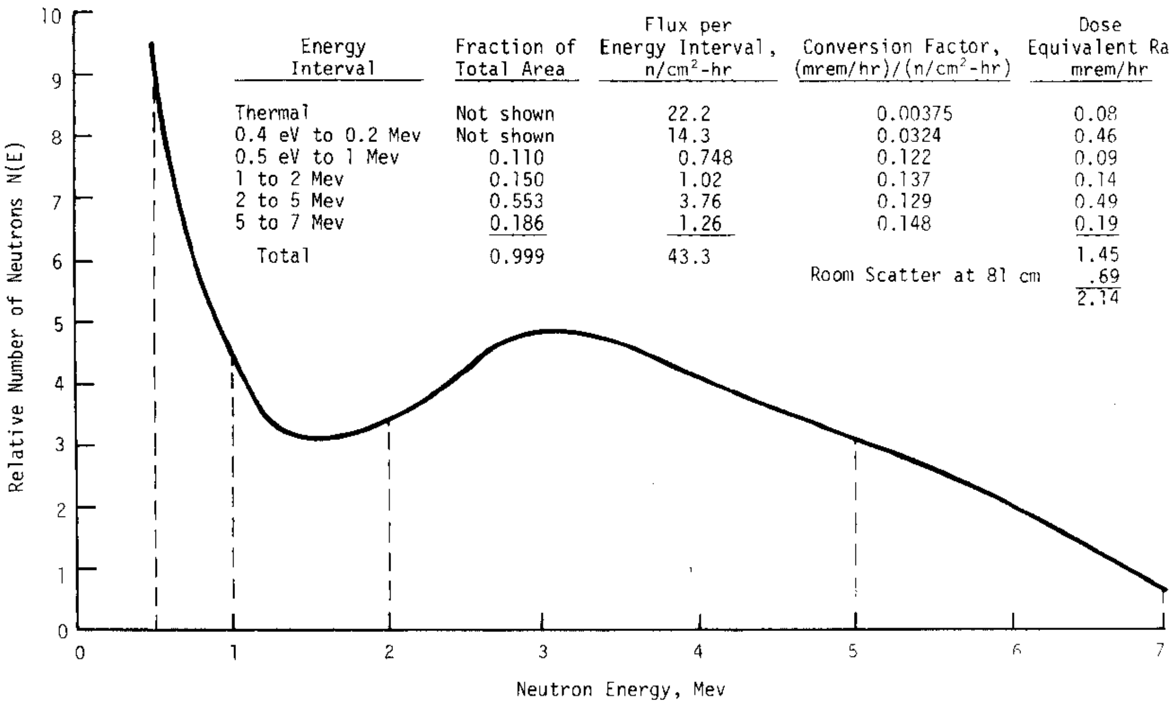


FIG. C-4 10-Ci ^{239}PuBe SOURCE 330 mm INSIDE HEAVY WATER TANK: NEUTRON SPECTRUM AND DOSE EQUIVALENT RATE 81 cm FROM TANK

Derivation of Dose Equivalent Rate from Neutron Spectrum

Energy Interval	Fraction of Total Area	Flux per Energy Interval, n/cm^2-hr	Conversion Factor, (mrem/hr)/(n/cm^2-hr)	Dose Equivalent Rate, mrem/hr
0.3 to 0.5 Mev	0.0206	0.0266×10^5	2.0×10^{-5}	0.053
0.5 to 0.75 Mev	0.0823	0.106×10^5	2.8×10^{-5}	0.297
0.75 to 1 Mev	0.0884	0.114×10^5	3.7×10^{-5}	0.422
1 to 2 Mev	0.331	0.427×10^5	3.8×10^{-5}	1.623
2 to 3 Mev	0.201	0.259×10^5	3.5×10^{-5}	0.906
3 to 6 Mev	0.223	0.288×10^5	3.7×10^{-5}	1.066
6 to 9 Mev	0.053	0.068×10^5	4.3×10^{-5}	0.292
Total	0.999	1.285×10^5		4.7

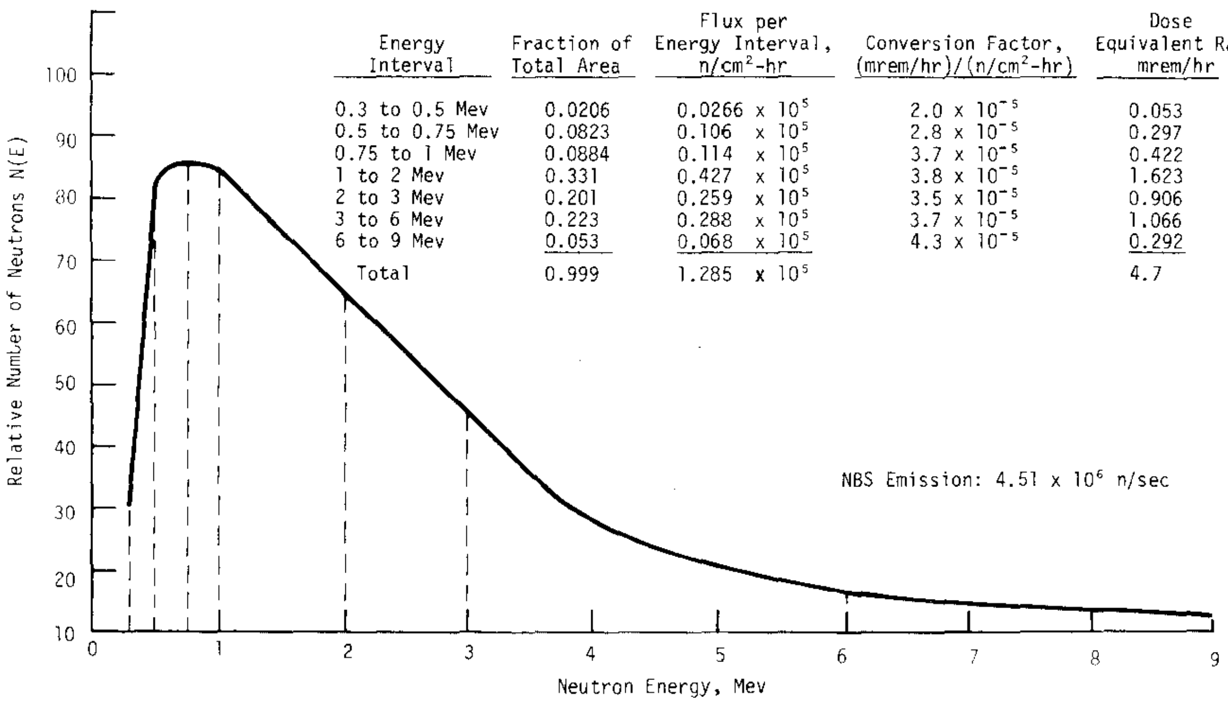


FIG. C-5 3 μg ^{252}Cf SOURCE IN AIR: NEUTRON SPECTRUM AND DOSE EQUIVALENT RATE 100 cm FROM SOURCE

Derivation of Dose Equivalent Rate from Neutron Spectrum

Energy Interval	Fraction of Total Area	Flux per Energy Interval, n/cm^2-hr	Conversion Factor, $(mrem/hr)/(n/cm^2\text{-}hr)$	Dose Equivalent Rate, mrem/hr
0.25 to 0.5 Mev	0.049	0.257×10^5	2.0×10^{-5}	0.514
0.5 to 0.75 Mev	0.060	0.315×10^5	2.8×10^{-5}	0.882
0.75 to 1 Mev	0.053	0.278×10^5	3.4×10^{-5}	0.956
1 to 3 Mev	0.202	1.06×10^5	3.6×10^{-5}	3.82
3 to 6 Mev	0.492	2.58×10^5	3.8×10^{-5}	9.80
6 to 7.8 Mev	0.144	0.756×10^5	4.2×10^{-5}	3.18
Total	1.000	5.246×10^5		19.2

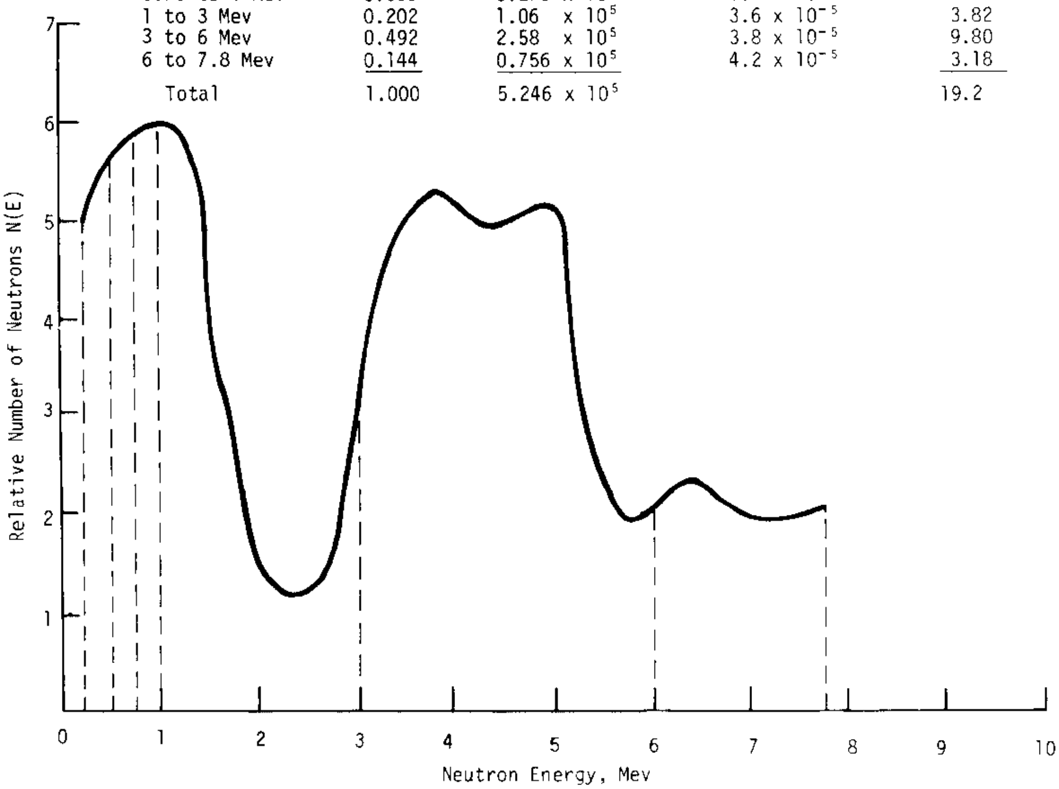
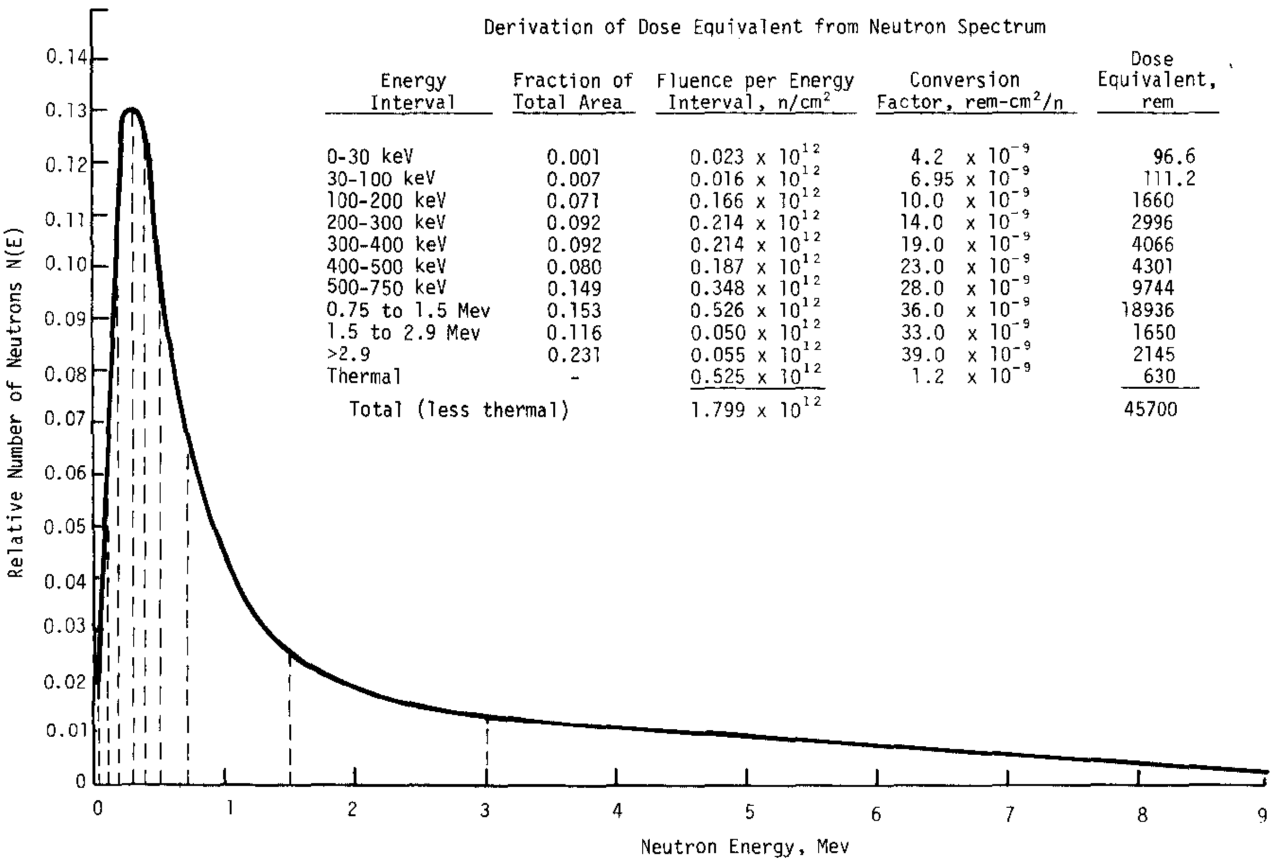


FIG. C-6 10-Ci ^{239}PuBe SOURCE IN AIR: NEUTRON SPECTRUM AND DOSE EQUIVALENT RATE 100 cm FROM SOURCE



Derivation of Dose Equivalent from Neutron Spectrum

Energy Interval	Fraction of Total Area	Fluence per Energy Interval, n/cm^2	Conversion Factor, $rem\text{-}cm^2/n$	Dose Equivalent, rem
0-30 keV	0.001	0.023×10^{12}	4.2×10^{-9}	96.6
30-100 keV	0.007	0.016×10^{12}	6.95×10^{-9}	111.2
100-200 keV	0.071	0.166×10^{12}	10.0×10^{-9}	1660
200-300 keV	0.092	0.214×10^{12}	14.0×10^{-9}	2996
300-400 keV	0.092	0.214×10^{12}	19.0×10^{-9}	4066
400-500 keV	0.080	0.187×10^{12}	23.0×10^{-9}	4301
500-750 keV	0.149	0.348×10^{12}	28.0×10^{-9}	9744
0.75 to 1.5 Mev	0.153	0.526×10^{12}	36.0×10^{-9}	18936
1.5 to 2.9 Mev	0.116	0.050×10^{12}	33.0×10^{-9}	1650
>2.9	0.231	0.055×10^{12}	39.0×10^{-9}	2145
Thermal	-	0.525×10^{12}	1.2×10^{-9}	630
Total (less thermal)		1.799×10^{12}		45700

FIG. C-7 FLATTOP ASSEMBLY: NEUTRON SPECTRUM AND DOSE EQUIVALENT AT 100 cm

Derivation of Dose Equivalent from Neutron Spectrum

Energy Interval	Fraction of Total Area	Fluence per Energy Interval, n/cm^2	Conversion Factor, rem-cm^2/n	Dose Equivalent, rem
0-30 keV	0.0199	0.032×10^9	2.1×10^{-9}	0.067
30-100 keV	0.0512	0.083×10^9	6.4×10^{-9}	0.53
100-200 keV	0.0603	0.098×10^9	12.4×10^{-9}	1.21
200-300 keV	0.0555	0.090×10^9	16.0×10^{-9}	1.44
300-400 keV	0.0555	0.106×10^9	19.0×10^{-9}	2.01
400-500 keV	0.0645	0.104×10^9	23.0×10^{-9}	2.39
500-750 keV	0.1830	0.296×10^9	28.0×10^{-9}	8.28
0.75 to 1.5 Mev	0.1630	0.100×10^9	36.0×10^{-9}	3.60
1.5 to 2.9 Mev	0.1965	0.120×10^9	35.0×10^{-9}	4.2
>2.9 Mev	0.1410	0.086×10^9	42.0×10^{-9}	3.6
Thermal	-	7.65×10^9	1.2×10^{-9}	9.2
Total	1.000	8.765×10^9		36.5

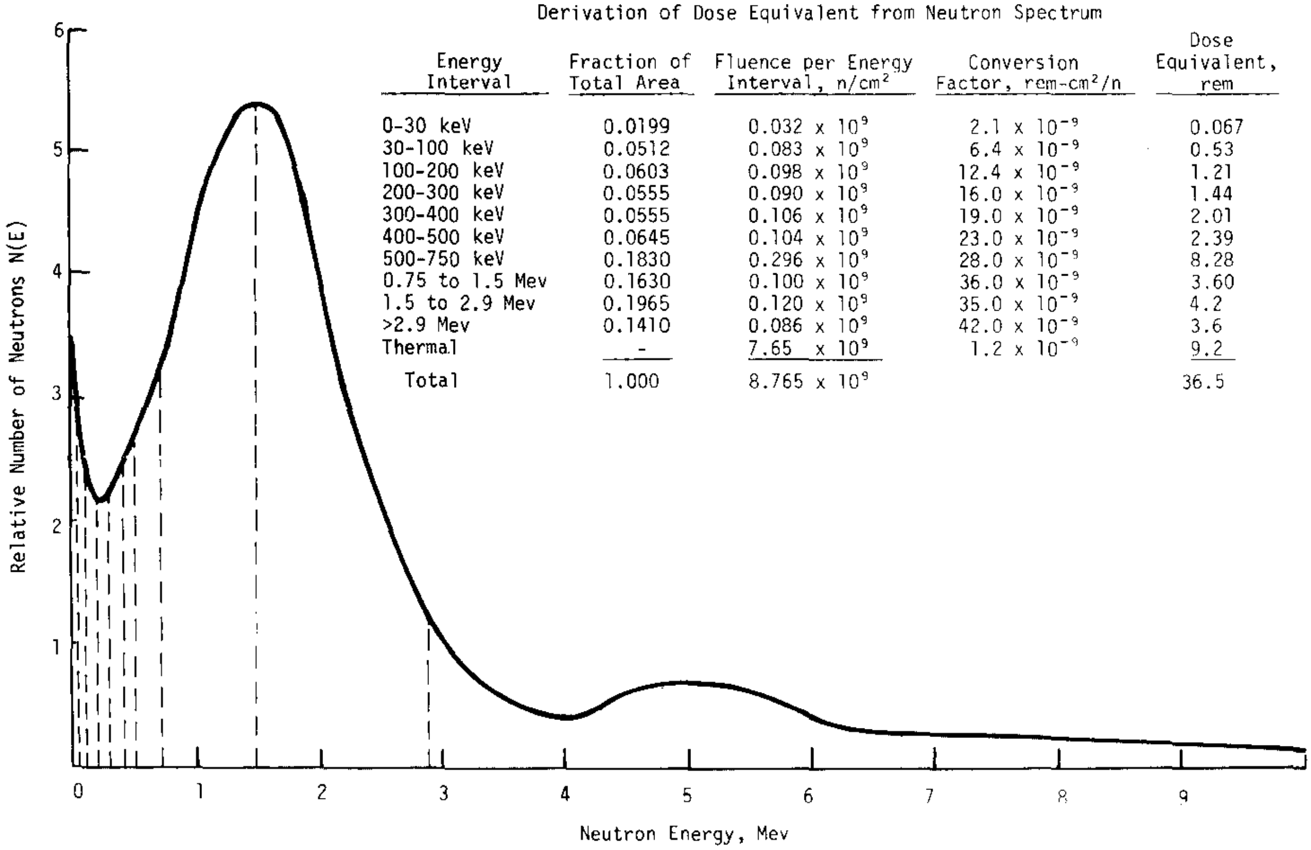


FIG. C-8 PARKA ASSEMBLY: NEUTRON SPECTRUM AND DOSE EQUIVALENT at 95.5 cm

Derivation of Dose Equivalent from Neutron Spectrum

Energy Interval	Fraction of Total Area	Fluence per Energy Interval, n/cm^2	Conversion Factor, $rem\text{-}cm^2/n$	Dose Equivalent, rem
Thermal	-	0.21×10^9	1.2×10^{-9}	0.25
30 to 100 kev	0.004	0.025×10^9	6.8×10^{-9}	0.17
100 to 200 kev	0.025	0.173×10^9	12.0×10^{-9}	2.08
200 to 300 kev	0.023	0.158×10^9	16.0×10^{-9}	2.53
300 to 400 kev	0.032	0.218×10^9	19.0×10^{-9}	4.99
400 to 500 kev	0.040	0.275×10^9	22.0×10^{-9}	6.05
500 to 750 kev	0.139	0.949×10^9	28.0×10^{-9}	26.57
0.75 to 1.5 kev	0.104	0.711×10^9	36.0×10^{-9}	25.6
1.5 to 2.9 Mev	0.254	1.734×10^9	33.0×10^{-9}	57.2
2.9 to 4.0 Mev	0.131	0.897×10^9	36.0×10^{-9}	32.3
4.0 to 6.0 Mev	0.153	1.043×10^9	38.0×10^{-9}	39.6
6.0 to 9.0 Mev	0.083	0.570×10^9	42.0×10^{-9}	23.9
9.0 to 10.0 Mev	0.011	0.076×10^9	45.0×10^{-9}	3.4
Total	0.999	6.83×10^9		224.6

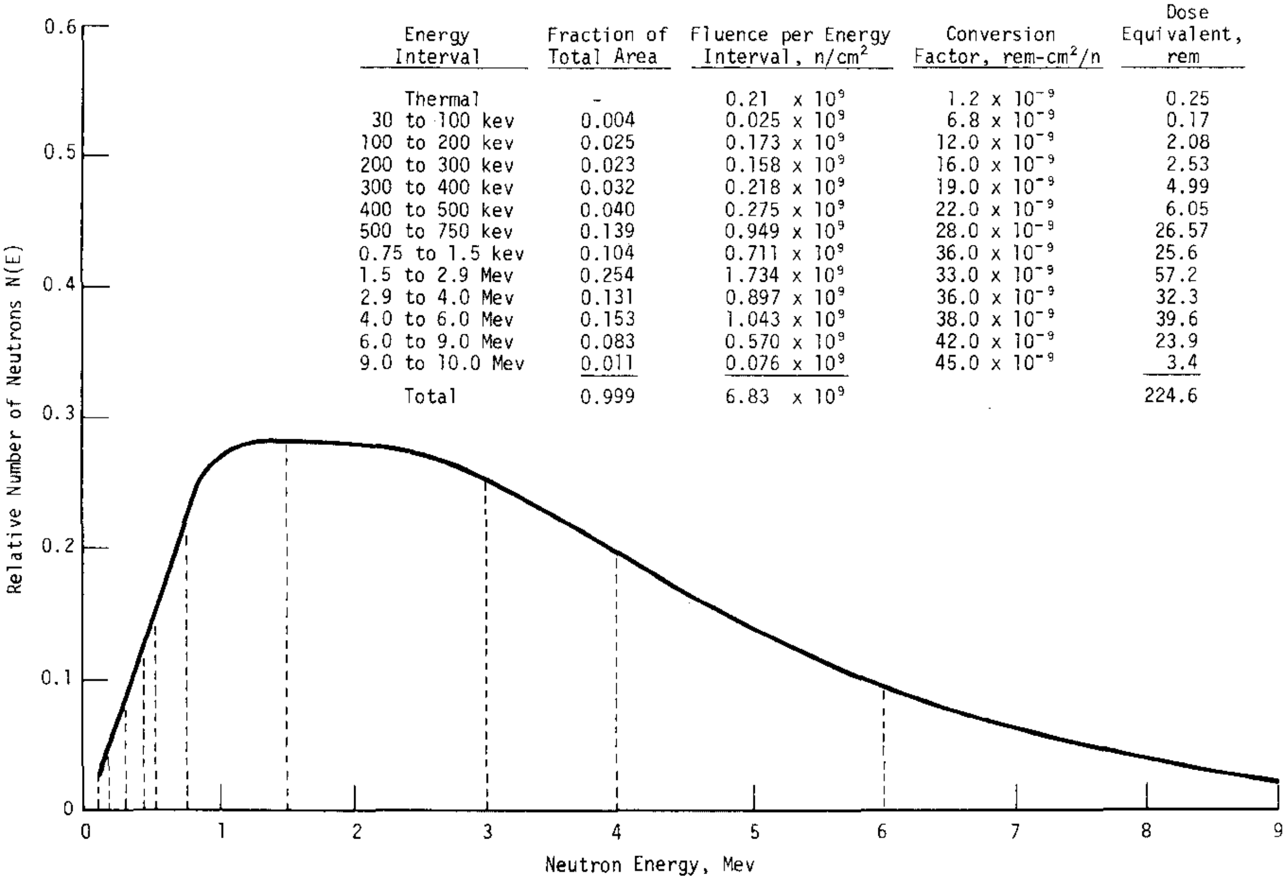
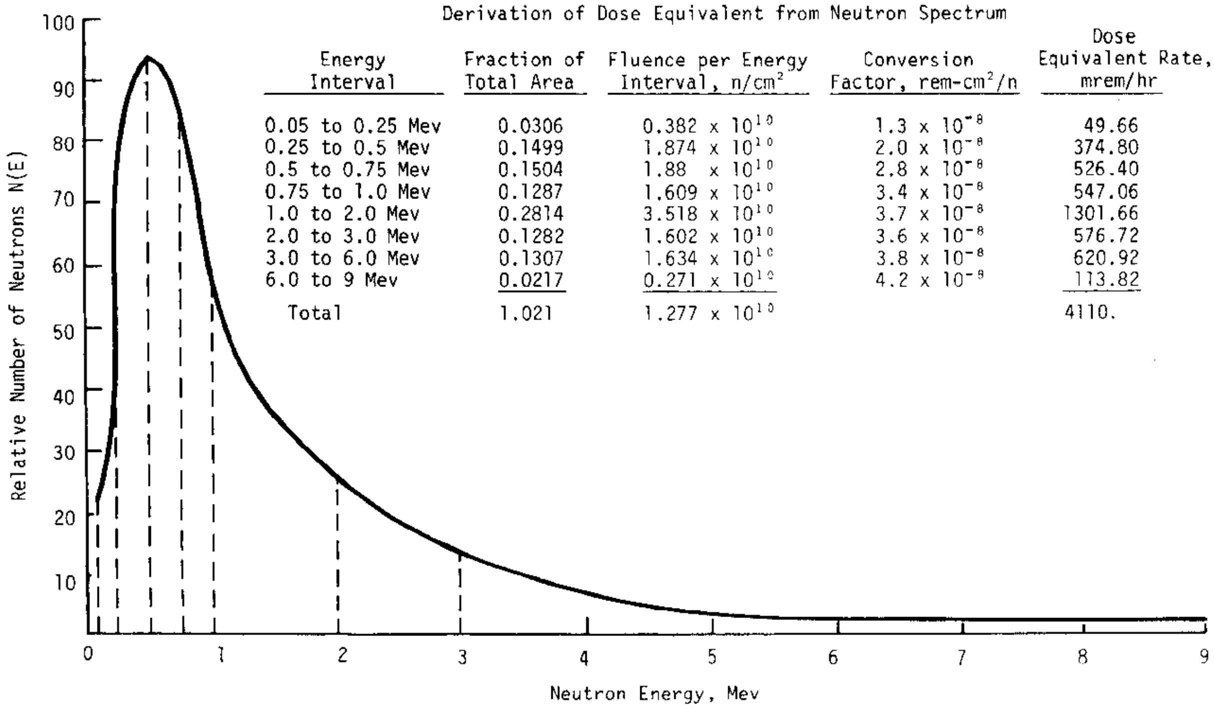


FIG. C-9 JEZEBEL ASSEMBLY: NEUTRON SPECTRUM AND DOSE EQUIVALENT AT 3 m



Derivation of Dose Equivalent from Neutron Spectrum

Energy Interval	Fraction of Total Area	Fluence per Energy Interval, n/cm²	Conversion Factor, rem-cm²/n	Dose Equivalent Rate, mrem/hr
0.05 to 0.25 Mev	0.0306	0.382×10^{10}	1.3×10^{-8}	49.66
0.25 to 0.5 Mev	0.1499	1.874×10^{10}	2.0×10^{-8}	374.80
0.5 to 0.75 Mev	0.1504	1.88×10^{10}	2.8×10^{-8}	526.40
0.75 to 1.0 Mev	0.1287	1.609×10^{10}	3.4×10^{-8}	547.06
1.0 to 2.0 Mev	0.2814	3.518×10^{10}	3.7×10^{-8}	1301.66
2.0 to 3.0 Mev	0.1282	1.602×10^{10}	3.6×10^{-8}	576.72
3.0 to 6.0 Mev	0.1307	1.634×10^{10}	3.8×10^{-8}	620.92
6.0 to 9 Mev	0.0217	0.271×10^{10}	4.2×10^{-8}	113.82
Total	1.021	1.277×10^{10}		4110.

FIG. C-10 HPRR SPECTRUM IN AIR AT 3 m

Derivation of Dose Equivalent from Neutron Spectrum

Energy Interval	Fraction of Total Area	Fluence per Energy Interval, n/cm^2	Conversion Factor, rem-cm^2/n	Dose Equivalent, rem
Thermal	-	1.12×10^{10}	12.0×10^{-10}	13.4
1 to 750 keV	0.5513	0.816×10^{10}	200×10^{-10}	163.2
0.75 to 1.5 Mev	0.1947	0.288×10^{10}	380×10^{-10}	109.4
1.5 to 2.5 Mev	0.1085	0.161×10^{10}	340×10^{-10}	54.7
2.5 to 3.0 Mev	0.1454	0.215×10^{10}	350×10^{-10}	75.3
Total	0.999	1.48×10^{10}		415.

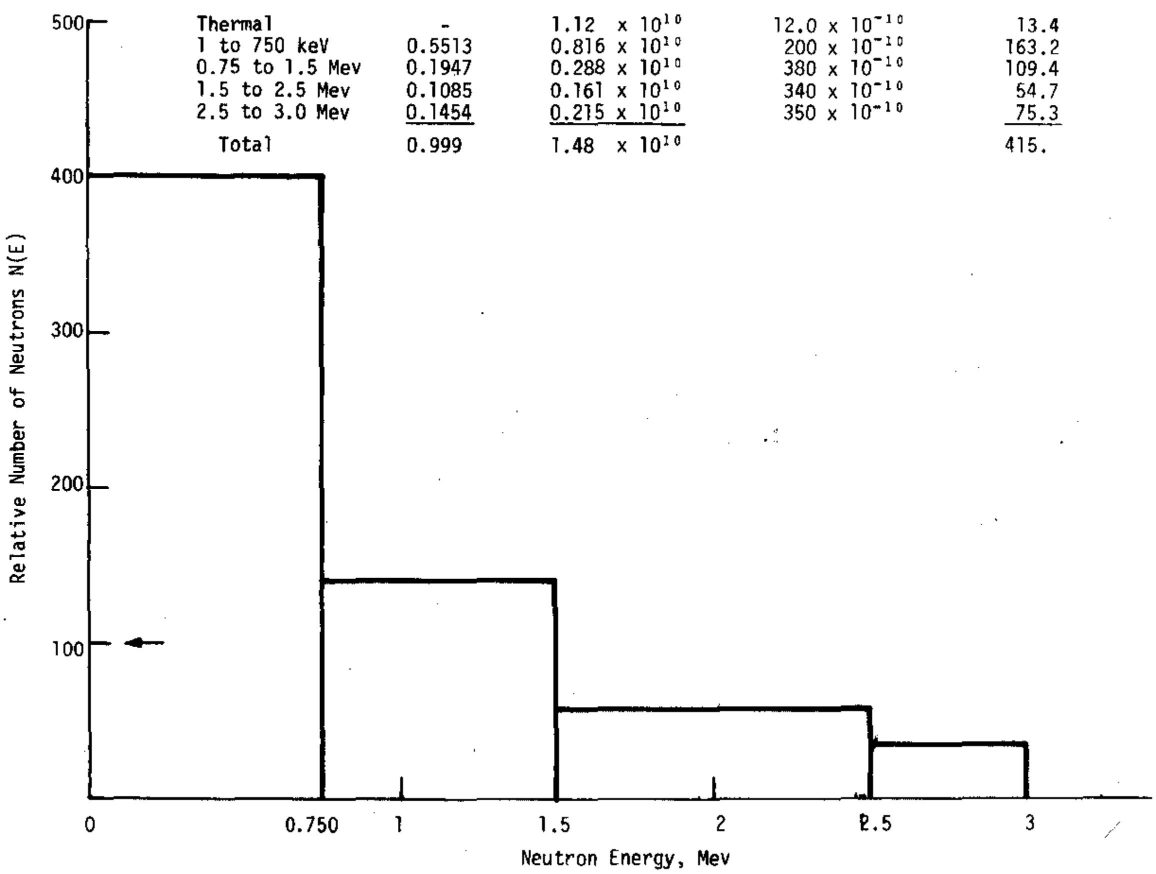
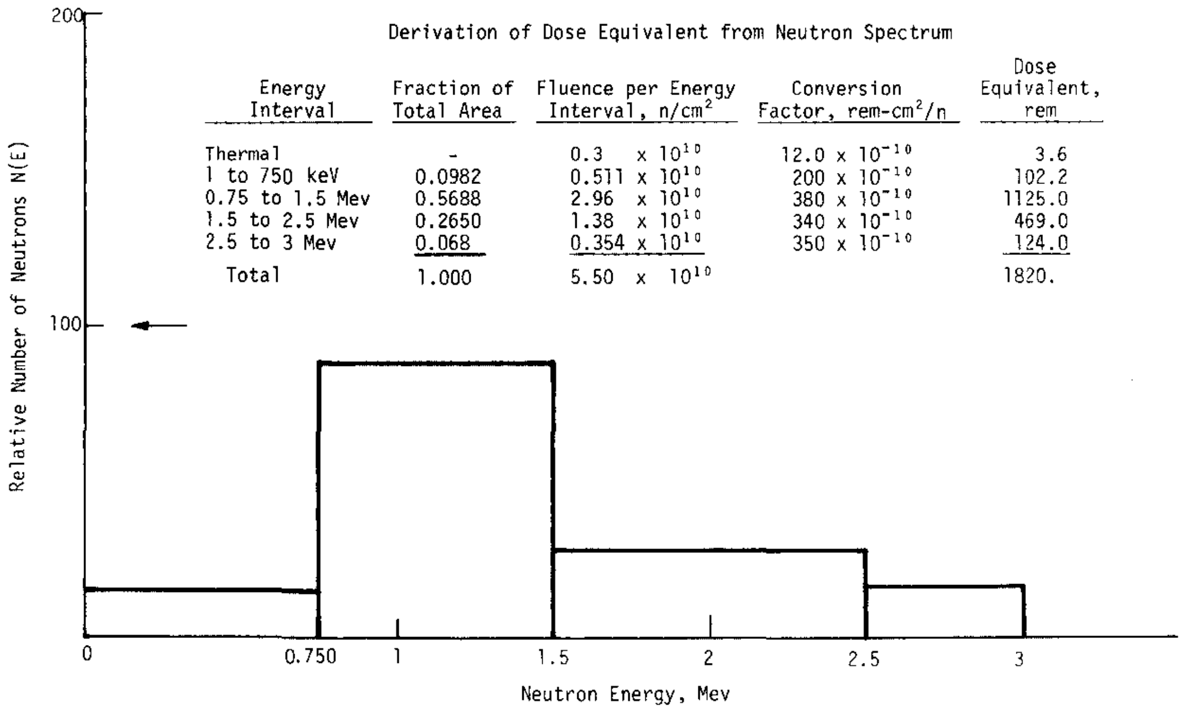


FIG. C-11 HPRR SPECTRUM SHIELDED BY 12 cm OF "LUCITE"



Derivation of Dose Equivalent from Neutron Spectrum

Energy Interval	Fraction of Total Area	Fluence per Energy Interval, n/cm²	Conversion Factor, rem-cm²/n	Dose Equivalent, rem
Thermal	-	0.3×10^{10}	12.0×10^{-10}	3.6
1 to 750 keV	0.0982	0.511×10^{10}	200×10^{-10}	102.2
0.75 to 1.5 Mev	0.5688	2.96×10^{10}	380×10^{-10}	1125.0
1.5 to 2.5 Mev	0.2650	1.38×10^{10}	340×10^{-10}	469.0
2.5 to 3 Mev	0.068	0.354×10^{10}	350×10^{-10}	124.0
Total	1.000	5.50×10^{10}		1820.

FIG. C-12 HPRR SPECTRUM MODERATED BY 13 cm IRON

LIST OF FIGURES

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