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NEW METHODS FOR INTERPRETING NEUTRON DATA
OBTAINED FROM Y-12 NUCLEAR ACCIDENT
DOSIMETER, ORGDP FILM BADGE,
AND HURST THRESHOLD DETECTOR UNIT

AUTHOR:

J. C. Bailey

UNION CARBIDE CORPORATION
NUCLEAR DIVISION
OAK RIDGE GASEOUS DIFFUSION PLANT

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P. O. Box P
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Oak Ridge Gaseous Diffusion Plant
Oak Ridge, Tennessee

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A B S T R A C T

Improved methods have been developed for the interpretation of neutron data obtained from the Y-12 Type Nuclear Accident Dosimeter, the Union Carbide Corporation, Nuclear Division, film badge, and the Hurst threshold detector unit, all of which are in use at the Oak Ridge Gaseous Diffusion Plant. Basic experimental information on which these improved interpretation methods were based was obtained through the use of a new dosimeter developed at ORGDP.

The data-interpretation methods were applied to dosimeter data obtained through the exposure of dosimetry units and phantoms, simulating human exposures, to radiation from the Health Physics Research Reactor at the Oak Ridge National Laboratory under four different shielding configurations, which provided widely different spectra. Neutron dose values determined with a Hurst absolute proportional counter served as the basis of comparison for all units.

With a single exception involving a very heavily shielded configuration--a case which should be recognizable even in an accident situation--dose values derived from film badges on the phantoms facing the reactor were accurate to within 22%. These determinations included badges which were shielded from the reactor by the phantoms. All of the dosimeters yielded dose values which were biased low by about 25% for phantoms turned with a side toward the reactor--a result considered to be consistent with a lower effective dose for a side exposure due to the higher degree of self-shielding of the body for this position as compared to a front or back exposure.

It was concluded that wearing of film badges by all personnel constitutes adequate monitoring for nuclear accidents, but that the continuing use of fixed-location dosimeters in areas where the criticality potentials are greatest provides desirable back-up capabilities.

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NEW METHODS FOR INTERPRETING NEUTRON DATA
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INTRODUCTION

Improved methods have been developed for the interpretation of neutron data obtained from the Y-12 Nuclear Accident Dosimeter,¹ the Union Carbide Corporation, Nuclear Division (UCC-ND) film badge,² and the Hurst threshold detector unit (TDU)*^{3,4} all of which are in use at the Oak Ridge Gaseous Diffusion Plant (ORGDP). Basic experimental information on which these improved interpretation methods were based was obtained through the use of a new dosimeter developed at ORGDP.⁵

Difficulties existing in the interpretation of data obtained with the present devices prior to this development were as follows:

1. No satisfactory method had been formalized for relating the measurements obtained from the Y-12 dosimeter to the exposure of personnel.
2. Quantitative interpretation of film-badge data depended primarily on the concurrent presence in an accident location of fixed-location Hurst threshold detector units (TDU's), the dose to the film badge being related to the dose to the TDU on the basis of the relative activation of the sulfur in the two devices. This method would obviously be invalid if the badge were shielded from the radiation source by the body of the individual wearing the badge.
3. The interpretation of personnel doses on the basis of measurements obtained with the TDU was not fully satisfactory in that the earlier method did not take into account the effects of neutrons in the resonance-energy region in terms of the activation of blood sodium.

This study was directed toward the resolution of these difficulties; the objectives, all of which were attained, were:

1. To formalize a method for determining personnel doses on the basis of spectral data obtained from the Y-12 dosimeter.
2. To develop a method for estimating personnel neutron doses from the film badge independent of an area dosimetry system.
3. To provide a basis for estimating the low-energy neutron fluence, and to establish necessary correction factors for the low-energy neutrons in estimating personnel doses on the basis of TDU data.

* Throughout this report the abbreviation TDU refers to the Hurst Threshold Detector Unit.

The methods and devices described are applicable only to the determination of neutron doses arising from criticality accidents or fission-spectrum energy distributions, and would not be applicable, for example, to exposures to accelerator-produced neutrons.

A limitation of the devices mentioned is that each provides only one neutron fluence* measurement in the resonance-energy region, namely a measurement with cadmium-covered gold, which exhibits a strong resonance at 4.9 ev. The ORGDP dosimeter utilized both cadmium-covered gold and boron-shielded copper, the copper providing an additional estimate of the fluence at the 580-ev resonance value. Thus, measurements obtained with the new dosimeter provided a better basis for estimating the effects of the low-energy part of the neutron spectrum.

All of the dosimeters mentioned, including the film badge, provide fluence measurements from which the dose at the position of the dosimeter itself can be calculated. The primary purpose of these devices, however, is to permit the determination of doses to individuals exposed in a possible accident as an aid to the medical personnel who will be involved in treating the exposure. All the devices, including the film badge, make this possible by providing *relative* fluence measurements in the various energy regions which then permits the calculation of the dose to an individual on the basis of the activation of blood sodium. Thus, a blood-sodium activation measurement is basic to the estimation of personnel doses with any of these devices.

Following the Summary, which is immediately below, the experimental method is described. Then each of the four dosimeters used in this study is taken up in separate sections, each section consisting of a physical description of the neutron-detecting components of the dosimeter followed by a description of the data-interpretation methods. Discussions of the ORGDP dosimeter and TDU are descriptive only, since they are described fully in the referenced reports.

SUMMARY

Experimental evidence regarding the form of the neutron spectrum to be expected from critical assemblies, obtained through use of the ORGDP nuclear accident dosimeter, was used to develop improved data-interpretation methods for the Y-12 Nuclear Accident Dosimeter, the Union Carbide Corporation, Nuclear Division film badge, and the Hurst threshold detector unit. The methods were applied to dosimeter data obtained through the exposure of dosimetry units to radiation from the Health Physics

* Fluence, expressed in units of neutrons per cm^2 , is the total number of neutrons penetrating a spherical surface having a cross-sectional area of 1 cm^2 . Fluence is thus time-integrated flux, expressed in units of neutrons per cm^2 per second.

Research Reactor at the Oak Ridge National Laboratory under four different shielding configurations, which provided widely different spectra. Neutron dose values determined with a Hurst proportional counter under steady-state operation of the reactor were related to the burst exposures of the dosimetry units on the basis of sulfur samples exposed in all bursts and with steady-state reactor operation. Dose values so determined were used as the basis of comparison for all units.

Of primary interest were dose values for simulated personnel exposures, obtained through the use of man-shaped polyethylene phantoms. These were filled with saline solution from which simulated blood-sodium activation analyses were obtained.

The most accurate values for these "personnel" exposures were obtained from film badges on the fronts of phantoms which were facing the reactor. Errors in these cases were less than 10%. Badges on the backs of the phantoms gave errors less than 22% in exposures where the radiation was predominantly unidirectional. With the 23-cm-thick plastic shield, however, an appreciable fraction of the dose was due to radiation which was scattered to the area behind the shield from the surroundings, and in this case the dose was overestimated by 49%. Such a special case should be recognizable, even under accident conditions. The existence of this discrepancy under extremely heavy shielding conditions is not considered to be of major importance, since with shielding of similar effectiveness, an individual would probably not receive a medically serious exposure in an accident situation.

In the case of phantom side exposures, all dosimetry units, including the film badges on the phantoms, gave results which were biased low by about 25%. This was due to the fact that the blood-sodium measurement is the basis for such determinations; as in the case of a human, the area exposed is less and the thickness in the direction of neutron motion is greater in the case of a side exposure than for a front or back exposure. Thus, with the side exposed, the body provides a degree of self-protection with respect to both neutron activation of the blood and radiation damage. Since the primary objective is to provide a dose estimate which will best represent biological damage to the body, it is considered that the lower dose values obtained with side exposures are appropriate.

It was concluded that the film badge provides as reliable dose estimates as any of the fixed-location dosimeters, and that wearing of film badges by all personnel constitutes adequate monitoring for nuclear accidents. The presence of fixed-location dosimeters, however, provides desirable back-up capabilities, and the use of fixed-location dosimeters should be continued in areas where the criticality potentials are greatest.

EXPERIMENTAL METHOD

Exposures of the dosimetry units were carried out at the Dosimetry Application and Research (DOSAR) Facility of the Oak Ridge National Laboratory (ORNL), most of the exposures being made in conjunction with some nuclear-accident dosimetry intercomparison studies involving a number of other installations.

Dosimeters of the various types were exposed directly to radiation from the Health Physics Research Reactor in air and behind shields of iron (12.7 cm thick) and methyl methacrylate plastic of two different thicknesses (23 and 11.4 cm). However, the height of the latter was twice that of the thicker shield to reduce backscatter radiation. This provided exposure to four widely different neutron spectra. In each case, the shields were placed 2 meters from the reactor; the fixed-location dosimeters were placed 3 meters from the reactor and 150 cm above the floor, which was also the height of the reactor. Exposed simultaneously with the dosimetry devices were man-shaped Bomab polyethylene phantoms filled with saline solution to simulate human exposures. The phantoms were either 3 or 4 meters from the reactor, depending largely on space requirements in the intercomparison studies. Some of the phantoms were turned to simulate face-on exposures and others were turned to simulate a side exposure. Badges were placed on both the fronts and backs of the phantoms; the front badges simulated exposure situations for individuals wearing the badges in the normal front position, while the back badges showed the attenuating effects of the body as would be the case for individuals with badges in the normal front position but receiving an exposure from the back.

Dosimeter exposures were made with the reactor in burst operation, with about 5×10^{16} fissions per exposure. Neutron doses at the locations of the dosimeters and phantoms were determined by measurements made with a Hurst absolute proportional counter⁶ under the same shielding conditions, with the reactor in low-power steady-state operation. In these measurements, the spectrum of the proton-recoil pulses detected in the counter chamber was determined with a 100-channel or 256-channel pulse analyzer. This spectrum was extrapolated from the region above the low-energy cutoff bias of the counter to zero, and corrections were applied for the energy lost below the bias.⁷ These measurements were related to the bursts of the tests on the basis of the activities of sulfur detectors placed near the reactor in each case.

DOSIMETERS

ORGDP DOSIMETER

DESCRIPTION

The ORGDP nuclear accident dosimeter utilizes bare and cadmium-covered gold for the measurement of thermal neutrons, cadmium-covered gold and boron-shielded copper for the measurement of resonance neutrons, and

sulfur and boron-shielded indium for the measurement of fast neutrons. To avoid orientation effects resulting from self-shielding, the gold and copper are in the form of spherical shells.

INTERPRETATION OF DATA

Interpretation of the measurements is based upon a spectral model comprising a thermal region from 0 to 0.5 ev; a slowing-down region from 0.5 ev upward, in which the fluence per unit logarithm of energy is a linear function of $\log E$, but weighted by the fraction of neutrons originating in the fission spectrum at energies higher than E ; and a fast region having the shape of a fission spectrum but with energies which may be shifted from those of the fission spectrum by a constant factor. The copper and gold measurements are considered to define the entire slowing-down spectrum, and the shift of the fast portion of the spectrum is determined from the indium-to-sulfur response ratio after these respective values have been corrected for the contribution of the slowing-down spectrum.

Y-12 DOSIMETER

DESCRIPTION

The Y-12 dosimeter consists of a sulfur disc, a pair of cadmium-shielded uranium foils, and a complete film badge. The uranium foils are of different enrichments, one being normal uranium (0.72% U-235) and the other depleted uranium (0.22% U-235). Neutron detectors are incorporated in the film badge and consist of bare and cadmium-covered gold foils, an indium strip, and a small sulfur pellet.

This dosimeter gives five measurements relative to the neutron spectrum--the fluence above the sulfur threshold, fluence above the uranium threshold, fluence above the $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ threshold, a measurement in the resonance-energy region at the 4.9 ev resonance of gold, and thermal fluence.

INTERPRETATION OF DATA

Spectral Model

Since the number of neutron measurements obtained with the dosimeter are not sufficient to define in detail the distribution of neutrons as a function of energy, that is, to completely describe the spectrum, interpretation of the measurements must be based upon some assumed spectral model. Measurements obtained with the Y-12 dosimeter were interpreted in terms of a spectral model in which the spectrum consists of (1) a thermal region, (2) a slowing-down region extending from 0.5 ev to 1 Mev and (3) a fast region with the shape of a fission spectrum which may be shifted in energy from the initial fission energy by some constant factor.

Thermal Fluence. The thermal fluence is determined in the usual way by the cadmium-difference method with the gold foils of the film badge.

Slowing-Down Fluence. Since only one measurement is obtained in the slowing-down part of the spectrum, this part of the spectrum cannot be determined precisely. However, measurements with the ORGDP dosimeter, which provides two measurements in this region, have indicated that the ratio of the fluence per unit logarithmic energy interval at the 580-eV resonance of copper to that at the 4.9-eV resonance of gold varied from about 1.5 to 3 in the spectra studied. As an approximation, therefore a standard ratio of 2 for the fluence at these energies was chosen for the model. The slowing-down spectrum was considered to terminate abruptly at the 1 MeV upper limit.

Numerical integration of the first-collision dose curve weighted by this spectrum yielded a value for the integral dose of 13.1×10^{-9} rad with unit fluence per unit logarithmic energy interval at 4.9 eV. Integration of the capture-probability curve weighted by this same spectrum yielded a capture integral of 12.2 with unit fluence per unit logarithmic energy interval at the gold resonance.

The results for the cadmium-covered gold foil in the film badge is expressed in terms of the thermal fluence which would induce the same activity in a similar bare gold foil. This value is designated N_{Au} . A conversion factor is needed to convert this number to fluence per unit $\ln E$ interval, this fluence value being designated $\phi(Au)$. A direct comparison between the ORGDP dosimeter measurements, made with gold spheres calibrated directly in terms of fluence per unit $\ln E$ interval, was obtained in the intercomparison tests. The results are given in Table 1.

Table 1. Comparison of Gold Fluence Values of
Y-12 and ORGDP Dosimeters
(Fluence in units of 10^{10} neutrons/cm²)

Test. No.	ORGDP $\phi(Au)$	Y-12 N_{Au}	$\frac{\phi(Au)}{N_{Au}}$
1	0.143	0.454	0.315
2	0.034	0.109	0.312
3	0.097	0.307	0.316
Mean	--	--	0.314

The ratio $\phi(Au)/N_{Au}$ of 0.314 found in these tests was in very good agreement with the gold foil calibration factor of 0.297 derived in K-1618⁸ for relating the response of flat gold foils in an isotropic flux to the fluence per unit logarithmic interval in a $1/e$ flux distribution.

Applying this experimental factor to the Y-12 dosimeter foils, we obtain the conversion factors for dose and capture integrals to yield the following relationships:

$$\text{Dose for slowing-down neutrons} = 4.1 \times 10^{-9} \times N_{\text{Au}}$$

$$\text{Capture integral} = 3.8 \times N_{\text{Au}}.$$

Fast Fluence. Measurements relating to the fast portion of the spectrum are the fast neutron reactions with sulfur, uranium, and indium. Since the uranium and indium reactions are related to neutrons in almost the same energy range, these detectors may be paired separately with the sulfur measurement to provide two evaluations of the fast spectrum.

The uranium foils are paired to permit a correction factor to be applied to the measurements in order to correct for the fissions induced in the U-235 isotope, the desired measurement being the fissions induced in U-238, or more precisely, the apparent neutron fluence above the U-238 threshold. For this calculation, the two foils were counted in the same way as the uranium foils of the TDU, and the apparent fluence was calculated for each one by the equation:

$$\phi = \frac{C}{W \times N(t)},$$

where

ϕ = apparent fluence,

C = count rate in counts per minute at time t,

W = foil weight in grams, and

$N(t)$ = count rate per gram for exposure to 10^{10} neutrons/cm²

above the U-238 threshold.

The U-238 fluence, $\phi(U)$, was then calculated from these two apparent fluence values by the formula derived as follows: Assume the curve of response versus enrichment to be a straight line for any given exposure. Then the pure U-238 response is:

$$\phi(U) = (D) - \frac{\phi(N) - \phi(D)}{0.72 - 0.22} \times 0.22$$

Here $\phi(D)$ and $\phi(N)$ are the apparent fluences indicated by the depleted and normal uranium foils, respectively.

For use in the equations developed here, the uranium fluence values were multiplied by an empirical factor of 1.15. This adjustment was based on the results of experiments regarding the testing of the spectral model used with the ORGDP dosimeter. These experiments are described in reference 5. Briefly, measurements obtained with the ORGDP dosimeter were compared with TDU measurements which, as is described in a later section, included uranium foil measurements. Values predicted for the fission foils on the basis of the indium measurements of the ORGDP dosimeter and the spectral model used with it were greater by 15% than the experimental values obtained with the TDU fission foils. In the calculation of doses, however, this bias was counterbalanced by the methods of weighting the spectra, the doses in both cases being quite accurate. Since the spectral model used with the Y-12 dosimeter makes use of the shifted-fission-spectrum concept, similar to that used with the ORGDP dosimeter, the uranium fluence values, $\phi(U)$, were multiplied by the factor 1.15 to provide values which would be consistent with the use of the spectral model for dose calculations. The new value is designated $\phi'(U)$; this is the value used in the equations given below for the calculation of neutron dose.

As noted at the beginning of this section, the 4.5-hour half-life indium resulting from the $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ reaction with fast neutrons may also be used to evaluate the fast portion of the spectrum. Precautions must be taken to avoid or to correct for activity of the 54-minute half-life $^{116\text{m}}\text{In}$ resulting from the $^{115}\text{In}(n,\gamma)^{116\text{m}}\text{In}$ in which results primarily with thermal and resonance-energy neutrons. Although this isotope produces a 0.406 MeV gamma ray which cannot be resolved fully from the 0.335 MeV gamma ray of $^{115\text{m}}\text{In}$ with the sodium-iodide scintillation spectrometer used in these studies, such an instrument can be used to determine the activity of interest in three ways.

1. A measurement under the 0.335 MeV gamma peak may be made directly some 20 hours after exposure, this delay being sufficient to allow the 54-minute half-life activity to decay to a negligible level.
2. Two measurements may be made under the 0.335 MeV gamma peak at different times after partial decay of the 54-minute activity--say 4 to 6 hours after exposure and again some 2 hours later. From these two measurements, the count rates in the peak region of the 54-minute activity and the 3.5-hour activity may be calculated separately. Equations for this calculation are derived as follows:

$$C(t_1) = C_1 e^{-\lambda_1 t_1} + C_2 e^{-\lambda_2 t_1};$$

$$C(t_2) = C_1 e^{-\lambda_1 t_2} + C_2 e^{-\lambda_2 t_2};$$

where.

$C(t_1)$ = Total count rate (in this case in the 0.335 MeV peak region) at time of first count,

$C(t_2)$ = Total count rate at time of second count,

C_1 = Count rate at zero time for isotope 1,

C_2 = Count rate at zero time for isotope 2,

t_1 = Time of first count,

t_2 = Time of second count,

λ_1 and λ_2 = Decay constants of isotopes 1 and 2, respectively
($\lambda = 0.693/\text{half life}$).

From these equations,

$$C_1 = [C(t_1)e^{\lambda_2 t_1} - C(t_2)e^{\lambda_2 t_2}] / [e^{(\lambda_2 - \lambda_1)t_1} - e^{(\lambda_2 - \lambda_1)t_2}]$$

$$C_2 = [C(t_2) - C_1 e^{-\lambda_1 t_2}] e^{\lambda_2 t_2}$$

3. The counts in the channels covering the 0.335 MeV gamma can be corrected for the contribution of the 54-minute indium on the basis of the count under one of the gamma peaks of higher energy. From a sample in which the activity is essentially all due to the 54-minute indium (which may be readily obtained by thermal neutron irradiation of an indium sample), the ratio of the counts in the 1.274 MeV peak to the counts in the channels corresponding to the 0.335 MeV peak may be determined. This ratio may then be applied to the 1.274 MeV peak to determine the correction for the 0.335 MeV peak value for samples in which both activities are present.

In the present experiments, the measurement of 0.335 MeV gamma activity was made some 20 hours after exposure as described for the first method above. The indium foil was removed from the film badge of the dosimeter and placed directly on the face of the 3 x 3 in. sodium iodide crystal of a 400-channel gamma spectrometer. The instrument was adjusted so that the 0.662 MeV gamma photopeak of Cs-137 was in channel 166, and the 0.335 MeV peak of ^{115m}In was summed over channels 77 through 95. A calibration factor to relate the resulting peak sum to fluence was obtained by comparing the count rate on a 1-in.-diam foil supported 3 cm from the crystal face, for which the calibration was known, to the count rate on an indium strip exposed at the same time. The resulting conversion factor is $1.80 \times 10^6 \times (\text{n/cm}^2)/(\text{c/min/g})$. That is,

$$\phi(\text{In}) = 1.80 \times 10^6 \times C,$$

where C is the photopeak count rate in counts per minute, corrected for decay to time of exposure.

Calculation of Doses

The basic dose calculations involve integration over the entire neutron spectrum of the product of neutron dose as a function of energy and the fluence per energy increment. In order to facilitate the application of the spectral models already described, the necessary integrations were carried out numerically for the fission spectrum and for fission-shaped spectra shifted from the original energy by selected factors. Similarly, numerical integrations were performed for capture probability and for the response curves of uranium, indium, and sulfur for these same spectra. Since the uranium-to-sulfur fluence ratio and the indium-to-sulfur fluence ratio are both characteristic of the degree of spectral shift which has occurred, either of these ratios provides an index as to the values of the dose and capture integrals to be used for dose computations. Therefore, the dose and capture integrals are shown graphically in Figs. 1 and 2, respectively, as functions of the uranium-to-sulfur fluence ratio (the uranium fluence being modified by the factor 1.15 as specified earlier) and in Figs. 2 and 3, respectively, as functions of the indium-to-sulfur fluence ratio.

Neutron Dose at Dosimeter. The value for the sulfur fluence may be paired with either the uranium or indium fluence value for dose calculations. The neutron dose at the position of the dosimeter is given by either of the two equations:

$$\text{Dose} = 0.028 \times 10^{-9} \phi_{\text{th}} + 4.1 \times 10^{-9} N_{\text{Au}} + \frac{\text{Fast Dose Integral}}{\phi'(U)} \times \phi'(U).$$

$$\text{Dose} = 0.028 \times 10^{-9} \phi_{\text{th}} + 4.1 \times 10^{-9} N_{\text{Au}} + \frac{\text{Fast Dose Integral}}{\phi(\text{In})} \times \phi(\text{In}).$$

Here the factor 0.028×10^{-9} is the dose in rads per unit thermal fluence, 4.1×10^{-9} is the average dose per unit gold fluence (as defined on page 12), and the ratios

$$\frac{\text{Fast Dose Integral}}{\phi'(U)} \quad \text{and} \quad \frac{\text{Fast Dose Integral}}{\phi(\text{In})}$$

are obtained from Figs. 1 and 3, respectively, on the basis of the uranium-to-sulfur or the indium-to-sulfur ratios.

Personnel Dose. The dose to an individual (or phantom) is related to the sodium activation of the blood on the basis of the total capture integral as computed from the spectral information obtained from the dosimeter. The equations for the total capture integral (T), relating to the use of uranium and indium, respectively, are:

$$T = 0.18 \phi_{\text{th}} + 3.8 N_{\text{Au}} + \frac{\text{Fast Capture Integral}}{\phi'(U)} \phi'(U).$$

$$T = 0.18 \phi_{th} + 3.8 N_{Au} + \frac{\text{Fast Capture Integral}}{\phi(In)} \times \phi(In).$$

Values for the ratios $\frac{\text{Fast Capture Integral}}{\phi'(U)}$ and $\frac{\text{Fast Capture Integral}}{\phi(In)}$

are taken from Figs. 2 and 4, respectively, again on the basis of the uranium-to-sulfur or indium-to-sulfur fluence ratio.

The equation for dose to a phantom (or an individual) exposed to neutrons with the same spectral distribution as was the dosimeter is:

$$D_n = 9.5 \times 10^8 \times \frac{\text{Dose to Dosimeter}}{\text{Total Capture Integral}} \times K$$

where

D_n = first collision neutron dose in rads;

K = disintegrations per second per milliliter blood serum with a sodium concentration of 3.2 mg/ml.

FILM BADGE

Description

As indicated under the description of the Y-12 dosimeter, the neutron detectors of the film badge are sulfur, indium, and bare and cadmium-covered gold. The badge itself is described in detail in reference 2.

Data Interpretation

The equations for the basic dose calculations are the same as given for the Y-12 dosimeter where indium and sulfur are considered to be the fast neutron detectors. These equations are:

Neutron dose to badge:

$$\text{Dose} = 0.028 \times 10^{-9} \phi_{th} + 4.1 \times 10^{-9} N_{Au} + \frac{\text{Fast Dose Integral}}{\phi(In)} \phi(In),$$

where the value of the ratio $\frac{\text{Fast Dose Integral}}{\phi(In)}$ is taken from Fig. 3.

Total capture integral:

$$T = 0.18 \phi_{th} + 3.8 N_{Au} + \frac{\text{Fast Capture Integral}}{\phi(In)} \times \phi(In),$$

where the ratio $\frac{\text{Fast Capture Integral}}{\phi(I_n)}$ is taken from Fig. 4.

Dose to individual based on blood sodium analysis:

$$D_n = 9.5 \times 10^8 \times \frac{\text{Dose to Badge}}{\text{Total Capture Integral}} \times K$$

where K is the disintegrations per second per milliliter of blood serum with a sodium concentration of 3.2 mg/ml.

Since the neutron field to which a badge is exposed is affected by the presence of the phantom or individual, a correction must be applied to the results obtained from the equations above to account for these effects. Changes in dose values obtained from a badge on a phantom as compared to a dosimeter exposed in air may be summarized as follows:

1. A badge on the side of a phantom which is toward the reactor will tend to indicate a dose above the free air dose since it is exposed to the full intensity of incident neutrons, to which will be added the effects of neutrons reflected from the phantom.
2. The reflected neutrons will be reduced in energy by scattering effects, and therefore the spectrum indicated by the badge will be of lower energy than the spectrum of the incident neutrons. Since a blood-sodium interpretation of dose does not depend upon the absolute fluence but only upon the relative distribution of the neutrons in the spectrum, a blood-sodium interpretation based upon this softened spectrum will tend to be lower than the free air dose, and therefore also lower than the dose indicated by the badge.
3. The dose to a badge on the side of a phantom turned away from the reactor will tend to be low because of the obvious shielding effects of the phantom itself.
4. The spectrum at the back of a phantom will tend to be softer than that at the front because of attenuation and moderation of incident neutrons which penetrate the phantom, plus the fact that neutrons scattered from the surrounding, and which are therefore reduced energy, will also be incident on the badge and will constitute a relatively large fraction of the total incident fluence. Again, because of these spectral-softening effects, the dose as interpreted from blood sodium will tend to be low.

The above explanations are supported by the experimental results. As will be discussed more fully under the section describing the results of these experiments, the ratio of badge dose to dose determined from sodium measurements was a valid indicator of orientation; badges on a side of the phantom toward the reactor were uniformly higher than

the dose determined from sodium activation measurements, whereas badges shielded from the reactor by the phantom were uniformly lower.

Empirical correction factors to be applied to the dose values determined on the basis of the blood-sodium analyses were developed from the experimental results. The correction factors are:

1. If the data indicate that the badge was on a side of the phantom or individual which was turned toward the reactor (badge result higher than the blood-sodium result) multiply the blood-sodium dose determination by the factor 1.25.
2. If the data indicate that the badge was shielded from the reactor by the phantom (badge dose lower than the blood-sodium result) multiply the blood-sodium determination by the factor 1.50.

Dose values determined with these correction factors were considered to represent the best estimate of the dose to the phantom.

HURST THRESHOLD DETECTOR UNIT (TDU)

Description

With the Hurst threshold detector unit (TDU), thermal neutrons are measured by the difference in the activities of bare and cadmium-covered gold foils; fast neutrons in various energy regions are measured by the activity produced through the $S^{32}(n,p)P^{32}$ reaction in a sulfur disc and by the gamma emission from the products of nuclear fissions produced in plutonium-239, neptunium-237, and uranium-238 fission foils. These fission foils are contained in an isotopically enriched boron-10 shell which effectively shields them from slow neutrons. Use of the boron shell permits plutonium, which fissions with both fast and slow neutrons, to be used as a detector of only fast neutrons. It also eliminates slow neutron fission in the small amounts of uranium-235 which are present in the uranium foil; and eliminates the production, through slow-neutron capture in the neptunium foil, of the gamma-emitter, neptunium-238.

Interpretation of Data

Interpretation of the activities induced in the sulfur and fission foils of the TDU is based upon the concept of idealized step-function cross sections which satisfactorily approximate the actual activation cross sections of these materials. That is, each detecting material is considered to have, in effect, a cross section which is constant for neutrons with energies above an effective threshold energy and zero for neutrons with energies below this value.

Activities induced in the threshold foils are interpreted in terms of the number of neutrons per square centimeter, or fluence, above each of the effective threshold energies. The difference between the fluence values indicated by any two of the foils thus represents the fluence in the energy range between the two thresholds. Thermal fluence is determined from the difference in activities produced in bare and cadmium-covered gold foils, the bare foil being activated by both thermal and resonance neutrons and the cadmium-covered foil by resonance neutrons only.

The first-collision neutron dose at the position of the TDU is computed by multiplying the flux in each energy group by an appropriate factor, representing the average dose per neutron per square centimeter for that energy group, and adding the products. The equation is:

$$D_n = \left[0.028 \phi_{th} + 1.4(\phi_{Pu} - \phi_{Np}) + 2.4(\phi_{Np} - \phi_U) + 3.0(\phi_U - \phi_S) + 3.8 \phi_S \right] \times 10^{-9}$$

where the dose is in tissue rads, and the symbol ϕ refers to the number of neutrons per square centimeter as obtained from each of the detectors. This may be rewritten as:

$$D_n = \left[0.028 \phi_{th} + 1.4 \phi_{Pu} + 1.0 \phi_{Np} + 0.6 \phi_U + 0.8 \phi_S \right] \times 10^{-9} \text{ rads.}$$

The coefficient for the sulfur fluence in the above equation is applicable to a 2.5 MeV effective threshold for sulfur. In this report, sulfur fluences are given as relating to a 2.9 MeV effective threshold these fluences being lower than those for a 2.5 MeV threshold by the factor 0.771. Therefore, the equation for dose applicable to the expression of sulfur fluence in terms of a 2.9 MeV threshold is:

$$D_n = \left[0.028 \phi_{th} + 1.4 \phi_{Pu} + 1.0 \phi_{Np} + 0.6 \phi_U + 1.04 \phi_S \right] \times 10^{-9} \text{ rads.}$$

The calculation for phantom or personnel dose based on blood sodium activation as given in ORNL-2748, Part A, is:

$$D_n = 9.5 \times 10^8 \frac{\int_0^{\infty} D_0(E) \phi(E) dE}{\int_0^{\infty} \epsilon(E) \phi(E) dE}$$

where D_n is the neutron dose in rads, $\phi(E)dE$ is the number of neutrons per cm^2 between E and $E + dE$, $D_0(E)$ is the first collision dose per unit neutron fluence at energy E , $\epsilon(E)$ is the probability that neutrons of energy E will be captured in the phantom as thermal neutrons, and K is the sodium activity in disintegrations per sec (corrected back to time of exposure) per cm^3 of blood serum. This expression applies to the case where the concentration of sodium in serum is 3.2 mg/cm^3 .

The integral in the denominator of the above equation is referred to throughout the remainder of this report as the capture integral. As given in ORNL-2748, Part A, the capture integral is represented by the expression:

$$\text{Capture Integral} = 0.18 \phi_{th} + 0.5 \phi_{Pu}.$$

It is noted that this expression accounts for activation of sodium in the body induced only by thermal neutrons and neutrons in the plutonium response region; thus neutrons in the broad resonance energy region are not included.

Corrections to TDU Phantom-Dose Calculations

A correction to the TDU data was made to account for neutrons below the effective plutonium threshold on the basis of the same spectral model used with the Y-12 dosimeters and badges. The amount by which the capture integral would be underestimated by the expression for the capture integral used in ORNL-2748-A was calculated by numerical integration of the spectrum over the plutonium cross-section curve, and this amount was found to be equal to $5.65 \phi(Au)$. Using the ratio between $\phi(Au)$ and N_{Au} derived previously, this amounts to $1.77 N_{Au}$.

The total capture integral as determined with a TDU was therefore considered to be:

$$\text{Capture Integral} = 0.18 \phi_{th} + 0.5 \phi_{Pu} + 1.77 N_{Au}.$$

Thus, the equation for the neutron dose to a phantom is:

$$D_n = 9.5 \times 10^8 \times \frac{\text{Dose to Dosimeter}}{0.18 \phi_{th} + 0.5 \phi_{Pu} + 1.77 N_{Au}} \times K.$$

RESULTS

Error ranges for the various dosimetry units are shown in Table 2, while the detailed measurements and doses calculated from them are given in Tables 3 through 8, which are at the end of the report. Percentage errors are also shown graphically for the various dosimetry determinations in Figs. 5 through 8. The basis for evaluating these errors was the measurements made with the Hurst absolute proportional counter with correction for loss of energy under the bias.^{6,7}

Table 2. Accuracy Range for Dosimeter Values
(Percent Error)

Dosimeter Type	Dosimeter Dose	Phantom Doses	
		Front Exposure	Side Exposure
ORGDP	-27; +25	-8; +19	-39; -12
TDU	-26; -3	-20; +42	-43; -18
Y-12			
Uranium and Sulfur	-42; +44	+34; -18	-37; -31
Indium and Sulfur	-25; +48	-10; +31	-37; -23
Film Badges			
Front*	-	-8; +7	-32; -2
Back	-	-14; +49**	-41; -1

* Front badges were those on phantom surfaces turned toward the reactor. Back badges were shielded from the reactor by the phantom.

** With the exception of this 49% error, discussed in the text, the range was from -14% to +22%.

Of the three fixed-location dosimeters used in these experiments, the Hurst threshold detector unit gave the most accurate results with regard to the dose to the dosimeter itself, the error range being from -26% to -3%. With regard to the determination of personnel doses, however, which is the primary purpose of the fixed-location dosimeter, the ORGDP dosimeter yielded the most accurate values. It is noted that the TDU provides the most complete data on the fast portion of the neutron spectrum, which produces most of the neutron dose. On the other hand, the ORGDP dosimeter provides the most complete data on the resonance energy region, from which a large fraction of the neutron activation of blood sodium may derive, the final evaluation of dose being better with the ORGDP dosimeter.

The most accurate values for these "personnel" exposures, however, were obtained from film badges on the fronts of phantoms which were facing the reactor. Errors in these cases were less than 10%. Badges on the backs of the phantoms gave errors less than 22% in exposures where the radiation was predominantly unidirectional. With the 23-cm thick plastic shield, however, an appreciable fraction of the dose was

due to radiation which was scattered to the area behind the shield from the surroundings, and in this case the dose was overestimated by 49%. Such a special case should be recognizable, even under accident conditions. The existence of this discrepancy under extremely heavy shielding conditions is not considered to be of major importance, since with shielding of similar effectiveness an individual would probably not receive a medically serious exposure in an accident situation.

In the case of phantom side exposures, all dosimetry units, including the film badges on the phantoms, gave results which were biased low by about 25%. This was due to the fact that the blood-sodium measurement is the basis for such determinations; as in the case of a human, the area exposed is less and the thickness in the direction of neutron motion is greater in the case of a side exposure than for a front or back exposure. Thus, with the side exposed, the body provides a degree of self-protection with respect to both neutron activation of the blood and radiation damage. Since the primary objective is to provide a dose estimate which will best represent biological damage to the body, it is considered that the lower dose values obtained with side exposures are appropriate.

CONCLUSIONS

It was concluded that the film badge provides as reliable dose estimates as any of the fixed-location dosimeters, and that wearing of film badges by all personnel constitutes adequate monitoring for nuclear accidents. The presence of fixed-location dosimeters however, provides desirable back-up capabilities, and the use of fixed-location dosimeters should be continued in areas where the criticality potentials are greatest.

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Table 3
ORGDP DOSIMETER DOSE VALUES

Shield	No. Fissions	Fluence (units 10^{10} n/cm ²)					$\frac{\phi(\text{Cu})}{\phi(\text{Au})}$	$\frac{\phi(\text{S})_{\text{res}}}{\phi(\text{Au})}$	$\frac{\phi(\text{In})_{\text{res}}}{\phi(\text{Au})}$	$\phi(\text{S})_{\text{fast}}$ (10^{10} n/cm ²)	$\phi(\text{In})_{\text{fast}}$ (10^{10} n/cm ²)	$\frac{\phi(\text{In})_{\text{f}}}{\phi(\text{S})_{\text{f}}}$	(Dose Int.) _{res}	(Dose Int.) _f	Dose, rad	Proportional Counter Dose, rad	% Error
		ϕ Thermal	$\phi(\text{Au})$	$\phi(\text{Cu})$	$\phi(\text{In})$	$\phi(\text{S})$							$\frac{\phi(\text{Au})}{10^{-9} \text{ rad}}$ n/cm ²	$\frac{\phi(\text{In})}{10^{-9} \text{ rad}}$ n/cm ²			
None	6.36×10^{16}	0.535	0.143	0.384	4.23	1.51	2.69	0.525	2.20	1.43	3.91	2.73	27.0	5.68	261	265	- 2
	7.52×10^{16}	0.697	0.172	0.458	4.98	1.74	2.66	0.511	2.20	1.65	4.60	2.79	26.7	5.7	308	313	- 2
	"	0.896	0.191	0.495	5.09	1.81	2.59	0.499	2.10	1.72	4.69	2.73	25.9	5.7	317	313	+ 1
Iron 12.7cm	7.09×10^{16}	0.378	0.097	0.282	1.14	0.268	2.91	0.59	2.5	0.211	0.90	4.26	30.0	6.75	90	112	-20
	"	0.441	0.103	0.288	1.09	0.270	2.80	0.55	2.35	0.213	0.848	3.98	28.6	6.45	84	119	-29
	"	0.505	0.104	0.274	1.06	0.239	2.63	0.51	2.19	0.186	0.802	4.31	26.3	6.70	81	111	-27
Plastic 23 cm	5.38×10^{16}	0.350	0.034	0.050	0.190	0.067	1.47	0.215	0.90	0.060	0.159	2.65	11.7	5.61	12.9	16.3	-21
	5.46×10^{16}	0.377	0.035	0.051	0.170	0.084	1.46	0.21	0.90	0.077	0.138	1.79	12.5	5.2	11.6	13.1	-11.5
	"	0.441	0.051	0.079	0.250	0.105	1.55	0.23	1.00	0.093	0.199	2.14	15.0	5.4	18.4	20.8	-11.5
Plastic 11.4cm	3.99×10^{16}	0.431	0.082	0.096	0.521	0.198	1.17	0.13	0.60	0.187	0.472	2.52	7.7	5.5	35.1	28.1	+25
	"	0.433	0.081	0.095	0.523	0.205	1.17	0.13	0.60	0.194	0.474	2.44	7.7	5.5	35.1	28.4	+24

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Table 4
PHANTOM DOSE VALUES DERIVED FROM ORGDP DOSIMETER RESULTS

Shield	No. Fissions	Phantom Dist., m	Orien- tation*	(Capt. Int.) _{res} ϕ (Au)	(Capt. Int.) _f ϕ (In)	Total Capture Integral (units 10^{10})	Dose per Unit Sodium Activity, rad dis/sec/3.2mg	^{24}Na Activity, dis/sec 3.2 mg	Phantom Dose, rad	Propor- tional Counter Dose, rad	% Error
None	6.36×10^{16}	4	F	17.9	1.02	6.65	3.73	47.4	177	157	+13
	7.52×10^{16}	3	F	17.9	1.03	8.33	3.51	89.4	314	313	+0.3
	"	3	S	17.3	1.02	8.66	3.48	55.0	191	313	-39
Iron 12.7cm	7.09×10^{16}	4	F	19.5	1.40	3.22	2.64	24.6	64.9	54.6	+19
	"	3	F	18.9	1.34	3.16	2.53	44.1	112	119	-5.9
	"	3	S	17.6	1.42	3.43	2.55	30.3	68.2	111	-39
Plastic 23 cm	5.38×10^{16}	3	F	9.1	1.0	0.531	2.31	6.68	15.4	16.3	-6
	5.46×10^{16}	3	F	9.1	0.77	0.493	2.24	5.40	12.1	13.1	-8
	"	3	S	9.8	0.87	0.797	2.19	7.23	15.8	20.8	-24
Plastic 11.4cm	3.99×10^{16}	3	F	7.0	0.97	1.16	2.87	11.78	33.4	28.1	+19
	"	3	S	7.0	0.95	1.14	2.93	8.58	25.1	28.4	-12

*F - Facing Reactor; S - Side Toward Reactor

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Table 5
HURST THRESHOLD DETECTOR UNIT MEASUREMENTS

Shield	No. Fissions	Phantom		Fluence (units of 10^{10} n/cm ²)						Dosimeter Dose, rad	Total Capture Integral (units of 10^{10})	Dose per Unit Sodium Activity, rad per dis/sec/3.2mg	²⁴ Na Activity dis/sec/3.2mg	Phantom Dose, rad	Proportional Counter Dose Values, rad		% Error	
		Dist. m	Orien-tation*	N _{Thermal}	N(Au)	φ (Pu)	φ (Np)	φ (U)	φ (S)						Dosimeter	Phantom	Dosimeter	Phantom
None	6.36×10^{16}	4	F	0.566	0.454	9.34	7.12	3.44	1.51	238	5.57	4.06	47.4	192	265	157	-10	+22
	7.52×10^{16}	3	F	0.653	0.592	10.9	8.65	4.07	1.74	282	6.62	4.05	89.4	362	313	313	-10	+16
	"	3	S	0.879	0.647	11.1	8.67	4.12	1.81	286	6.85	3.97	55.0	218	313	313	-8.6	-30
Iron 12.7cm	7.09×10^{16}	4	F	0.362	0.307	5.12	2.81	0.811	0.066	105	3.17	3.15	24.6	77.5	112	54.6	-6	+42
	"	3	F	0.340	0.300	4.96	2.75	0.771	0.270	104	3.07	3.22	44.1	142	119	119	-13	+19
	"	3	S	0.348	0.429	5.18	2.72	0.740	0.239	107	3.41	2.98	30.3	90	111	111	-4	-19
Plastic 23 cm	5.38×10^{16}	3	F	0.398	0.109	0.532	0.320	0.154	0.250	12.3	0.529	2.21	6.68	14.8	16.3	16.3	-25	-9
	5.46×10^{16}	3	F	0.391	0.107	0.410	0.227	0.141	0.084	9.73	0.465	1.99	5.40	10.7	13.1	13.1	-26	-18
	"	3	S	0.454	0.155	0.898	0.513	0.222	0.105	20.1	0.805	2.37	7.23	17.1	20.8	20.8	-3	-18
Plastic 11.4cm	3.99×10^{16}	3	F	1.38	0.289	0.981	0.693	0.398	0.198	25.1	1.25	1.91	11.78	22.5	28.1	28.1	-11	-20
	"	3	S	1.40	0.288	0.948	0.686	0.397	0.205	24.6	1.24	1.88	8.58	16.1	28.4	28.4	-13	-43

*F - Facing Reactor; S - Side Toward Reactor

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Table 6
Y-12 DOSIMETER DOSE VALUES BASED ON URANIUM AND SULFUR

Shield	No. Fissions	Phantom		Fluence (units of 10^{10} n/cm ²)					(Dose Int.) _f	Dosimeter Dose, rad	(Capt. Int.) _f	Total Capt. Integral (units 10^{10})	Dose per Unit Sodium Activity, rad per dis/sec/3.2 mg	²⁴ Na Activity dis/sec 3.2 mg	Phantom Dose, rad	Proportional Counter Dose Values, rad		% Error	
		Dist. m	Orien-tation*	N _{Thermal}	N(Au)	ϕ' (U)**	ϕ (S)	$\frac{\phi' (U)}{\phi (S)}$								Dosimeter	Phantom	Dosimeter	Phantom
None	6.36×10^{16}	4	F	0.566	0.454	3.94	1.51	2.61	6.0	255	1.11	6.20	3.91	47.4	185	265	157	- 4	+18
	7.52×10^{16}	3	F	0.653	0.592	4.72	1.60	2.95	6.85	348	1.32	8.60	3.85	89.4	344	313	313	+11	+10
	"	3	S	0.879	0.647	4.86	1.73	2.81	6.50	343	1.22	8.55	3.82	55.0	210	313	313	+10	-33
Iron 12.7cm	7.09×10^{16}	4	F	0.362	0.307	0.937	0.250	3.75	8.6	93	1.86	2.98	2.97	24.6	73.1	112	54.6	-17	+34
	"	3	F	0.340	0.300	0.789	0.254	3.11	7.20	69.2	1.42	2.32	2.83	44.1	125	119	119	-42	+5.0
	"	3	S	0.348	0.429	0.789	0.250	3.16	7.30	75.3	1.45	2.84	2.52	30.3	76.3	111	111	-32	-31
Plastic 23 cm	5.38×10^{16}	3	F	0.398	0.109	0.177	0.0664	2.67	6.15	15.5	1.13	0.684	2.15	6.68	14.4	16.3	16.3	- 5	-12
	5.46×10^{16}	3	S	0.454	0.155	0.229	0.107	2.14	4.90	17.7	0.80	0.85	1.98	7.23	14.3	20.8	20.8	-15	-31
Plastic 11.4 cm	3.99×10^{16}	3	F	1.38	0.289	0.485	0.226	2.15	5.0	36	0.80	1.73	1.95	11.8	23	28.1	28.1	+28	-18
	"	3	S	1.40	0.288	0.520	0.218	2.39	5.5	41	0.95	1.84	2.10	8.58	18	28.4	28.4	+44	-37

*F - Facing Reactor; S - Side Toward Reactor

** $\phi' (U) = 1.15 \times \phi (U)$

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Table 7
Y-12 DOSIMETER DOSE VALUES BASED ON INDIUM AND SULFUR

Shield	No. Fissions	Phantom		Fluence (units of 10^{10} n/cm ²)						Dosimeter Dose, rad	Capt. Int. ϕ (In)	Total Capt. Integral (units 10^{10})	Dose per Unit Sodium Activity, rad per dis/sec/3.2mg	²⁴ Na Activity, dis/sec 3.2mg	Phantom Dose, rad	Proportional Counter Dose Values, rad		% Error	
		Dist. m	Orien- tation*	N _{Thermal}	N(Au)	ϕ (In)	ϕ (S)	ϕ (In) ϕ (S)	Dose Int. ϕ (In)							Dosimeter	Phantom	Dosimeter	Phantom
None	6.36×10^{16}	4	F	0.566	0.454	4.31	1.51	2.85	5.75	267	1.05	6.35	3.99	47.4	189	265	157	+ 1	+20
	7.52×10^{16}	3	F	0.653	0.592	4.81	1.60	3.00	5.81	304	1.09	7.61	3.79	89.4	339	313	313	- 3	+ 8
	"	3	S	0.879	0.647	4.95	1.73	2.86	5.75	311	1.05	7.81	3.78	55.0	208	313	313	- 1	-34
Iron 12.7cm	7.09×10^{16}	4	F	0.362	0.307	1.29	0.250	5.16	7.25	106	1.62	3.33	3.02	24.6	74.3	112	54.6	- 5	+31
	"	3	F	0.340	0.300	1.13	0.254	4.44	6.82	90	1.44	2.83	2.99	44.1	132	119	119	-25	+11
	"	3	S	0.348	0.429	1.08	0.250	4.32	6.52	88	1.42	3.23	2.61	30.3	79	111	111	-21	-29
Plastic 23cm	5.38×10^{16}	3	F	0.398	0.109	0.196	0.066	2.95	5.80	16.0	1.08	0.696	2.18	6.68	14.6	16.3	16.3	- 2	-10
	5.46×10^{16}	3	F	0.391	0.107	0.227	0.084	2.70	5.65	17.3	1.01	0.706	2.33	5.40	12.6	13.1	13.1	+32	- 4
	"	3	S	0.454	0.155	0.274	0.107	2.56	5.58	21.8	0.97	0.937	2.21	7.23	16.0	20.8	20.8	+ 5	-23
Plastic 11.4cm	3.99×10^{16}	3	F	1.38	0.289	0.540	0.226	2.39	5.45	41.7	0.92	1.84	2.15	11.8	25.4	28.1	28.1	+48	-10
	"	3	S	1.40	0.288	0.504	0.218	2.31	5.42	39.5	0.91	1.81	2.07	8.58	17.8	28.4	28.4	+39	-37

*F - Facing Reactor; S - Side Toward Reactor

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Table 8
FILM BADGE DOSE RESULTS

Shield	No. Fissions	Phantom		Badge** Position	Fluence (units 10 ¹⁰ n/cm ²)				ϕ (In) ϕ (S)	(Dose Int.) _f ϕ (In)	Badge Dose, rad	(Capt. Int.) _f ϕ (In) (units 10 ¹⁰)	Total Capt. Int. (units 10 ¹⁰)	Dose per Unit Sodium Activity rad dis/sec/3.2 mg	²⁴ Na Activity dis/sec 3.2 mg	Dose Based on Blood Sodium, rad	Badge Dose Blood Sodium Dose	Dose Est., rad	Prop. Counter Dose Value, rad	% Error
		Dist. m.	Orien- tation*		N _{Thermal}	N(Au)	ϕ (In)	ϕ (S)												
None	6.36 x 10 ¹⁶	4	F	F	2.30	0.844	2.55	0.835	3.05	5.85	184	1.10	6.43	2.72	47.4	129	1.43	163	157	+ 4
	"	4	F	B	0.505	0.162	0.277	0.094	2.95	5.8	22.8	1.08	1.00	2.17	47.4	103	0.22	155	157	- 1
	7.52 x 10 ¹⁶	3	F	F	4.11	1.56	5.33	1.63	3.27	6.0	385	1.16	12.9	2.84	89.4	253	1.52	319	313	+ 2
	"	3	F	F	3.93	1.64	5.22	1.61	3.24	6.0	382	1.15	12.9	2.81	89.4	251	1.52	316	313	+ 1
	"	3	F	B	1.33	0.372	0.612	0.197	3.11	5.9	51.7	1.12	2.34	2.10	89.4	188	0.28	282	313	- 10
	"	3	F	B	1.48	0.471	0.828	0.217	3.82	6.4	72.7	1.29	3.12	2.21	89.4	198	0.37	297	313	- 5
	"	3	S	F	2.75	1.22	5.54	1.36	4.07	6.6	416	1.35	12.6	3.13	55.0	173	2.4	218	313	- 30
	"	3	S	B	1.10	0.455	0.792	0.172	4.60	6.9	73.6	1.48	3.10	2.26	55.0	124	0.6	186	313	- 41
	"	3	S	F	2.52	0.977	5.38	1.81	2.97	5.8	353	1.08	9.98	3.36	55.0	185	1.9	233	313	- 26
	"	3	S	B	0.98	0.384	0.864	0.126	6.86	8.3	87.7	1.98	3.35	2.49	55.0	137	0.6	205	313	- 35
Iron 12.7 cm	7.09 x 10 ¹⁶	4	F	F	1.18	0.500	0.545	0.143	3.81	6.4	55.7	1.29	2.82	1.88	24.7	46.4	1.20	58.5	54.6	+ 7
	"	4	F	B	0.459	0.157	0.098	0.026	3.77	6.35	12.8	1.28	0.805	1.51	24.7	37.3	0.34	56.0	54.6	+ 3
	7.09 x 10 ¹⁶	3	F	F	2.0	0.867	1.33	0.268	4.96	7.2	132	1.57	5.74	2.18	44.1	96.3	1.37	122	119	+ 3
	"	3	F	F	1.98	0.934	1.37	0.256	5.33	7.4	140	2.08	6.76	1.97	44.1	86.8	1.61	110	119	- 8
	"	3	F	B	0.512	0.151	0.110	0.037	2.97	5.8	12.7	1.08	0.78	1.55	44.1	68.2	0.19	102	119	- 14
	"	3	F	B	0.651	0.235	0.221	0.055	4.02	6.5	24.2	1.34	1.31	1.75	44.1	77.4	0.31	116	119	- 3
	"	3	S	F	1.32	0.530	0.995	0.209	4.76	7.0	91.7	1.52	3.76	2.32	30.3	70.2	1.31	88.4	111	- 20
	"	3	S	B	0.568	0.199	0.209	0.040	5.23	7.3	23.6	2.07	1.29	1.74	30.3	52.7	0.45	79.2	111	- 29
	"	3	S	F	1.35	0.555	0.928	0.209	4.44	6.8	86.2	1.97	4.18	1.96	30.3	59.4	1.45	75.0	111	- 32
	"	3	S	B	0.699	0.258	0.244	0.044	5.55	7.5	29.1	2.11	1.62	1.71	30.3	51.7	0.56	77.6	111	- 30
Plastic 23 cm	5.38 x 10 ¹⁶	3	F	F	0.742	0.152	0.193	0.080	2.41	5.5	17.1	0.94	0.893	1.82	6.68	12.2	1.40	15.4	16.3	- 5
	"	3	F	B	0.219	0.076	0.073	0.033	2.21	5.4	7.1	0.88	0.392	1.72	6.68	11.5	0.61	17.3	16.3	+ 6
	5.46 x 10 ¹⁶	3	F	F	0.704	0.134	0.196	0.066	2.97	5.8	17.1	1.08	0.848	1.92	5.40	10.3	1.66	13.0	13.1	- 1
	"	3	F	F	0.623	0.122	0.167	0.069	2.42	5.5	14.4	0.94	0.733	1.87	5.40	10.1	1.43	12.7	13.1	- 3
	"	3	F	B	0.184	0.060	0.135	0.039	3.46	6.1	10.7	1.20	0.423	2.40	5.40	13.0	0.82	19.5	13.1	+ 49
	"	3	S	F	0.480	0.092	0.175	0.027	6.48	8.1	18.1	1.90	0.769	2.24	7.23	16.2	1.12	20.4	20.8	- 2
	"	3	S	B	0.230	0.068	0.108	0.018	6.00	7.8	9.33	1.80	0.494	1.79	7.23	13.0	0.72	20.5	20.8	- 1
	"	3	S	F	0.689	0.177	0.229	0.064	3.58	6.2	21.6	1.23	1.08	1.90	7.23	13.7	1.57	17.3	20.8	- 17
	"	3	S	B	0.321	0.135	0.095	0.045	2.11	5.3	10.7	0.85	0.652	1.56	7.23	11.3	0.95	17.0	20.8	- 18

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Table 8
FILM BADGE DOSE RESULTS
(Continued)

Shield	No. Fissions	Phantom		Badge** Position	Fluence (units 10^{10} n/cm ²)				ϕ (In) ϕ (S)	(Dose Int.) _f ϕ (In)	Badge Dose, rad	(Capt. Int.) _f ϕ (In) (units 10^{10})	Total Capt. Int. (units 10^{10})	Dose per Unit Sodium Activity rad dis/sec/3.2 mg	²⁴ Na Activity dis/sec 3.2 mg	Dose Based on Blood Sodium, rad	Badge Dose Blood Sodium Dose rad	Dose Est., rad	Prop. Counter Dose Value, rad	% Error
		Dist. m.	Orien- tation*		N _{Thermal}	N(Au)	ϕ (In)	ϕ (S)												
Plastic 11.4 cm	3.99×10^{16}	3	F	F	2.33	0.357	0.612	0.201	3.04	5.8	50.8	1.10	2.45	1.97	11.78	23.2	2.2	29.2	28.1	+ 4
	"	3	F	F	2.63	0.428	0.612	0.161	3.80	6.4	57.5	1.29	2.89	1.89	11.78	22.3	2.6	28.1	28.1	0
	"	3	F	F	2.06	0.344	0.522	0.106	4.92	7.1	51.7	1.56	2.49	1.97	11.78	23.2	2.2	29.2	28.1	+ 4
	"	3	F	F	2.53	0.397	0.576	0.202	2.85	5.75	50.1	1.05	2.57	1.85	11.78	21.8	2.3	27.5	28.1	- 2
	"	3	F	B	0.353	0.078	0.054	0.024	2.25	5.4	6.2	0.89	0.41	1.44	11.78	17.0	0.36	25.5	28.1	- 9
	"	3	F	B	0.357	0.074	0.108	0.034	3.18	6.0	9.6	1.13	0.47	1.94	11.78	22.9	0.42	34.4	28.1	+ 22
	"	3	F	B	0.439	0.098	0.108	0.036	3.00	5.8	10.4	1.09	0.57	1.73	11.78	20.4	0.51	30.6	28.1	+ 9
	"	3	F	B	0.359	0.084	0.108	0.035	3.09	5.9	9.9	1.11	0.50	1.88	11.78	22.1	0.45	33.2	28.1	+ 18
	"	3	S	F	1.26	0.222	0.450	0.136	3.31	6.0	36.5	1.17	1.60	2.17	8.58	18.6	2.0	23.4	28.4	- 28
	"	3	S	B	0.473	0.105	0.090	0.021	4.29	6.7	10.5	1.40	0.61	1.64	8.58	14.1	0.74	21.2	28.4	- 26
	"	3	S	F	1.59	0.266	0.450	0.192	2.34	5.45	35.9	0.91	1.71	1.99	8.58	17.1	2.1	21.5	28.4	- 24
	"	3	S	B	0.412	0.085	0.090	0.010	9.00	9.5	12.2	2.40	0.61	1.90	8.58	16.3	0.75	24.5	28.4	- 14
	"	3	S	B	0.502	0.092	0.090	0.022	4.09	6.6	9.9	1.35	0.56	1.68	8.58	14.4	0.69	21.6	28.4	- 24
	"	3	S	F	1.56	0.290	0.540	0.194	2.78	5.7	43.1	1.03	1.94	2.11	8.58	18.1	2.4	22.8	28.4	- 20
	"	3	S	F	1.60	0.286	0.504	0.160	3.15	5.9	41.9	1.11	1.93	2.06	8.58	17.7	2.4	22.3	28.4	- 21

*Phantom orientation: F-facing reactor; S-side toward reactor.

**Badge Position: F - Badge on a surface turned toward reactor.

B - Badge on a back surface; i. e., shielded from the reactor by the phantom.

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Figure 1
DOSE INTEGRAL FOR SHIFTED FISSION SPECTRA

Figure 2
CAPTURE INTEGRAL FOR SHIFTED FISSION SPECTRUM

Figure 3
DOSE INTEGRAL FOR SHIFTED FISSION SPECTRA

Figure 4
CAPTURE INTEGRAL FOR SHIFTED FISSION SPECTRA

Figure 5
ACCURACY OF DOSIMETER DOSE VALUES

Figure 6
ACCURACY OF FRONT-FACING PHANTOM DOSES
DERIVED FROM FIXED-LOCATION DOSIMETERS

Figure 7
ACCURACY OF
SIDE-EXPOSURE PHANTOM DOSES DERIVED FROM
FIXED-LOCATION DOSIMETERS

Figure 8
ACCURACY OF PHANTOM DOSES
DERIVED FROM FILM BADGES