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DP-1006

AEC RESEARCH AND DEVELOPMENT REPORT

SAVANNAH RIVER PLANT CRITICALITY DOSIMETRY SYSTEM

C. N. WRIGHT
J. E. HOY
W. F. SPLICHAL, JR.

DUPONT
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Savannah River Laboratory

Aiken, South Carolina

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Author(s) C. N. Wright, J. E. Hoy, and W. F. Splichal
2852

Title Savannah River Plant Criticality Dosimetry System

Division Radiological and Environmental Sciences

Indicate known prior art _____

Approval (Introduction, Summary, Cover Letter) C. H. Lee Section Director — A. A. Johnson 8/17/65 Laboratory Director

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Title: **Savannah River Plant Criticality Dosimetry System**

Author: **C. N. Wright, J. E. Hoy, W. F. Splichal, Jr.**

Contractual Origin: AT(07-2)-1

Present Classification: **Unclassified**

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by C. N. Wright, J. E. Hoy, and W. F. Splichal, Jr.

In September 1960, the "emergency dosimetry" capabilities of the Savannah River Plant were reported by Hoy in DP-472. This report described a criticality neutron dosimeter which was unique at that time and remains today the least expensive of several similar systems.

Last year the dosimeter and our abilities to interpret its usefulness were twice tested, utilizing the Health Physics Research Reactor at Oak Ridge National Laboratory.

The attached report updates the previous DP report, and includes in the appendix methods used in computing dose determinations. It shows the steps required - screening, preliminary dose estimates, and final dose determinations - in the event a nuclear incident involving personnel occurred.

C. M. Patterson

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Dear Mr. Erdley:

REQUEST FOR PATENT REVIEW

Please review for patent matter:

DP-1006, Savannah River Plant Criticality Dosimetry System, by C. N. Wright, J. E. Hoy, and W. F. Splichal, Jr.

If any technical clarification is needed please call J. E. Beach whose document review is attached.

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SAVANNAH RIVER PLANT CRITICALITY DOSIMETRY SYSTEM

by

Charles N. Wright
John E. Hoy
William F. Splichal, Jr.

Approved by

C. M. Patterson, Research Manager
Radiological and Environmental Sciences Division

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E. I. DU PONT DE NEMOURS & COMPANY
SAVANNAH RIVER LABORATORY
AIKEN, SOUTH CAROLINA

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ABSTRACT

Fissionable materials are handled and processed at the Savannah River Plant. Although the probability of an accidental criticality occurring is small, the possibility must be considered. This report describes the methods which would be used to segregate exposed and nonexposed personnel and to determine neutron and gamma doses received by those who were exposed.

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SAVANNAH RIVER PLANT CRITICALITY DOSIMETRY SYSTEM

INTRODUCTION

At any location where fissionable materials are processed, the possibility of a criticality accident must be considered. In the event of such an accident, it is desirable that the radiation dose received by personnel be quickly determined.

The Savannah River Plant (SRP) system of dosimetry for use after a criticality accident may be considered in three phases. The first is separation of exposed and unexposed personnel immediately after the accident. Indium foils in the film dosimeter and identification badge would be activated by neutron exposure and provide an indicator of exposed personnel. In the second phase, a preliminary estimate of neutron dose would be made by measurement of the ^{24}Na activation in the blood and body of exposed persons; the gamma dose would be indicated from film dosimeter results.

For the final phase, a more precise determination of dose would be made. This determination requires a simple, dependable dosimeter that could be worn by all employees working with fissionable material. In addition, the dosimeter would give accurate indications over a wide range of neutron and gamma doses, be lightweight, small, and convenient to wear. (Dosimeters that are mounted in fixed positions in work areas may be difficult to recover after an accident, and errors in interpretation of dose may arise because these instruments are at different locations from the exposed personnel.) The criticality neutron dosimeter (CND) was designed so that only one dosimeter would be required to determine the total dose to the wearer. Low cost was a design objective because several hundred units are required.

This report describes the techniques and instruments for the three phases of dosimetry. Emphasis is placed on the more exact measurements that are made with the criticality neutron dosimeter.

SUMMARY

In the event of a criticality accident at SRP, the radiation doses received by personnel involved would be determined by use of indium foils in the identification badges and film dosimeters, from blood sodium activation, and from analysis of the SRP Criticality Neutron Dosimeter (CND). The CND contains cadmium-shielded and unshielded indium foils, a copper foil, sulfur powder, sodium fluoride powder, and thermoluminescent dosimeters (TLD).

From determination of induced radioactivity in the foils and powders, the neutron fluence in five energy intervals may be determined. By comparing activity induced in the sodium fluoride with the activity induced in the blood of the wearer, the direction from which he was exposed may be estimated. The TLD's enable measurement of gamma exposure between 1R and 10^5R.

Appendices describe both manual and computer methods used in processing the data obtained.

DISCUSSION

SCREENING

A basic requirement of a criticality dosimetry system is the ability to screen quickly a large number of persons, in order to segregate the exposed and unexposed individuals. At SRP this screening ability is provided by 0.005-inch-thick indium foils in the identification badges and film dosimeters. The foil in the identification badge is approximately 1/2-inch square. After activation, it will cause a response of at least 100 c/m on a GM survey instrument (when placed one centimeter from the wall of the GM tube) for every Rad of neutron exposure. The foil in the film badge has four times the area of that in the identification badge with a corresponding increase in sensitivity. Because it is almost impossible that a criticality accident at SRP will involve persons not wearing an identification badge, and highly unlikely that persons not wearing a film badge would be exposed, the health physicist is able to quickly segregate exposed and unexposed persons. No dose estimates are made from the GM instrument readings.

PRELIMINARY DOSE ESTIMATES

A rough estimate (±100%) of the neutron dose received can be made by placing the probe of a GM survey instrument against the abdomen of a person, having him bend at the waist, and using the following equation:

$$\text{Dose (Rads)} = \frac{2 \times \text{cpm}}{\text{body wt in lb}}$$

where cpm is the count rate per minute (corrected for decay to time of incident) from activation of sodium in the body. Estimates by this method may be pessimistic because short-lived ^{38}Cl is also a product of neutron activation of body elements. Persons determined by these screening methods to have been exposed are checked for contamination, decontaminated if necessary (unless badly injured), and placed under

medical care. A 50 ml sample of blood is obtained from each of these people for further analysis before any sodium is administered, either through food or medical treatment. Ammonium oxalate is added to the sample to prevent coagulation. The blood is centrifuged and a 10 ml sample of plasma is decanted into a standard culture tube (15 mm x 125 mm). The tube is placed horizontally on a 3" x 3" sodium iodide scintillation counter, and the gamma ray activity between 2.6 and 3.0 MeV is determined. This energy interval is chosen to avoid interference from ^{38}Cl.

Test exposures to neutron spectra, ranging from an unmoderated fission spectrum at the ORNL Health Physics Research Reactor (HPRR)[1] to a thermal spectrum at the Standard Pile (SP) reactor at the Savannah River Plant,[2] have provided factors for converting ^{24}Na activity in blood plasma to a Rad dose (Figure 1). For first estimates of dose, an assumption of the neutron spectrum must be made from knowledge of the type of critical assembly involved. The broad characterizations used are "Fast," "Semimoderated," and "Thermal." Conversion factors used for these cases are:

Fast	1.65×10^5 Rad/μc/cc
Semimoderated	1.00×10^5 Rad/μc/cc
Thermal	0.75×10^5 Rad/μc/cc

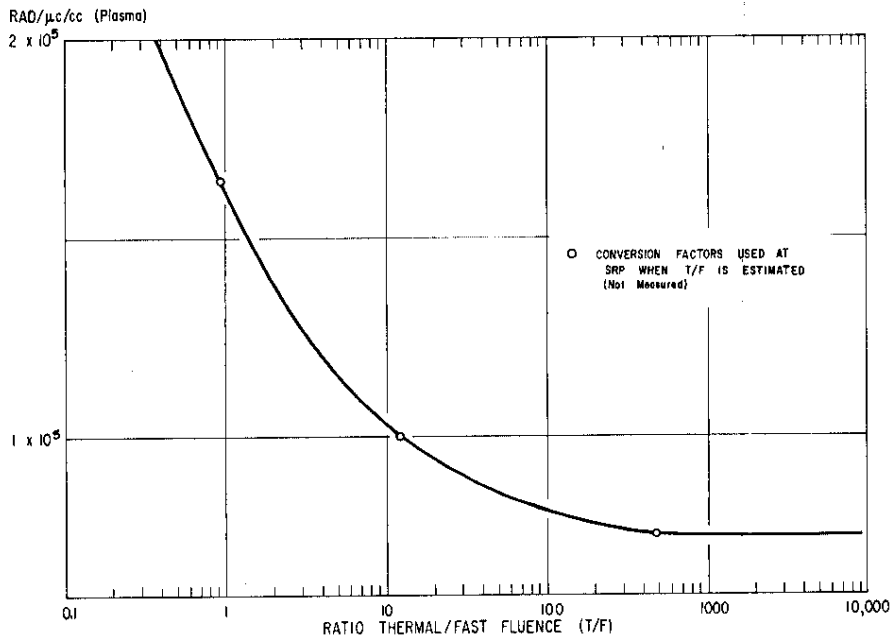

FIG. 1 EFFECT OF NEUTRON ENERGY ON BLOOD SODIUM ACTIVATION

After the CND is processed, a better definition of the spectrum will be available and the dose estimate can be refined.

Preliminary gamma dose measurements are obtained from the film dosimeter. The front face, containing the indium foil is removed from the film badge (Figure 2) as soon as practicable after the accident so that the film will not be exposed excessively by the indium. The film from the badge is pencil-marked in the darkroom to retain identity, in case the X-rayed identification has been obliterated by the exposure, and then is processed normally. Because of silver activation and other neutron effects, the indicated gamma reading may be high. Experimental data indicate that Du Pont type 1290 dosimeter film is not more than 7 percent sensitive to fast neutrons. (If the film is exposed to 100 Rad of fast neutrons, the indicated gamma reading will be not more than 7 Rad.)

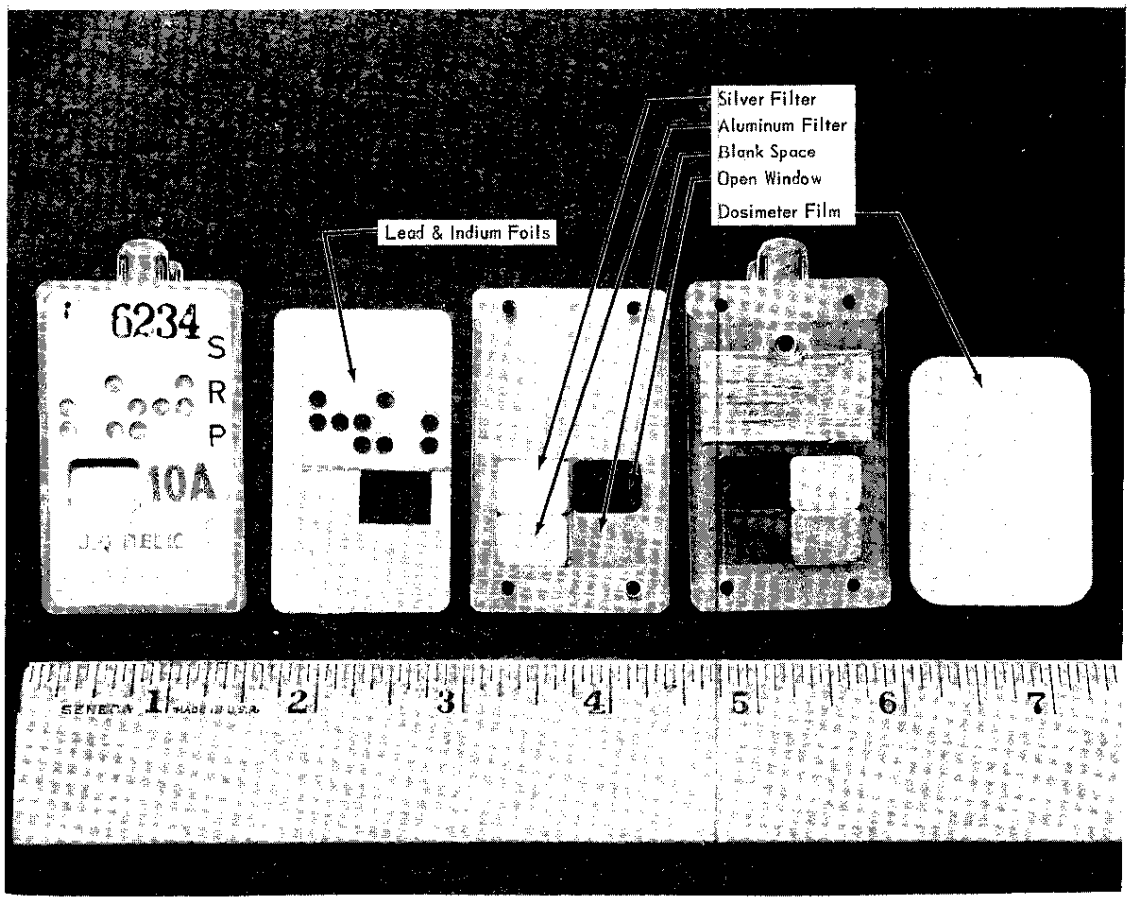

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FIG. 2 FILM BADGE

FINAL DOSE DETERMINATION

The most accurate determination of dose received would be made from the evaluation of the components contained in the Criticality Neutron Dosimeter.

DESCRIPTION OF DOSIMETER

The original model of the dosimeter and its early development were described by Hoy[s] in 1960. A diagram of the current model dosimeter is given in Figure 3. The components are housed in a "Mylar"* tube, 4-1/2 inches long and 1/2 inch in diameter, with a pocket clip attached. Indium, copper, and cadmium foils are shaped into hollow cylinders to reduce directional effects. These foils, and specific amounts of sodium fluoride and sulfur, are contained in three small polystyrene vials.

* Du Pont trademark for polyester film.

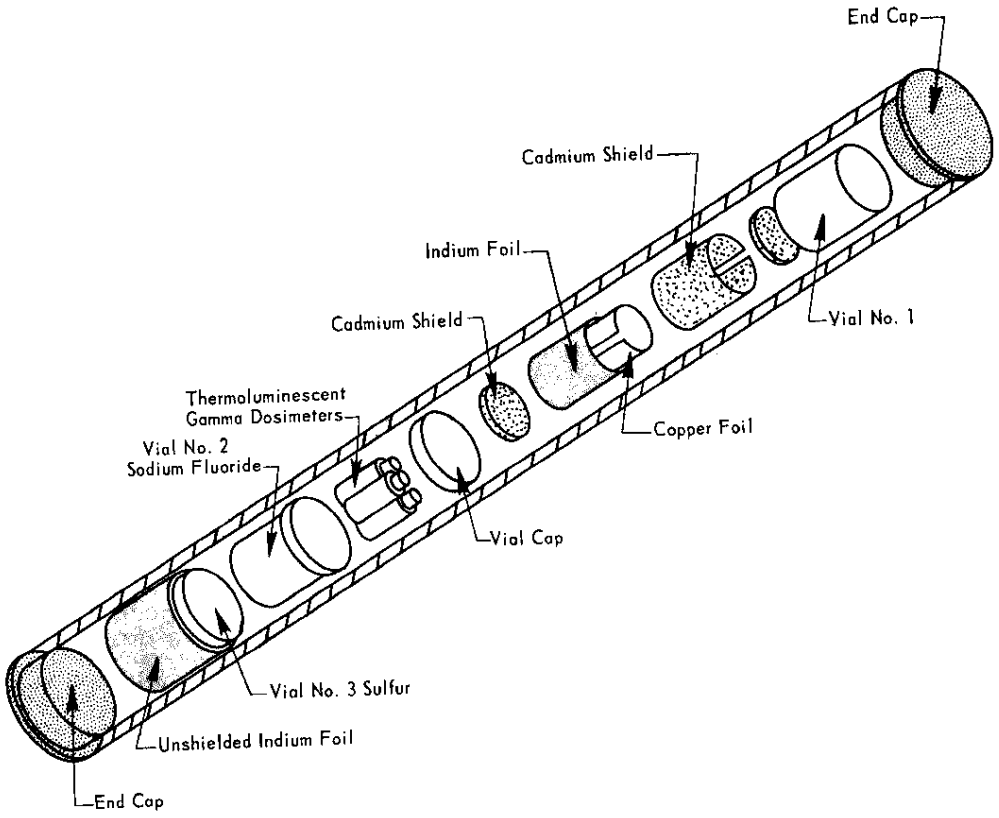

FIG. 3 SRP CRITICALITY DOSIMETER

To expedite processing of the components after an accident, all activation materials are preweighed; the exact weights are listed in each dosimeter during assembly. The thermoluminescent lithium fluoride powder is contained in three smaller polyethylene vials. Table I describes the components. Cost of the dosimeter is approximately $15.

TABLE I

Dosimeter Components

Material	Size or Weight
Cadmium (3 pieces)	1" x 5/8" x 1/32"
	3/8" diameter x 1/32" (2 pieces)
Indium (2 pieces)	15/16" x 5/8" x 0.005"
	1-7/16" x 5/8" x 0.005"
Copper	15/16" x 5/8" x 0.005"
Sulfur	1.00 gram
Sodium fluoride	1.50 grams
Lithium fluoride (3 vials)	40 mg per vial

ASSESSMENT OF NEUTRON FLUENCE

The dosimeter measures the neutron fluence (time-integrated flux density) in five energy intervals across the fission neutron spectrum. Evaluation of fluence is based on activation of indium, copper, and sulfur. These were selected primarily because they have useful activation cross-section characteristics over a desired range of neutron energies. The neutron fluence, n/cm^2, in each energy range is obtained from the following expression:

$$n/cm^2 = \frac{d/s}{\sigma_{act}\, NG\lambda e^{-\lambda t}}$$

where, N = the number of nuclei with an activation cross section, σ_{act}

G = a constant to correct the observed counting rate, d/s, to the absolute disintegration rate of the target material

t = the elapsed time in seconds between exposure and counting.

This equation assumes that exposure was short, because a criticality alarm system causes immediate evacuation of personnel from work areas. An average activation cross-section is assigned to each energy interval.

All foils and powders are counted in a lead-shielded β-γ GM Counter, for which a Ra D&E conversion factor has been obtained. The GM tube is an end-window type with a window thickness of ~2 mg/cm^2. All foils are centered on the planchet with the outside surface facing the tube window. Powders are evenly spread over the surface of the planchet before counting.

Energy Intervals of 0.0 eV to 0.5 eV, and 0.5 eV to 2 eV

Bare and cadmium-shielded indium foils are used to determine neutron fluence in these energy intervals. Indium has two stable isotopes: ^{113}In (4.28% abundance) and ^{115}In (95.72% abundance). ^{115}In (n,γ) ^{116}In is the reaction of interest. ^{116}In has three isomers, two of which have very short half-lives (2 seconds and 14 seconds); the third isomer, with a 54-minute half-life, is the principal source of radioactivity for a few hours after neutron irradiation. ^{114}In, resulting from the ^{113}In (n,γ) ^{114}In reaction, has an isomer with a 50-day half-life which is useful when a considerable delay occurs before counting.

The cross section of indium approximates a 1/v relationship from 0.025 eV to 0.3 eV, that is, the cross section decreases as neutron velocity increases. Above 0.3 eV, resonances occur, with the principal peak at 1.4 eV.

Cadmium of the thickness used in the dosimeter (0.03 inch) absorbs essentially all neutrons with energies less than 0.5 eV, the cadmium cutoff. Therefore, the difference between the activities of the bare and the cadmium-shielded indium foils is caused by neutrons in the energy interval from 0.0 eV to 0.5 eV. A factor of 1.15 is used to correct for the resonant absorption of neutrons by the cadmium shield.[4] The effective activation cross section for this interval is 145 barns. Activation of cadmium-shielded indium is used to measure the fluence in the energy interval from 0.5 eV to 2 eV. A resonance integral (corrected for flux depression) of 650 barns is assigned to this interval.

Energy Interval of 2 eV to 1 MeV

The reaction of interest for this energy interval is ^{63}Cu (n,γ) ^{64}Cu. ^{64}Cu has a 12.8-hour half-life. A competing activity resulting from ^{66}Cu (5-minute half-life) can be minimized by counting the copper no sooner than 1 hour after irradiation. The average activation cross section for ^{63}Cu in this energy range is 0.300 barn.[5] Above 1 MeV the inelastic neutron scattering cross section becomes dominant. The copper is shielded by indium and cadmium to minimize the effect of neutrons below 2 eV.

Energy Interval of 1 MeV to 2.9 MeV

The inelastic scattering reaction of fast neutrons with ^{115}In is used to determine fluence above 1 MeV. While the threshold for this reaction is about 450 keV, its cross section becomes significant at 1 MeV. The average cross section above 1 MeV is 0.180 barn. The ^{115m}In isomer has a 4.5-hour half-life. Gamma spectrometric analysis is used to distinguish the activation due to the inelastic reaction from the activation by the epithermal neutrons. The cadmium-shielded indium foil is used for this measurement.

The cross section for this reaction remains significant to values above 5 MeV. Therefore, the fluence of neutrons above 2.9 MeV, as determined from the activation of sulfur, must be subtracted from the fluence calculated from indium activation. The difference represents the fluence in the range from 1 MeV to 2.9 Mev.

Energy Interval above 2.9 MeV

The threshold reaction, ^{32}S (n,p) ^{32}P, is used to measure fluence above 2.9 MeV. Sulfur has a relatively high and constant cross section, and ^{32}P has a conveniently long half-life.

ORIENTATION OF DOSIMETER

Since the dosimeter is worn on the front of a person and the activation of the dosimeter is affected by the body's moderation and shielding, it is necessary to determine whether or not the wearer faced the neutron source in order to correct the fluence values. The correction is made by comparing the ^{24}Na in the wearer's blood with that in the sodium fluoride of the dosimeter. Blood sodium activation is relatively independent of body orientation to the source but is influenced by the energy of the neutrons. Activation of the sodium in the dosimeter is affected by both the body orientation and the neutron spectrum.

To determine the relationship between the activation of the sodium (in the dosimeter and blood), the neutron energy, and the orientation of the dosimeter, polyethylene phantoms were exposed to neutrons from a graphite-moderated reactor and an unmoderated critical assembly. The phantoms contained sodium chloride solution of the same concentration as in human blood. Dosimeters were attached to the front (surface facing the source), side, and back of the phantom. From these experiments, the following relationships were empirically established:

Case 1. When the source is to the side of the wearer

$$Y = \frac{^{24}\text{Na per gram of sodium in NaF}}{^{24}\text{Na per gram of sodium in blood}} = 0.14 \left[\frac{\text{thermal fluence}}{\text{fast fluence}}\right]^{0.61}, \text{ where}$$

the thermal fluence and the fast fluence (greater than 2.9 MeV) were those values determined by the dosimeter (uncorrected for direction).

Case 2. When the wearer is facing the source

$$\frac{^{24}\text{Na per gram of sodium in NaF}}{^{24}\text{Na per gram of sodium in blood}} > Y + 0.3Y$$

Case 3. When the wearer is facing away from the source

$$\frac{^{24}\text{Na per gram of sodium in NaF}}{^{24}\text{Na per gram of sodium in blood}} < Y - 0.3Y$$

The following factors are used to correct the fluence for the effects of body moderation and shielding.

Energy Range	Side Exposure	Wearer Facing Source	Wearer Facing Away from Source
0.0 eV - 0.5 eV	1	0.23	0.64
0.5 eV - 2 eV	1	0.35	1.6
2 eV - 1 MeV	1	0.45	2.0
1 MeV - 2.9 MeV	1	0.79	5.4
Above 2.9 MeV	1	0.72	6.7

DETERMINATION OF NEUTRON DOSE

The dose equivalent is the product of absorbed dose (Rad), quality factor, dose distribution factor, and other modifying factors. The quality factor for criticality neutron exposures has not been established. For this reason the neutron dose is given in Rads.

Snyder's[6] calculations were used to determine the tissue dose at a 5 cm depth, the depth of blood-forming organs. From these calculations, the relationship between Rad/(n)(cm^2) and neutron energy was

established. An average dose conversion factor was selected for each neutron energy interval (Table II).

TABLE II

Average Dose Conversion Factors

Energy Interval	Rad/(n)(cm^2)
Thermal to 0.5 eV	0.16×10^{-9}
0.5 eV to 2.0 eV	0.22×10^{-9}
2.0 eV to 1 MeV	1.25×10^{-9}
1 MeV to 2.9 MeV	2.87×10^{-9}
2.9 MeV and above	4.80×10^{-9}

Manual calculation of the dose is lengthy and involved (see Appendix I); however, an electronic computer program which accepts counting data, decay and exposure times, and weights of activated materials is available (see Appendix II). The computer calculates the orientation of the dosimeter, fluence and dose in the five energy intervals, and total dose.

If a dose value other than that at 5 cm depth is desired, suitable factors for converting fluence to dose for each spectrum segment may be applied to the fluence values which are obtained.

GAMMA DOSE

Lithium fluoride is used to measure the gamma dose from 25 mR to 10^6 R. Three polyethylene vials, each containing 40 mg of LiF, are included in the dosimeter. One of the LiF vials is used to establish the correct range setting for the electronic reader, and the other two vials are used for duplicate dose measurements. LiF, enriched to 99.91% ^{7}Li, is relatively insensitive to neutrons. A 100 Rad neutron dose, primarily fast neutrons, resulted in an apparent gamma dose of 7 R.

Gamma dose results are corrected for the direction of exposure as follows:

Front Exposure	0.7
Side Exposure	1.0
Back Exposure	2.1

Tests of Dosimeter

Several tests of the dosimetry system have been made to determine the quality of performance. The first of the tests was made at Los Alamos using the Godiva critical assembly as the neutron source. In this test, pairs of dosimeters were mounted on the outside of a 6-1/2 gallon cylindrical polyethylene container filled with an aqueous solution of sodium chloride having the same sodium concentration as human blood. Arrangement of the components around the Godiva assembly is shown in Figure 4. Data in Table III indicate that the maximum deviation of the dosimeter from the calculated neutron doses was 12%, with an average deviation of 6.8%. The orientation or position of the dosimeter was correctly calculated for over 80% of the dosimeters.

TABLE III

Results of Dosimeter Test at Godiva[(a)]

Position on Phantom	Computer Calculated Position	Dosimeter Results (Neutron Dose), Rad
Front	Front	659
Front	Front	683
Side	Side	759
Side	Front	747
Back	Back	752
Back	Back	792
Not on Phantom		746
		779
Not on Phantom		757
		669

(a) LASL calculated neutron dose was 708 Rad.

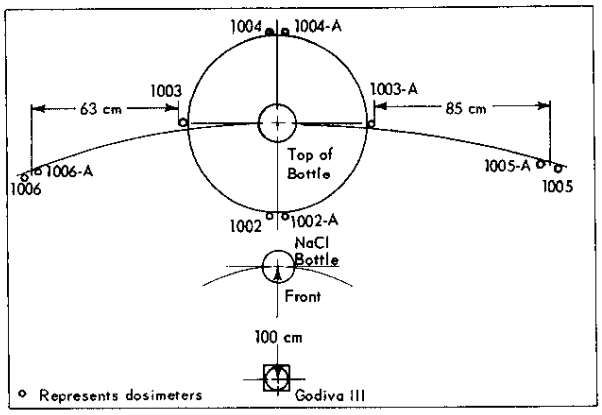

FIG. 4 ARRANGEMENT OF COMPONENTS - GODIVA

Another test of the system was made at the Health Physics Research Reactor (HPRR) at Oak Ridge National Laboratory. Its primary purpose was to investigate the validity of using salt-solution-filled polyethylene containers as phantoms. In this test pigs were used as phantoms. The thickness of their bodies and the salt content of their blood closely approximates these characteristics in man. Pigs and containers were paired as shown in Figure 5. The average deviation between the doses was 13%, as measured by the dosimeters attached to the containers and pigs.

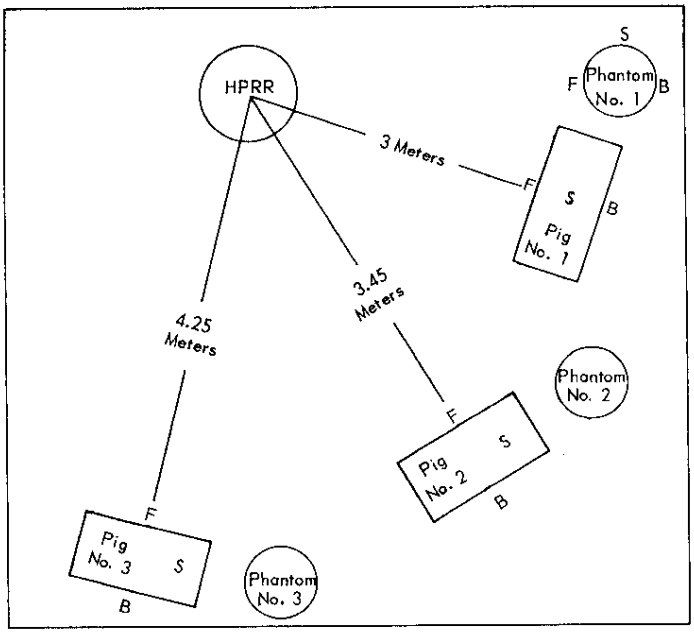

FIG. 5 ARRANGEMENT OF COMPONENTS - HPRR

The entire dosimetry system was tested at HPRR. The dosimeters were exposed, quickly returned to the Savannah River Plant, and processed according to procedures. The people who evaluated the doses had no knowledge of the test conditions. There was close agreement between the results of duplicate dosimeters at each position, and the correct orientation was calculated in the majority of the cases (Table IV). Where errors were made in the calculation of orientation, a front exposure was selected instead of a side exposure. The difference in dose correction factors for these two positions is usually comparatively

small. These tests showed that the dosimetry system can evaluate doses received by up to 20 workers within six hours after exposure, and can provide a preliminary estimate within one hour.

TABLE IV

Results of HPRR Test

Dosimeter Position	Computer Calculated Position		Dosimeter Results (Neutron Dose), rad	
	Dosimeter A[a]	Dosimeter B[a]	A	B
Phantom No. 1				
Front	Front	Front	904	862
Side	Side	Front	722	753
Back	Back	Back	856	864
Phantom No. 2				
Front	Front	Front	598	568
Side	Front	Front	396	498
Back	Back	Back	624	554
Phantom No. 3				
Front	Front	Front	391	400
Side	Side	Front	373	461
Back	Back	Back	410	484

(a) Two dosimeters were located at each position.

In a more recent test, the criticality dosimeters used at seven nuclear energy facilities in the USA were compared. The dosimeters were exposed to a burst from the HPRR, and the air dose was measured. The SRP results for neutron dose were within three percent[7] of the average of all results.

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APPENDIX I

MANUAL CALCULATION OF CND RESULTS

In the calculations that follow, the same identifying terminology is used in both the computer and manual methods to facilitate comparing the methods.

If the indium foils are counted less than 10 hours after activation, the 54-minute ^{116}In is used for determination. The correction factors used in this case are:

$$DK = (1 - e^{(-2.16 \times 10^{-4} T_1)})\ 60 \quad \text{(Buildup Correction)}$$

$$DK1 = e^{(-0.0128 \times T_{21})} \quad \text{(Decay Correction for Bare Indium Foil)}$$

$$DK2 = e^{(-0.0128 \times T_{22})} \quad \text{(Decay Correction for Cd-In Foil)}$$

$$CONST = 0.648 \quad \text{(Counter Factor)}$$

$$ECON = 0.102 \quad \text{(Counter Factor)}$$

where, T_1 = estimated exposure time in seconds

T_{21} = decay time in minutes from time of exposure to time bare indium foil was counted.

T_{22} = decay time in minutes from time of exposure to time cadmium-covered indium foil was counted.

If the indium foils are counted after more than 20 hours have elapsed since activation, the 50-day ^{114}In is used for the determination. In the interval between 10 hours after exposure and 20 hours after exposure, a mixture of the two isotopes is present, and inaccurate results may be obtained if the foils are counted during this time interval.

The constants used for the ^{114}In calculations are:

$$DK = (1 - e^{(-1.6 \times 10^{-7})T_1})\ 60$$

$$DK1 = e^{(-9.62 \times 10^{-6})T_{21}}$$

$$DK2 = e^{(-9.62 \times 10^{-6})T_{22}}$$

$$CONST = 0.0112$$

$$ECON = 7.1$$

Thermal

The thermal flux is calculated using the disintegration rates of the cadmium-shielded indium and bare indium foils as follows:

$$\text{FXTH} = \text{Thermal flux, } n/(cm^2)(sec) = \frac{\frac{(DPM1)}{(DK)}\frac{(WTCI)}{(WTBI)} - \frac{(DPM2)}{(DK)}\frac{(1.15)}{(DK2)}}{(CONST)\ (WTCI)},$$

where, DPM1 = net activity (d/m) of the bare indium foil, not corrected for decay.

DPM2 = net activity (d/m) of the cadmium-covered indium foil, not corrected for decay.

WTBI = weight of bare indium foil in grams.

WTCI = weight of cadmium-covered indium foil in grams.

Epithermal (0.4 eV to 2 eV)

The epithermal flux is calculated from the activity of the cadmium-covered indium foil as follows:

$$\text{EPTHF} = \text{Epithermal flux, } n/(cm^2)(sec) = \frac{(DPM2)\ \ (ECON)}{(WTCI)\ (DK)\ (DK2)}$$

Resonance (2 eV to 1 MeV)

The resonance flux is calculated from the activity of the copper foil as follows:

$$\text{FXCU} = \text{Resonance flux, } n/(cm^2)(sec) = \frac{(DPM4)\ (61.3)}{(WTCU)(1-e^{(-1.5 \times 10^{-5})T_1})(e^{(-9.02 \times 10^{-4})T_{24}})}$$

where, DPM4 = net activity (d/m) of copper foil, not corrected for decay.

WTCU = weight of copper foil in grams.

T^{24} = decay time in minutes from time of exposure to time copper foil was counted.

T_1 = exposure time in seconds.

Fast (3.0 to 10 MeV)

The fast flux is determined from activation of the sulfur powder in the CND as follows:

$$FXS = \text{Fast Flux} = \frac{(DPM5)\ (4.30)}{(WTS)\ (1-e^{(-0.561 \times 10^{-6})T_1})\ (e^{(-3.37 \times 10^{-5})T_{25}})}$$

where, DPM5 = net activity (d/m) of sulfur powder, not corrected for decay.

WTS = weight of sulfur powder in grams.

T_1 = exposure time in seconds.

T_{25} = decay time in minutes from time of exposure to time sulfur was counted.

Medium Energy (1 - 3 MeV)

The medium and high energy flux is calculated from the activation of the ^{115m}In isomer (4.5-hour half-life) in the cadmium-covered indium foil, as determined by pulse height analysis. The calculation below determines the total flux above 1 MeV.

FXINM = Medium and high energy flux =

$$\frac{(DPM6)\ (192.0)}{(WTBI)\ (1-e^{(-4.28 \times 10^{-5})T_1})\ (e^{(-0.154)T_{26}})}$$

where DPM6 = c/m in channels 31-36 not corrected for decay, but corrected for interference by other indium isotopes.

T_{26} = Decay time in hours from time of exposure to time foil was counted.

To determine the medium energy flux, subtract the fast flux (above 3 MeV) as obtained by sulfur counting from the fast and medium flux (above 1 MeV), as obtained above. The resulting difference is the medium energy flux.

Determination of Directional Corrections

Since the activity of the foils is influenced by the relative positions of the wearer's body and the exposing source, i.e., whether

he had his back to the source or was facing it, it is necessary to determine this relationship so that appropriate correction factors may be applied. This is done by calculating the ratio of the ^{24}Na activation in the sodium fluoride powder of the CND and the ^{24}Na activity in a sample of the wearer's blood as follows:

$$DPGNF = \frac{(DPM3)\ (1.82)}{(WTNAF)\ (e^{(-0.75 \times 10^{-3})T_{23}})} \qquad (1)$$

where, DPM3 = net activity (d/m) of sodium fluoride powder not corrected for decay.

WTNAF = weight of the sodium fluoride powder in grams.

T_{23} = decay time in minutes from time of exposure to time sodium fluoride was counted.

$$DPGBS = (UCPCC)\ (0.694 \times 10^{9}) \qquad (2)$$

where, UCPCC = $\mu c/cc$ of ^{24}Na found in the serum of the wearer's blood - corrected for decay.

$$RATIO = \frac{DPGNF}{DPGBS} \qquad (3)$$

Ideally, this ratio would be unity if the sodium fluoride vial was exposed to the same flux as the body of the wearer (side exposure); the vial activity would be higher if it were closer to the source than the wearer (front exposure); and would be lower if the vial were on the opposite side of the person from the source (rear exposure).

In actuality, this ratio is also affected by the relative "hardness," energy-wise, of the spectrum, so that the predicted "ideal" ratio for the spectrum being measured must be calculated from the thermal and fast fluxes as follows:

$$Y = 0.14\ \left(\frac{\text{Thermal flux}}{\text{Fast flux}}\right)^{0.61}$$

where, Y = the predicted ratio.

To determine direction, compare the RATIO previously computed with (Y ±0.3Y). If the RATIO falls within these limits, the exposure is called a "side" exposure, and no directional corrections will be made in calculating dose, i.e., all correction factors are 1.0.

If the RATIO is greater than $(Y + 0.3Y)$, the majority of the dose was received from the front. The following correction factors will be used in dose computation in this case:

CT = Thermal correction = 0.23

CET = Epithermal correction = 0.35

CCU = Resonance correction = 0.45

CM = Medium energy correction = 0.79

CS = Fast correction = 0.72

If the RATIO is less than $(Y - 0.3Y)$, the majority of the dose was received from the rear. In this case, the correction factors are:

CT = Thermal correction = 0.64

CET = Epithermal correction = 1.6

CCU = Resonance correction = 2.0

CM = Medium energy correction = 5.4

CS = Fast correction = 6.7

Dose Calculation

Thermal dose (THRAD) = (Thermal flux) (CT) (0.17×10^{-9}) (T_1)

Epithermal dose (ETRAD) = (Epithermal flux) (CET) (0.22×10^{-9}) (T_1)

Resonance dose (CURAD) = (Resonance flux) (CCU) (1.25×10^{-9}) (T_1)

Medium Energy dose (RADIN) = (Medium energy flux) (CM) (2.87×10^{-9}) (T_1)

High Energy dose (SRAD) = (Fast flux) (CS) $(3.6 \times 10^{-9})(T_1)$

TOTAL = THRAD + ETRAD + CURAD + RADIN + SRAD = entire neutron dose in Rads.

APPENDIX II

FORTRAN PROGRAM FOR COMPUTATION OF DOSE FROM CND DATA

```
SEQ    STMNT      FORTRAN STATEMENT

  1               PRINT 60
  2               PRINT 61
  3               PRINT 66
  4               PRINT 67
  5               PRINT 64
  6        82     IND=0
  7               PRINT 81
  8         1     READ 50, IDEN
  9               IF(IDEN)83,83,84
 10        84     READ 74,T1
 11               READ 72,WTBI
 12               READ 51,T21
 13               READ 74,DPM1
 14               READ 72,WTCI
 15               READ 51,T22
 16               READ 74,DPM2
 17               READ 72,WTNAF
 18               READ 51,T23
 19               READ 74,DPM3
 20               READ 72,WTCU
 21               READ 51,T24
 22               READ 74,DPM4
 23               READ 72, WTS
 24               READ 51, T25
 25               READ 74, DPM5
 26               READ 74, UCPCC
 27               READ 72,T26
 28               READ 74, DPM6
 29               READ 72,WTINM
 30               IF(SENSE SWITCH 2) 5,6
 31         5     CAMS=1.6E-07
 32               CAMM=9.62E-06
 33               CONST=.0112
 34               ECON=7.1
 35               GO TO 7
 36         6     CAMS=2.16E-04
 37               CAMM=.0128
 38               CONST=.648
 39               ECON=.102
 40         7     DK=(1.-(1./EXPF(CAMS*T1)))*60.
 41               DK1=1./EXPF(CAMM*T21)
 42               DK2=1./EXPF(CAMM*T22)
 43               EXTH=((DPM1*WTCI/(DK*DK1*WTBI))-(DPM2*1.15/(DK*DK2)))/(CONST*WTCI)
 44               DKS=(1.-1./EXPF(.561E-06*T1))*(1./EXPF(3.37E-05*T25))
 45               EXS=DPM5*4.30/(WTS*DKS)
```

```
SEQ    STMNT      FORTRAN STATEMENT

 46               EPTHF=DPM2*ECON/(WTCI*DK*DK2)
 47               DKCU=(1.-1./EXPF(1.5E-05*T1))*(1./EXPF(.000902*T24))
 48               FXCU=DPM4*61.3/(DKCU*WTCU)
 49               DKINM=(1.-(1./EXPF((4.28E-05)*T1)))*(1./EXPF((1.54E-01)*T26))
 50               FXINM=DPM6*192.0/(WTINM*DKINM)
 51               DPGNF=DPM3*1.82/(WTNAF*(1./EXPF(.77E-03*T23)))
 52               DPGBS=UCPCC*.694E09
 53               RATIO=DPGNF/DPGBS
 54               Y=(0.14)*((FXTH/FXS)**.61)
 55               PRINT 53, IDEN
 56               PRINT 80,RATIO,Y
 57               Y1=Y+0.3*Y
 58               Y2=Y-0.3*Y
 59               IF(RATIO-Y1)3,3,13
 60         3     IF(RATIO-Y2)11,11,12
 61        11     CT=.64
 62               CET=1.6
 63               CCU=2.0
 64               CM=5.4
 65               CS=6.7
 66               PRINT 54
 67               GO TO 20
 68        12     CT=1.0
 69               CET=1.0
 70               CCU=1.0
 71               CS=1.0
 72               CM=1.0
 73               PRINT 55
 74               GO TO 21
 75        13     CT=.23
 76               CET=.35
 77               CCU=.45
 78               CM=.79
 79               CS=.72
 80               PRINT 56
 81        20     IF(SENSE SWITCH1)12,21
 82        21     TFXTH=FXTH*CT*T1
 83               TFXS=FXS*CS*T1
 84               THTFR=TFXTH/TFXS
 85               TFXTH=FXTH*T1
 86               TEPTH=EPTHF*T1
 87               TFXS=FXS*T1
 88               TFXCU=FXCU*T1
 89               TFINM=FXINM*T1
 90               TINMS=TFINM-TFXS
 91               TFLUX=TFXTH+TEPTH+TFXCU+TINMS+TFXS
 92               THRAD=TFXTH*CT*0.16E-09
 93               ETRAD=TEPTH*CET*0.22E-09
 94               CURAD=TFXCU*CCU*1.25E-09
 95               RADIN=TINMS*CM*2.87E-09
 96               SRAD=TFXS*CS*4.8E-09
 97               TOTAL=THRAD+ETRAD+CURAD+RADIN+SRAD
```

```
SEQ    STMNT      FORTRAN STATEMENT

 98               PRINT 30,TFXTH,THRAD
 99               PRINT 31, TEPTH,ETRAD
100               PRINT 32,TFXCU,CURAD
101               PRINT 33, TINMS, RADIN
102               PRINT 34,TFXS,SRAD
103               PRINT 35,TFLUX,TOTAL
104               PRINT 36, THTFR
105               IND=IND+1
106               IF(IND-3)1,82,82
107       83      PRINT 85
108               PRINT 81
109               PAUSE
110       30      FORMAT(15H THERMAL        ,E9.2,F9.3)
111       31      FORMAT(15H EPITHERMAL     ,E9.2,F9.3)
112       32      FORMAT(15H 2EV TO 1MEV    ,E9.2,F9.3)
113       33      FORMAT(15H 1MEV TO 3MEV   ,E9.2,F9.3)
114       34      FORMAT(15H ABOVE 3MEV     ,E9.2,F9.3)
115       35      FORMAT(1X/15H TOTAL          ,E9.2,F9.3)
116       36      FORMAT(1X//16H THERMAL/FAST   ,F9.2)
117       50      FORMAT (I4)
118       51      FORMAT(F6.0)
119       53      FORMAT(1X///15H PAYROLL NO.    ,I5)
120       54      FORMAT(1X/14H REAR EXPOSURE)
121       55      FORMAT(1X/14H SIDE EXPOSURE)
122       56      FORMAT(1X/15H FRONT EXPOSURE)
123       60      FORMAT(30H1CND DOSE CALCULATOR NOV. 1965)
124       61      FORMAT(21H PROGRAM INSTRUCTIONS)
125       64      FORMAT(45H INDIUM DECAY MORE THAN 20 HOURS  SWITCH C ON)
126       66      FORMAT(45H SWITCH B ON TO IGNORE DIRECTIONAL CORRECTION)
127       67      FORMAT(39H (CALCULATED DIRECTION WILL BE PRINTED))
128       72      FORMAT(F8.2)
129       74      FORMAT(E9.3)
130       80      FORMAT(7H RATIO E10.3,3X2HY E10.3)
131       81      FORMAT(1H1)
132               END
133       85      FORMAT(4H1EOJ)
```

INPUT DATA TO
FORTRAN PROGRAM FOR COMPUTATION
OF DOSE FROM CND DATA

Entry No.	Fortran Format	Description	Example
1	I4	Identification or badge No.	0028
2	E9.3	Exposure time, seconds (may be an estimate)	4.320E+02
3	F8.2	Weight of bare indium foil, gram	00000.49
4	F6.0	Decay time before counting bare In foil, minutes	00462.
5	E9.3	Net activity of bare indium foil,(a) d/m	1.040E+05
6	F8.2	Weight of Cd-covered In foil, gram	00000.34
7	F6.0	Decay time before counting Cd-In foil, minutes	00465.
8	E9.3	Net activity of Cd-In foil,(a) d/m	2.140E+04
9	F8.2	Weight of sodium fluoride powder, gram	00000.99
10	F6.0	Decay time before counting NaF, minutes	00507.
11	E9.3	Net activity of NaF powder,(a) d/m	3.570E+04
12	F8.2	Weight of copper foil, gram	00000.44
13	F6.0	Decay time before counting copper foil, minutes	00599.
14	E9.3	Net activity of copper foil,(a) d/m	2.330E+03
15	F8.2	Weight of sulfur powder, gram	00001.
16	F6.0	Decay time before counting sulfur powder, minutes	01313.
17	E9.3	Net activity of sulfur powder,(a) d/m	9.400E+02
18	E9.3	^{24}Na in blood sample at t = 0, μc/cc	4.721E-04
19	F8.2	Decay time(b) before counting for ^{115m}In, hours	000025.92
20	E9.3	Net activity of ^{115m}In, channels 31-36, c/m	7.920E+01
21	F8.2	Weight of In foil used for ^{115}In measurement, gram	00000.49

(a) (d/m = net c/m X CF (Ra D&E), not corrected for decay.
(b) Decay time for this item only is in HOURS; all others are in minutes.